ATTACHMENT B

MARKED-UP TSUP PAGES
DRESDEN AND QUAD CITIES NUCLEAR POWER STATIONS
LICENSE NOS. DPR-19, DPR-25, DPR-29 AND DPR-30

Issue	Page	Applicable Plant
10, 31	1-2	Dresden & Quad Cities
31	1-3	Dresden & Quad Cities
10, 31	1-4	Dresden & Quad Cities
10	1-8	Dresden & Quad Cities
10	2-1	Dresden & Quad Cities
10	2-2	Dresden & Quad Cities
15	3/4.0-1	Dresden & Quad Cities
5	3/4.0-2	Dresden & Quad Cities
5	3/4.0-3	Dresden & Quad Cities
32, 31	3/4.1-1	Dresden & Quad Cities
17	3/4.1-7	Dresden & Quad Cities
20	3/4.1-9	Dresden & Quad Cities
17	3/4.1-10	Dresden & Quad Cities
31	3/4.2-1	Dresden & Quad Cities
24	3/4.2-3	Dresden & Quad Cities
48	3/4.2-4	Dresden
48	3/4.2-7	Dresden
29, 22	3/4.2-8	Dresden
22	3/4.2-8	Quad Cities
10, 29	3/4.2-9	Dresden
29	3/4.2-9	Quad Cities
22	3/4.2-10	Dresden & Quad Cities
31	3/4.2-11	Dresden & Quad Cities
36	3/4.2-14	Dresden
36	3/4.2-15	Dresden
29	3/4.2-18	Dresden
19, 31	3/4.2-21	Dresden & Quad Cities
19	3/4.2-22	Dresden & Quad Cities
31	3/4.2-25	Dresden & Quad Cities
30	3/4.2-26	Dresden
29	3/4.2-27	Dresden
10	3/4.2-27	Quad Cities
10, 44	3/4.2-29	Dresden
22, 44, 10	3/4.2-30	Dresden
10, 44	3/4.2-30	Quad Cities
22, 43	3/4.2-31	Dresden
10, 2, 44, 22	3/4.2-31	Quad Cities
22, 44	3/4.2-32	Quad Cities
2, 10, 22, 44	3/4.2-33	Dresden

ATTACHMENT B
(continued)

MARKED-UP TSUP PAGES
DRESDEN AND QUAD CITIES NUCLEAR POWER STATIONS
LICENSE NOS. DPR-19, DPR-25, DPR-29 AND DPR-30

Issue	Page	Applicable Plant
22, 17, 29	3/4.2-34	Dresden
2, 44	3/4.2-34	Quad Cities
22, 17, 29	3/4.2-35	Dresden & Quad Cities
22, 17, 44	3/4.2-36	Dresden
10, 22, 17, 29	3/4.2-36	Quad Cities
17	3/4.2-37	Quad Cities
22, 10	3/4.2-38	Dresden
10	3/4.2-39	Dresden
22	3/4.2-39	Quad Cities
10	3/4.2-40	Dresden & Quad Cities
22, 29	3/4.2-41	Dresden
22	3/4.2-42	Dresden
22	3/4.2-42	Quad Cities
2, 10	3/4.2-43	Dresden
22	3/4.2-43	Quad Cities
10	3/4.2-44	Dresden
2	3/4.2-44	Quad Cities
10	3/4.2-45	Quad Cities
10	3/4.2-46	Dresden
31	3/4.2-47	Dresden
10	3/4.2-47	Quad Cities
44	3/4.2-48	Dresden
31	3/4.2-48	Quad Cities
10, 44	3/4.2-49	Quad Cities
31	3/4.2-50	Dresden
44	3/4.2-51	Dresden
31	3/4.2-51	Quad Cities
44	3/4.2-52	Quad Cities
10	B 3/4.2-3	Dresden & Quad Cities
29	B 3/4.2-4	Dresden & Quad Cities
1	3/4.3-1	Dresden & Quad Cities
21	3/4.3-3	Dresden & Quad Cities
10	3/4.3-6	Dresden & Quad Cities
33	3/4.3-12	Dresden
33	3/4.3-14	Dresden
30	3/4.3-18	Dresden & Quad Cities
1	B 3/4.3-1	Dresden & Quad Cities
33	B 3/4.3-5	Dresden & Quad Cities
10	B 3/4.3-7	Dresden

ATTACHMENT B
(continued)

MARKED-UP TSUP PAGES
DRESDEN AND QUAD CITIES NUCLEAR POWER STATIONS
LICENSE NOS. DPR-19, DPR-25, DPR-29 AND DPR-30

Issue	Page	Applicable Plant
30	3/4.5-1	Dresden & Quad Cities
9, 18, 31	3/4.5-2	Dresden & Quad Cities
27, 9, 18, 31, 34	3/4.5-3	Dresden
9, 18, 34	3/4.5-3	Quad Cities
34	3/4.5-4	Dresden
27, 31	3/4.5-4	Quad Cities
10, 34	3/4.5-5	Quad Cities
30, 31, 35	3/4.5-9	Dresden
9, 31, 30	3/4.5-10	Quad Cities
9	3/4.5-11	Quad Cities
28, 10	B 3/4.5-2	Dresden & Quad Cities
30, 10	B 3/4.5-3	Dresden
7	3/4.6-3	Dresden
46	3/4.6-6	Dresden
47	3/4.6-7	Dresden & Quad Cities
16, 47, 31	3/4.6-8	Dresden & Quad Cities
47	3/4.6-9	Dresden & Quad Cities
10	3/4.6-15	Quad Cities
10	3/4.6-25	Quad Cities
10	3/4.6-27	Quad Cities
46	B 3/4.6-2	Dresden
7	B 3/4.6-2	Quad Cities
10	B 3/4.6-4	Dresden
10	B 3/4.6-7	Dresden & Quad Cities
10	B 3/4.6-8	Dresden & Quad Cities
37	3/4.7-1	Dresden & Quad Cities
8	3/4.7-2	Dresden & Quad Cities
8	3/4.7-3	Dresden & Quad Cities
10	3/4.7-5	Dresden & Quad Cities
10, 29	3/4.7-12	Dresden
10, 29	3/4.7-13	Quad Cities
3	3/4.7-14	Dresden
(a)	3/4.7-15	Dresden
3	3/4.7-15	Quad Cities
(a)	3/4.7-16	Quad Cities
38	3/4.7-17	Dresden

(a) No changes. These pages are provided for continuity and for information only.

ATTACHMENT B
(continued)

MARKED-UP TSUP PAGES
DRESDEN AND QUAD CITIES NUCLEAR POWER STATIONS
LICENSE NOS. DPR-19, DPR-25, DPR-29 AND DPR-30

Issue	Page	Applicable Plant
38	3/4.7-18	Quad Cities
37	3/4.7-20	Dresden
37	3/4.7-21	Quad Cities
39	3/4.7-23	Dresden
10, 39	3/4.7-24	Dresden
39	3/4.7-24	Quad Cities
10, 39	3/4.7-25	Quad Cities
3	B 3/4.7-4	Dresden & Quad Cities
37	B 3/4.7-7	Dresden & Quad Cities
39	B 3/4.7-8	Dresden & Quad Cities
11	3/4.8-1	Dresden & Quad Cities
11	3/4.8-3	Dresden
23	3/4.8-6	Dresden & Quad Cities
10	3/4.8-14	Dresden
10	3/4.8-17	Dresden & Quad Cities
30	3/4.8-23	Dresden & Quad Cities
42	3/4.8-24	Quad Cities
23, 11	B 3/4.8-1	Dresden
23	B 3/4.8-1	Quad Cities
23	B 3/4.8-2	Dresden & Quad Cities
30	B 3/4.8-4	Dresden & Quad Cities
4	3/4.9-1	Dresden & Quad Cities
4	3/4.9-2	Dresden & Quad Cities
4	3/4.9-3	Dresden & Quad Cities
4	3/4.9-4	Dresden & Quad Cities
4	3/4.9-5	Dresden & Quad Cities
4	3/4.9-6	Dresden & Quad Cities
4	3/4.9-7	Quad Cities
4, 10	3/4.9-8	Dresden & Quad Cities
4	3/4.9-9	Dresden & Quad Cities
6	3/4.9-12	Dresden & Quad Cities
6	3/4.9-13	Dresden & Quad Cities
6	3/4.9-14	Dresden & Quad Cities
30	3/4.9-17	Dresden & Quad Cities
10	3/4.9-18	Dresden & Quad Cities
4, 10	B 3/4.9-1	Dresden & Quad Cities
4, 10	B 3/4.9-3	Dresden & Quad Cities
10, 4	B 3/4.9-7	Dresden & Quad Cities
40	3/4.10-10	Dresden & Quad Cities

ATTACHMENT B
(continued)

MARKED-UP TSUP PAGES
DRESDEN AND QUAD CITIES NUCLEAR POWER STATIONS
LICENSE NOS. DPR-19, DPR-25, DPR-29 AND DPR-30

Issue	Page	Applicable Plant
12	3/4.11-2	Dresden
41	3/4.11-3	Dresden & Quad Cities
12	3/4.11-4	Dresden
12	3/4.11-5	Dresden
12, 10	B 3/4.11-1	Dresden
10	B 3/4.11-1	Quad Cities
12, 41, 10	B 3/4.11-2	Dresden
10, 4	B 3/4.11-2	Quad Cities
12	B 3/4.11-3	Dresden
13	6-3	Dresden
10	6-9	Dresden
45	6-19	Dresden

(5) CECo, pursuant to the Act and 10 CFR Parts 30, ~~40,~~ and 70, to possess, but not separate, such byproduct special nuclear materials as may be produced by the operation of ~~the facility~~. Dresden Nuclear Power Station, Units Nos. 1, 2 and 3.

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 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

The licensee is authorized to operate the facility at steady state reactor core power levels not in excess of 2527 megawatts thermal (100 percent rated power) in accordance with the conditions specified herein.

(2) Technical Specifications

Am. 137
07/27/95

The Technical Specifications contained in Appendix A, as revised through Amendment No. 137, are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

(3) Operation in the coastdown mode is permitted to 40% power.

(4) The valves in the equalizer piping between the recirculation loop shall be closed at all times during reactor operation.

(5) The licensee shall maintain the commitments made in response to the March 14, 1983, NUREG-0737 Order, subject to the following provision:

The licensee may make changes to commitments made in response to the March 14, 1983, NUREG-0737 Order without prior approval of the Commission as long as the change would be permitted without NRC approval, pursuant to the requirements of 10 CFR 50.59. Consistent with this regulation, if the change results in an Unreviewed Safety Question, a license amendment shall be submitted to the NRC staff for review and approval prior to implementation of the change.

D. The facility has been granted certain exemptions from the requirements of Section III.G of Appendix R to 10 CFR Part 50, "Fire Protection Program for Nuclear Power Facilities Operating Prior to January 1, 1979." This section relates to fire protection features for ensuring the systems and associated circuits used to achieve and maintain safe shutdown are free of fire damage. These exemptions were granted and sent to the licensee in letters dated February 2, 1983, September 28, 1987, July 6, 1989, and August 15, 1989.

In addition, the facility has been granted certain exemptions from Sections II and III of Appendix J 10 CFR Part 50, "Primary Reactor Containment Leakage Testing for Water-Cooled Power Reactors." This section contains leakage test requirements, schedules and acceptance criteria for tests of the leak-tight integrity of the primary reactor containment and systems and components which penetrate the containment. These exemptions were granted and sent to the licensee in a letter dated June 25, 1982.

'Deleted'

Am. 62 02/06/81	J.	<u>Iodine Monitoring</u> The licensee shall implement a program which will ensure the capability to accurately determine the airborne iodine concentration in vital areas under accident conditions. This program shall include the following: 1. Training of personnel; 2. Procedures for monitoring, and 3. Provisions for maintenance of sampling and analysis equipment.
Am. 103 12/15/87	K.	Deleted
Am. 94 06/10/86	L.	<u>Post Accident Sampling</u> A program will be established, implemented, and maintained which will ensure the capability to obtain and analyze reactor coolant, radioactive iodines and particulates in plant chimney effluents, and containment atmosphere samples under accident conditions. The program shall include the following: 1. Training of personnel 2. Procedures for sampling and analysis, and 3. Provisions for maintenance of sampling and analysis equipment.

'Deleted'

Am. 128 02/31/91	4.	This license is effective as of the date of issuance, and shall expire at midnight, ~~December 14, 2012~~. October 1, 2011

Date of Issuance: December 14, 1972

k:nla:quad:amend.wpf:7

Am. 56
02/06/81

I. <u>Iodine Monitoring</u>
The licensee shall implement a program which will ensure the capability to accurately determine the airborne iodine concentration in vital areas under accident conditions. This program shall include the following:

1. Training of personnel;

2. Procedures for monitoring, and

3. Provisions for maintenance of sampling and analysis equipment.

'Deleted'

Am. 95
01/16/87

J. Deleted

Am. 90
06/10/86

K. <u>Post Accident Sampling</u>
A program will be established, implemented, and maintained which will ensure the capability to obtain and analyze reactor coolant, radioactive iodines and particulates in plant chimney effluents, and containment atmosphere samples under accident conditions. The program shall include the following:

1. Training of personnel

2. Procedures for sampling and analysis, and

3. Provisions for maintenance of sampling and analysis equipment.

'Deleted'

Am. 123
02/13/91

4. This license is effective as of the date of issuance, and shall expire at midnight, December 14, 2012.

March 31, 2012

Date of Issuance: December 14, 1972

1.0 DEFINITIONS

CHANNEL FUNCTIONAL TEST

A CHANNEL FUNCTIONAL TEST shall be:

a. Analog CHANNEL(s) - the injection of a simulated signal into the CHANNEL as close to the sensor as practicable to verify OPERABILITY including required alarm and/or trip functions and CHANNEL failure trips.

b. Bistable CHANNEL(s) - the injection of a simulated signal into the sensor to verify OPERABILITY including required alarm and/or trip functions.

The CHANNEL FUNCTIONAL TEST may be performed by any series of sequential, overlapping or total CHANNEL steps such that the entire CHANNEL is tested.

CORE ALTERATION

INSERT

~~CORE ALTERATION shall be the addition, removal, relocation or movement of fuel, sources, incore instruments or reactivity controls within the reactor pressure vessel with the vessel head removed and fuel in the vessel. Normal movement (including replacement) of the SRMs, IRMs, TIPs, LPRMs, or special movable detectors is not considered a CORE ALTERATION. Suspension of CORE ALTERATION(s) shall not preclude completion of the movement of a component to a safe conservative position.~~

CORE OPERATING LIMITS REPORT (COLR)

The CORE OPERATING LIMITS REPORT (COLR) shall be the unit specific document that provides core operating limits for the current operating cycle. These cycle specific core operating limits shall be determined for each operating cycle in accordance with Specification 6.9 ~~6.6~~. Plant operation within these operating limits is addressed in individual specifications.

CRITICAL POWER RATIO (CPR)

The CRITICAL POWER RATIO (CPR) shall be the ratio of that power in the assembly which is calculated by application of the applicable NRC approved critical power correlation to cause some point in the assembly to experience transition boiling, divided by the actual assembly power.

DOSE EQUIVALENT I-131

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

FRACTION OF RATED THERMAL POWER (FRTP)

The FRACTION OF RATED THERMAL POWER (FRTP) shall be the measured THERMAL POWER divided by the RATED THERMAL POWER.

1.0 DEFINITIONS

CHANNEL FUNCTIONAL TEST

A CHANNEL FUNCTIONAL TEST shall be:

a. Analog CHANNEL(s) - the injection of a simulated signal into the CHANNEL as close to the sensor as practicable to verify OPERABILITY including required alarm and/or trip functions and CHANNEL failure trips.

b. Bistable CHANNEL(s) - the injection of a simulated signal into the sensor to verify OPERABILITY including required alarm and/or trip functions.

The CHANNEL FUNCTIONAL TEST may be performed by any series of sequential, overlapping or total CHANNEL steps such that the entire CHANNEL is tested.

CORE ALTERATION

INSERT

CORE ALTERATION shall be the addition, removal, relocation or movement of fuel, sources, incore instruments or reactivity controls within the reactor pressure vessel with the vessel head removed and fuel in the vessel. Normal movement (including replacement) of the SRMs, IRMs, TIPs, LPRMs, or special movable detectors is not considered a CORE ALTERATION. Suspension of CORE ALTERATION(s) shall not preclude completion of the movement of a component to a safe conservative position.

CORE OPERATING LIMITS REPORT (COLR)

The CORE OPERATING LIMITS REPORT (COLR) shall be the unit specific document that provides core operating limits for the current operating cycle. These cycle specific core operating limits shall be determined for each operating cycle in accordance with Specification 6.9 ~~6.6~~. Plant operation within these operating limits is addressed in individual specifications.

CRITICAL POWER RATIO (CPR)

The CRITICAL POWER RATIO (CPR) shall be the ratio of that power in the assembly which is calculated by application of the applicable NRC approved critical power correlation to cause some point in the assembly to experience transition boiling, divided by the actual assembly power.

DOSE EQUIVALENT I-131

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

FRACTION OF LIMITING POWER DENSITY (FLPD)

The FRACTION OF LIMITING POWER DENSITY (FLPD) shall be the LHGR existing at a given location divided by the specified LHGR limit for that bundle.

INSERT

CORE ALTERATION shall be the movement of any fuel, sources, or reactivity control components, within the reactor vessel with the vessel head removed and fuel in the vessel. The following exceptions are not considered to be CORE ALTERATIONS:

a. Movement of source range monitors, local power range monitors, intermediate range monitors, traversing incore probes, or special movable detectors (including undervessel replacement); and

b. Control rod movement, provided there are no fuel assemblies in the associated control cell.

Suspension of CORE ALTERATIONS shall not preclude completion of movement of a component to a safe position.

1.0 DEFINITIONS

FREQUENCY NOTATION

The FREQUENCY NOTATION specified for the performance of Surveillance Requirements shall correspond to the intervals defined in Table 1-1.

FUEL DESIGN LIMITING RATIO (FDLRX)

The FUEL DESIGN LIMITING RATIO (FDLRX) shall be the limit used to assure that the fuel operates within the end-of-life steady-state design criteria by, among other items, limiting the release of fission gas to the cladding plenum.

FUEL DESIGN LIMITING RATIO for CENTERLINE MELT (FDLRC)

The FUEL DESIGN LIMITING RATIO for CENTERLINE MELT (FDLRC) shall be the limit used to assure that the fuel will neither experience centerline melt nor exceed 1% plastic cladding strain for transient overpower events beginning at any power and terminating at 120% of RATED THERMAL POWER.

IDENTIFIED LEAKAGE

IDENTIFIED LEAKAGE shall be: a) leakage into primary containment collection systems, such as pump seal or valve packing leaks, that is captured and conducted to a sump or collecting tank, or b) leakage into the primary containment atmosphere from sources that are both specifically located and known either not to interfere with the operation of the leakage detection systems or not to be PRESSURE BOUNDARY LEAKAGE.

LIMITING CONTROL ROD PATTERN (LCRP)

A LIMITING CONTROL ROD PATTERN (LCRP) shall be a pattern which results in the core being on a thermal hydraulic limit, i.e., operating on a limiting value for APLHGR, LHGR, or MCPR.

LINEAR HEAT GENERATION RATE (LHGR)

LINEAR HEAT GENERATION RATE (LHGR) shall be the heat generation per unit length of fuel rod. It is the integral of the heat flux over the heat transfer area associated with the unit length.

LOGIC SYSTEM FUNCTIONAL TEST (LSFT)

A LOGIC SYSTEM FUNCTIONAL TEST (LSFT) shall be a test of all required logic components, i.e., all required relays and contacts, trip units, solid state logic elements, etc, of a logic circuit, from as close to the sensor ~~through and including~~ as practicable up to, but not including the actuated device, to verify OPERABILITY. The LOGIC SYSTEM FUNCTIONAL TEST may be performed by means of any series of sequential, overlapping or total system steps ~~such~~ so that the entire logic system is tested.

MINIMUM CRITICAL POWER RATIO (MCPR)

The MINIMUM CRITICAL POWER RATIO (MCPR) shall be the smallest CPR which exists in the core.

1.0 DEFINITIONS

FRACTION OF RATED THERMAL POWER (FRTP)

The FRACTION OF RATED THERMAL POWER (FRTP) shall be the measured THERMAL POWER divided by the RATED THERMAL POWER.

FREQUENCY NOTATION

The FREQUENCY NOTATION specified for the performance of Surveillance Requirements shall correspond to the intervals defined in Table 1-1.

IDENTIFIED LEAKAGE

IDENTIFIED LEAKAGE shall be: a) leakage into primary containment collection systems, such as pump seal or valve packing leaks, that is captured and conducted to a sump or collecting tank, or b) leakage into the primary containment atmosphere from sources that are both specifically located and known either not to interfere with the operation of the leakage detection systems or not to be PRESSURE BOUNDARY LEAKAGE.

LIMITING CONTROL ROD PATTERN (LCRP)

A LIMITING CONTROL ROD PATTERN (LCRP) shall be a pattern which results in the core being on a thermal hydraulic limit, i.e., operating on a limiting value for APLHGR, LHGR, or MCPR.

LINEAR HEAT GENERATION RATE (LHGR)

LINEAR HEAT GENERATION RATE (LHGR) shall be the heat generation per unit length of fuel rod. It is the integral of the heat flux over the heat transfer area associated with the unit length.

as practicable up to, but not including

LOGIC SYSTEM FUNCTIONAL TEST (LSFT)

as close to the

A LOGIC SYSTEM FUNCTIONAL TEST (LSFT) shall be a test of all required logic components, i.e., all required relays and contacts, trip units, solid state logic elements, etc, of a logic circuit, from sensor ~~through and including~~ the actuated device, to verify OPERABILITY. The LOGIC SYSTEM FUNCTIONAL TEST may be performed by any series of sequential, overlapping or total system steps ~~such~~ that the entire logic system ~~is~~ tested.

so

means of

MAXIMUM FRACTION OF LIMITING POWER DENSITY (MFLPD)

The MAXIMUM FRACTION OF LIMITING POWER DENSITY (MFLPD) shall be the highest value of the FLPD which exists in the core.

MINIMUM CRITICAL POWER RATIO (MCPR)

The MINIMUM CRITICAL POWER RATIO (MCPR) shall be the smallest CPR which exists in the core for each class of fuel.

1.0 DEFINITIONS

OFFSITE DOSE CALCULATION MANUAL (ODCM)

The OFFSITE DOSE CALCULATION MANUAL (ODCM) shall contain the methodology and parameters used in the calculation of offsite doses resulting from radioactive gaseous and liquid effluents, in the calculation of gaseous and liquid effluent monitoring Alarm/Trip Setpoints, and in the conduct of the Environmental Radiological Monitoring Program. The ODCM shall also contain (1) the Radioactive Effluent Controls and Radiological Environmental Monitoring Programs required by Section 6.8 and (2) descriptions of the information that should be included in the Annual Radiological Environmental Operating and Semi-annual Radioactive Effluent Release Reports required by Specification 6.6.

Specification 6.9

OPERABLE - OPERABILITY safety

A system, subsystem, train, component, or device shall be OPERABLE or have OPERABILITY when it is capable of performing its specified function(s) and when all necessary attendant instrumentation, controls, electrical power, cooling or seal water, lubrication or other auxiliary equipment that are required for the system, subsystem, train, component or device to perform its function(s) are also capable of performing their related support function(s).

specified safety normal or emergency

OPERATIONAL MODE

An OPERATIONAL MODE, i.e., MODE, shall be any one inclusive combination of mode switch position and average reactor coolant temperature as specified in Table 1-2.

PHYSICS TESTS

PHYSICS TESTS shall be those tests performed to measure the fundamental nuclear characteristics of the reactor core and related instrumentation and 1) described in Chapter 14 of the FSAR, 2) authorized under the provisions of 10 CFR 50.59, or 3) otherwise approved by the Commission.

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PRESSURE BOUNDARY LEAKAGE

PRESSURE BOUNDARY LEAKAGE shall be leakage through a non-isolable fault in a reactor coolant system component body, pipe wall or vessel wall.

1.0 DEFINITIONS

OFFSITE DOSE CALCULATION MANUAL (ODCM)

The OFFSITE DOSE CALCULATION MANUAL (ODCM) shall contain the methodology and parameters used in the calculation of offsite doses resulting from radioactive gaseous and liquid effluents, in the calculation of gaseous and liquid effluent monitoring Alarm/Trip Setpoints, and in the conduct of the Environmental Radiological Monitoring Program. The ODCM shall also contain (1) the Radioactive Effluent Controls and Radiological Environmental Monitoring Programs required by Section 6.8 and (2) descriptions of the information that should be included in the Annual Radiological Environmental Operating and Semi-annual Radioactive Effluent Release Reports required by Specification 6.6.

Specification 6.9

OPERABLE - OPERABILITY

safety

A system, subsystem, train, component, or device shall be OPERABLE or have OPERABILITY when it is capable of performing its specified function(s) and when all necessary attendant instrumentation, controls, electrical power, cooling or seal water, lubrication or other auxiliary equipment that are required for the system, subsystem, train, component or device to perform its function(s) are also capable of performing their related support function(s).

specified safety

normal or emergency

OPERATIONAL MODE

An OPERATIONAL MODE, i.e., MODE, shall be any one inclusive combination of mode switch position and average reactor coolant temperature as specified in Table 1-2.

PHYSICS TESTS

PHYSICS TESTS shall be those tests performed to measure the fundamental nuclear characteristics of the reactor core and related instrumentation and 1) described in Chapter 14 of the FSAR, 2) authorized under the provisions of 10 CFR 50.59, or 3) otherwise approved by the Commission.

PRESSURE BOUNDARY LEAKAGE

PRESSURE BOUNDARY LEAKAGE shall be leakage through a non-isolable fault in a reactor coolant system component body, pipe wall or vessel wall.

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TABLE 1-1

SURVEILLANCE FREQUENCY NOTATION

	NOTATION	FREQUENCY
1. Shift	S	At least once per 12 hours
2. Day	D	At least once per 24 hours
3. Week	W	At least once per 7 days
4. Month	M	At least once per 31 days
5. Quarter	Q	At least once per 92 days
6. Semiannual	SA	At least once per 184 days
7. Annual	A	At least once per 366 days
8. Sesquiannual	E	At least once per 18 months (550 days)
9. Startup	S/U	Prior to each reactor startup
10. Not Applicable	N.A.	Not applicable

TABLE 1-1

SURVEILLANCE FREQUENCY NOTATION

	NOTATION	FREQUENCY
1. Shift	S	At least once per 12 hours
2. Day	D	At least once per 24 hours
3. Week	W	At least once per 7 days
4. Month	M	At least once per 31 days
5. Quarter	Q	At least once per 92 days
6. Semiannual	SA	At least once per 184 days
7. Annual	A	At least once per 366 days
8. Sesquiannual	E	At least once per 18 months (550 days)
9. Startup	S/U	Prior to each reactor startup
10. Not Applicable	N.A.	Not applicable

2.0 SAFETY LIMITS AND LIMITING SAFETY SYSTEM SETTINGS

2.1 SAFETY LIMITS

THERMAL POWER, Low Pressure or Low Flow

2.1.A THERMAL POWER shall not exceed 25% of RATED THERMAL POWER with the reactor vessel steam dome pressure less than 785 psig or core flow less than 10% of rated flow.

APPLICABILITY: OPERATIONAL MODE(s) 1 and 2.

ACTION:

With THERMAL POWER exceeding 25% of RATED THERMAL POWER and the reactor vessel steam dome pressure less than 785 psig or core flow less than 10% of rated flow, be in at least HOT SHUTDOWN within 2 hours and comply with the requirements of Specification 6.4.

6.7

THERMAL POWER, High Pressure and High Flow

2.1.B The MINIMUM CRITICAL POWER RATIO (MCPR) shall not be less than 1.08 with the reactor vessel steam dome pressure greater than or equal to 785 psig and core flow greater than or equal to 10% of rated flow. During single recirculation loop operation, this MCPR limit shall be increased by 0.01.

APPLICABILITY: OPERATIONAL MODE(s) 1 and 2.

ACTION:

With MCPR less than the above applicable limit and the reactor vessel steam dome pressure greater than or equal to 785 psig and core flow greater than or equal to 10% of rated flow, be in at least HOT SHUTDOWN within 2 hours and comply with the requirements of Specification 6.4.

6.7

2.0 SAFETY LIMITS AND LIMITING SAFETY SYSTEM SETTINGS

2.1 SAFETY LIMITS

THERMAL POWER, Low Pressure or Low Flow

2.1.A THERMAL POWER shall not exceed 25% of RATED THERMAL POWER with the reactor vessel steam dome pressure less than 785 psig or core flow less than 10% of rated flow.

APPLICABILITY: OPERATIONAL MODE(s) 1 and 2.

ACTION:

With THERMAL POWER exceeding 25% of RATED THERMAL POWER and the reactor vessel steam dome pressure less than 785 psig or core flow less than 10% of rated flow, be in at least HOT SHUTDOWN within 2 hours and comply with the requirements of Specification 6.4.

6.7

THERMAL POWER, High Pressure and High Flow

2.1.B The MINIMUM CRITICAL POWER RATIO (MCPR) shall not be less than 1.07 with the reactor vessel steam dome pressure greater than or equal to 785 psig and core flow greater than or equal to 10% of rated flow. During single recirculation loop operation, this MCPR limit shall be increased by 0.01.

APPLICABILITY: OPERATIONAL MODE(s) 1 and 2.

ACTION:

With MCPR less than the above applicable limit and the reactor vessel steam dome pressure greater than or equal to 785 psig and core flow greater than or equal to 10% of rated flow, be in at least HOT SHUTDOWN within 2 hours and comply with the requirements of Specification 6.4.

6.7

2.0 SAFETY LIMITS AND LIMITING SAFETY SYSTEM SETTINGS

Reactor Coolant System Pressure

2.1.C The reactor coolant system pressure, as measured in the reactor vessel steam dome, shall not exceed 1345 psig.

APPLICABILITY: OPERATIONAL MODE(s) 1, 2, 3 and 4.

ACTION:

With the reactor coolant system pressure, as measured in the reactor vessel steam dome, above 1345 psig, be in at least HOT SHUTDOWN with reactor coolant system pressure less than or equal to 1345 psig within 2 hours and comply with the requirements of Specification 6.4.

6.7

Reactor Vessel Water Level

2.1.D The reactor vessel water level shall be greater than or equal to twelve inches above the top of the active irradiated fuel.

APPLICABILITY: OPERATIONAL MODE(s) 3, 4 and 5.

ACTION:

With the reactor vessel water level at or below twelve inches above the top of the active irradiated fuel, manually initiate the ECCS to restore the water level, after depressurizing the reactor vessel, if required, and comply with the requirements of Specification 6.4.

6.7

2.0 SAFETY LIMITS AND LIMITING SAFETY SYSTEM SETTINGS

Reactor Coolant System Pressure

2.1.C The reactor coolant system pressure, as measured in the reactor vessel steam dome, shall not exceed 1345 psig.

APPLICABILITY: OPERATIONAL MODE(s) 1, 2, 3 and 4.

ACTION:

With the reactor coolant system pressure, as measured in the reactor vessel steam dome, above 1345 psig, be in at least HOT SHUTDOWN with reactor coolant system pressure less than or equal to 1345 psig within 2 hours and comply with the requirements of Specification 6.4.

6.7

Reactor Vessel Water Level

2.1.D The reactor vessel water level shall be greater than twelve inches above the top of the active irradiated fuel.

APPLICABILITY: OPERATIONAL MODE(s) 3, 4 and 5.

ACTION:

With the reactor vessel water level at or below twelve inches above the top of the active irradiated fuel, manually initiate the ECCS to restore the water level, after depressurizing the reactor vessel, if required, and comply with the requirements of Specification 6.4.

6.7

3.0 - LIMITING CONDITIONS FOR OPERATION

A. Compliance with the Limiting Conditions for Operation contained in the succeeding Specifications is required during the OPERATIONAL MODE(s) or other conditions specified therein; except that upon failure to meet the Limiting Conditions for Operation, the associated ACTION requirements shall be met.

B. Noncompliance with a Specification shall exist when the requirements of the Limiting Condition for Operation and associated ACTION requirements are not met within the specified time intervals. If the Limiting Condition for Operation is restored prior to expiration of the specified time intervals, completion of the ACTION requirements is not required.

C. When a Limiting Condition for Operation is not met, except as provided in the associated ACTION requirements, within one hour ACTION shall be initiated to place the unit in an OPERATIONAL MODE in which the Specification does not apply by placing it, as applicable, in:

1. At least HOT SHUTDOWN within the next 12 hours, and
2. At least COLD SHUTDOWN within the subsequent 24 hours.

Where corrective measures are completed that permit operation under the ACTION requirements, the ACTION may be taken in accordance with the specified time limits as measured from the time of failure to meet the Limiting Condition for Operation. Exceptions to these requirements are stated in the individual Specifications.

This Specification is not applicable in OPERATIONAL MODE 4 or 5.

D. Entry into an OPERATIONAL MODE or other specified condition shall not be made when the conditions for the Limiting Conditions for Operation are not met and the associated ACTION requires placing the plant in an OPERATIONAL MODE or other specified condition of operation in which the Limiting Condition for Operation does not apply if they are not met within a specified time interval. Entry into an OPERATIONAL MODE or other specified condition may be made in accordance with the ACTION requirements when conformance to them permits continued operation of the facility for an unlimited period of time. This provision shall not prevent passage through or to OPERATIONAL MODE(s) as required to comply with ACTION requirements. Exceptions to these requirements are stated in the individual Specifications.

'INSERT'

3.0 - LIMITING CONDITIONS FOR OPERATION

A. Compliance with the Limiting Conditions for Operation contained in the succeeding Specifications is required during the OPERATIONAL MODE(s) or other conditions specified therein; except that upon failure to meet the Limiting Conditions for Operation, the associated ACTION requirements shall be met.

B. Noncompliance with a Specification shall exist when the requirements of the Limiting Condition for Operation and associated ACTION requirements are not met within the specified time intervals. If the Limiting Condition for Operation is restored prior to expiration of the specified time intervals, completion of the ACTION requirements is not required.

C. When a Limiting Condition for Operation is not met, except as provided in the associated ACTION requirements, within one hour ACTION shall be initiated to place the unit in an OPERATIONAL MODE in which the Specification does not apply by placing it, as applicable, in:

1. At least HOT SHUTDOWN within the next 12 hours, and

2. At least COLD SHUTDOWN within the subsequent 24 hours.

Where corrective measures are completed that permit operation under the ACTION requirements, the ACTION may be taken in accordance with the specified time limits as measured from the time of failure to meet the Limiting Condition for Operation. Exceptions to these requirements are stated in the individual Specifications.

This Specification is not applicable in OPERATIONAL MODE 4 or 5.

D. Entry into an OPERATIONAL MODE or other specified condition shall not be made when the conditions for the Limiting Conditions for Operation are not met and the associated ACTION requires placing the plant in an OPERATIONAL MODE or other specified condition of operation in which the Limiting Condition for Operation does not apply if they are not met within a specified time interval. Entry into an OPERATIONAL MODE or other specified condition may be made in accordance with the ACTION requirements when conformance to them permits continued operation of the facility for an unlimited period of time. This provision shall not prevent passage through or to OPERATIONAL MODE(s) as required to comply with ACTION requirements. Exceptions to these requirements are stated in the individual Specifications.

INSERT

INSERT

When an LCO is not met, entry into a MODE or other specified condition in the Applicability shall not be made except when the associated ACTIONS to be entered permit continued operation in the MODE or other specified condition in the Applicability for an unlimited period of time. This Specification shall not prevent changes in MODES or other specified conditions in the Applicability that are required to comply with ACTIONS.

4.0 - SURVEILLANCE REQUIREMENTS

A. Surveillance Requirements shall be met during the reactor OPERATIONAL MODE(s) or other conditions specified for individual Limiting Conditions for Operation unless otherwise stated in an individual Surveillance Requirement.

B. Each Surveillance Requirement shall be performed within the specified surveillance interval with a maximum allowable extension not to exceed 25 percent of the surveillance interval.

C. Failure to perform a Surveillance Requirement within the allowed surveillance interval, defined by Specification 4.0.B, shall constitute noncompliance with the OPERABILITY requirements for a Limiting Condition for Operation. The time limits of the ACTION requirements are applicable at the time it is identified that a Surveillance Requirement has not been performed. The ACTION requirements may be delayed for up to 24 hours to permit the completion of the surveillance when the allowable outage time limits of the ACTION requirements are less than 24 hours. Surveillance requirements do not have to be performed on inoperable equipment.

D. Entry into an OPERATIONAL MODE or other specified applicable condition shall not be made unless the Surveillance Requirement(s) associated with the Limiting Condition for Operation have been performed within the applicable surveillance interval or as otherwise specified. This provision shall not prevent passage through or to OPERATIONAL MODE(s) as required to comply with ACTION requirements.

E. Surveillance Requirements for inservice inspection and testing of ASME Code Class 1, 2, and 3 components shall be applicable as follows:

1. Inservice Inspection of ASME Code Class 1, 2, and 3 components and inservice testing of ASME Code Class 1, 2, and 3 pumps and valves shall be performed in accordance with Section XI of the ASME Boiler and Pressure Vessel Code and applicable Addenda as required by 10 CFR Part 50, Section 50.55a(g), except where specific written relief has been granted by the Commission pursuant to 10 CFR Part 50, Section 50.55a(g)(6)(i).

and 50.55a(f), respectively

or 50.55a(f)(6)(i), respectively

4.0 - SURVEILLANCE REQUIREMENTS

A. Surveillance Requirements shall be met during the reactor OPERATIONAL MODE(s) or other conditions specified for individual Limiting Conditions for Operation unless otherwise stated in an individual Surveillance Requirement.

B. Each Surveillance Requirement shall be performed within the specified surveillance interval with a maximum allowable extension not to exceed 25 percent of the surveillance interval.

C. Failure to perform a Surveillance Requirement within the allowed surveillance interval, defined by Specification 4.0.B, shall constitute noncompliance with the OPERABILITY requirements for a Limiting Condition for Operation. The time limits of the ACTION requirements are applicable at the time it is identified that a Surveillance Requirement has not been performed. The ACTION requirements may be delayed for up to 24 hours to permit the completion of the surveillance when the allowable outage time limits of the ACTION requirements are less than 24 hours. Surveillance requirements do not have to be performed on inoperable equipment.

D. Entry into an OPERATIONAL MODE or other specified applicable condition shall not be made unless the Surveillance Requirement(s) associated with the Limiting Condition for Operation have been performed within the applicable surveillance interval or as otherwise specified. This provision shall not prevent passage through or to OPERATIONAL MODE(s) as required to comply with ACTION requirements.

E. Surveillance Requirements for inservice inspection and testing of ASME Code Class 1, 2, and 3 components shall be applicable as follows:

1. Inservice Inspection of ASME Code Class 1, 2, and 3 components and inservice testing of ASME Code Class 1, 2, and 3 pumps and valves shall be performed in accordance with Section XI of the ASME Boiler and Pressure Vessel Code and applicable Addenda as required by 10 CFR Part 50, Section 50.55a(g), except where specific written relief has been granted by the Commission pursuant to 10 CFR Part 50, Section 50.55a(g)(6)(i).

and 50.55a(f), respectively

or 50.55a(f)(6)(i), respectively

4.0 - SURVEILLANCE REQUIREMENTS

2. Surveillance intervals specified in Section XI of the ASME Boiler and Pressure Vessel Code and applicable Addenda for the inservice inspection and testing activities required by the ASME Boiler and Pressure Vessel Code and applicable Addenda shall be applicable as follows in these Technical Specifications:

ASME Boiler and Pressure Vessel Code and applicable Addenda terminology for inservice inspection and testing activities	Required Frequencies for performing inservice inspection and testing activities
Weekly	At least once per 7 days
Monthly	At least once per 31 days
Quarterly or every 3 months	At least once per 92 days
Semiannually or every 6 months	At least once per 184 days
Every 9 months	At least once per 276 days
Yearly or annually	At least once per 366 days

3. The provisions of Specification 4.0.B are applicable to the above required frequencies for performing inservice inspection and testing activities.

4. Performance of the above inservice inspection and testing activities shall be in addition to other specified Surveillance Requirements.

5. Nothing in the ASME Boiler and Pressure Vessel Code shall be construed to supersede the requirements of any Technical Specification.

6. The Inservice Inspection Program for piping identified in NRC Generic Letter 88-01 shall be performed in accordance with the staff positions on schedule, methods, and personnel and sample expansion included in Generic Letter 88-01 or in accordance with alternate measures approved by the NRC staff.

Biennially or every 2 years — At least once per 731 days

4.0 - SURVEILLANCE REQUIREMENTS

2. Surveillance intervals specified in Section XI of the ASME Boiler and Pressure Vessel Code and applicable Addenda for the inservice inspection and testing activities required by the ASME Boiler and Pressure Vessel Code and applicable Addenda shall be applicable as follows in these Technical Specifications:

ASME Boiler and Pressure Vessel Code and applicable Addenda terminology for inservice inspection and testing activities	Required Frequencies for performing inservice inspection and testing activities
Weekly	At least once per 7 days
Monthly	At least once per 31 days
Quarterly or every 3 months	At least once per 92 days
Semiannually or every 6 months	At least once per 184 days
Every 9 months	At least once per 276 days
Yearly or annually	At least once per 366 days

3. The provisions of Specification 4.0.B are applicable to the above required frequencies for performing inservice inspection and testing activities.

4. Performance of the above inservice inspection and testing activities shall be in addition to other specified Surveillance Requirements.

5. Nothing in the ASME Boiler and Pressure Vessel Code shall be construed to supersede the requirements of any Technical Specification.

6. The Inservice Inspection Program for piping identified in NRC Generic Letter 88-01 shall be performed in accordance with the staff positions on schedule, methods, and personnel and sample expansion included in Generic Letter 88-01 or in accordance with alternate measures approved by the NRC staff.

Biennially or every 2 years   At least once per 731 days

3.1 - LIMITING CONDITIONS FOR OPERATION

A. Reactor Protection System (RPS)

The reactor protection system (RPS) instrumentation CHANNEL(s) shown in Table 3.1.A-1 shall be OPERABLE.

APPLICABILITY:

As shown in Table 3.1.A-1.

ACTION:

1. With the number of OPERABLE CHANNEL(s) less than required by the Minimum CHANNEL(s) per TRIP SYSTEM requirement for one TRIP SYSTEM, place the inoperable CHANNEL(s) and/or that TRIP SYSTEM in the tripped condition[a] within 1 hour.
2. With the number of OPERABLE CHANNEL(s) less than required by the Minimum CHANNEL(s) per TRIP SYSTEM requirement for both TRIP SYSTEM(s), place at least one TRIP SYSTEM in the tripped condition[b] within 1 hour and take the ACTION required by Table 3.1.A-1.

4.1 - SURVEILLANCE REQUIREMENTS

A. Reactor Protection System

1. Each reactor protection system instrumentation CHANNEL shall be demonstrated OPERABLE by the performance of the CHANNEL CHECK, CHANNEL FUNCTIONAL TEST and CHANNEL CALIBRATION operations for the OPERATIONAL MODE(s) and at the frequencies shown in Table 4.1.A-1.
2. LOGIC SYSTEM FUNCTIONAL TEST(s) ~~and simulated automatic operation~~ of all CHANNEL(s) shall be performed at least once per 18 months.
3. The response time of each reactor trip functional unit shown in Table 3.1.A-1 shall be demonstrated at least once per 18 months. Each test shall include at least one CHANNEL per TRIP SYSTEM such that all CHANNEL(s) are tested at least once every N times 18 months where N is the total number of redundant CHANNEL(s) in a specific reactor TRIP SYSTEM. The system response time for each trip function from the opening of the sensor contact up to and including the opening of the trip actuator shall not exceed 50 milliseconds.

---

a An inoperable CHANNEL need not be placed in the tripped condition when this would cause the trip function to occur. In these cases, the inoperable CHANNEL shall be restored to OPERABLE status within 2 hours or the ACTION required by Table 3.1.A-1 for that trip function shall be taken.

b The TRIP SYSTEM need not be placed in the tripped condition if this would cause the trip function to occur. When a TRIP SYSTEM can be placed in the tripped condition without causing the trip function to occur, place the TRIP SYSTEM with the most inoperable CHANNEL(s) in the tripped condition; if both systems have the same number of inoperable CHANNEL(s), place either TRIP SYSTEM in the tripped condition.

3.1 - LIMITING CONDITIONS FOR OPERATION

A. Reactor Protection System (RPS)

The reactor protection system (RPS) instrumentation CHANNEL(s) shown in Table 3.1.A-1 shall be OPERABLE.

APPLICABILITY:

As shown in Table 3.1.A-1.

ACTION:

1. With the number of OPERABLE CHANNEL(s) less than required by the Minimum CHANNEL(s) per TRIP SYSTEM requirement for one TRIP SYSTEM, place the inoperable CHANNEL(s) and/or that TRIP SYSTEM in the tripped condition[a] within 1 hour.

2. With the number of OPERABLE CHANNEL(s) less than required by the Minimum CHANNEL(s) per TRIP SYSTEM requirement for both TRIP SYSTEM(s), place at least one TRIP SYSTEM in the tripped condition[b] within 1 hour and take the ACTION required by Table 3.1.A-1.

4.1 - SURVEILLANCE REQUIREMENTS

A. Reactor Protection System

1. Each reactor protection system instrumentation CHANNEL shall be demonstrated OPERABLE by the performance of the CHANNEL CHECK, CHANNEL FUNCTIONAL TEST and CHANNEL CALIBRATION operations for the OPERATIONAL MODE(s) and at the frequencies shown in Table 4.1.A-1.

2. LOGIC SYSTEM FUNCTIONAL TEST(s) ~~and simulated automatic operation~~ of all CHANNEL(s) shall be performed at least once per 18 months.

3. The response time of each reactor trip functional unit shown in Table 3.1.A-1 shall be demonstrated at least once per 18 months. Each test shall include at least one CHANNEL per TRIP SYSTEM such that all CHANNEL(s) are tested at least once every N times 18 months where N is the total number of redundant CHANNEL(s) in a specific reactor TRIP SYSTEM. The system response time for each trip function from the opening of the sensor contact up to and including the opening of the trip actuator shall not exceed 50 milliseconds.

---

a An inoperable CHANNEL need not be placed in the tripped condition when this would cause the trip function to occur. In these cases, the inoperable CHANNEL shall be restored to OPERABLE status within 2 hours or the ACTION required by Table 3.1.A-1 for that trip function shall be taken.

b The TRIP SYSTEM need not be placed in the tripped condition if this would cause the trip function to occur. When a TRIP SYSTEM can be placed in the tripped condition without causing the trip function to occur, place the TRIP SYSTEM with the most inoperable CHANNEL(s) in the tripped condition; if both systems have the same number of inoperable CHANNEL(s), place either TRIP SYSTEM in the tripped condition.

TABLE 4.1.A-1

REACTOR PROTECTION SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

Functional Unit	Applicable OPERATIONAL MODES	CHANNEL CHECK	CHANNEL FUNCTIONAL TEST	CHANNEL[a] CALIBRATION
1. Intermediate Range Monitor:				
a. Neutron Flux - High	2	SU, S, [b]	S/U[c], W[o]	
	3, 4, 5	S	W	
b. Inoperative	2, 3, 4, 5	NA	W[o]	NA
2. Average Power Range Monitor[f]:				
a. Setdown Neutron Flux - High	2	SU, S, [b]	S/U[c], W[o]	SA[o]
	3, 5[m]	S	W	SA
b. Flow Biased Neutron Flux - High	1	S, D[g]	W	W[d,e], SA
c. Fixed Neutron Flux - High	1	S	W	W[d], SA
d. Inoperative	1, 2, 3, 5[m]	NA	W	NA
3. Reactor Vessel Steam Dome Pressure - High	1, 2[i]	NA	M	Q
4. Reactor Vessel Water Level - Low	1, 2	D	M	E[h]
5. Main Steam Line Isolation Valve - Closure	1, 2[p]	NA	M	E
6. Main Steam Line Radiation - High	1, 2[i]	S	M	
7. Drywell Pressure - High	1, 2[n]	NA	M	Q

E(o,r)
E(o,r)
(o)
E(g)

TABLE 4.1.A-1

REACTOR PROTECTION SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

Functional Unit	Applicable OPERATIONAL MODES	CHANNEL CHECK	CHANNEL FUNCTIONAL TEST	CHANNEL[a] CALIBRATION
1. Intermediate Range Monitor:				
a. Neutron Flux - High	2 3, 4, 5	~~SU~~, S,[b] S	S/U[c], W[o] W (o)	$E^{(o,g)}$ $E^{(o,g)}$
b. Inoperative	2, 3, 4, 5	NA	W[o]	NA
2. Average Power Range Monitor[f]:				
a. Setdown Neutron Flux - High	2 3, 5[m]	~~SU~~, S,[b] S	S/U[c], W[o] W	SA[o] SA
b. Flow Biased Neutron Flux - High	1	S, D[g]	W	W[d,e], SA
c. Fixed Neutron Flux - High	1	S	W	W[d], SA
d. Inoperative	1, 2, 3, 5[m]	NA	W	NA
3. Reactor Vessel Steam Dome Pressure - High	1, 2[i]	NA	M	Q
4. Reactor Vessel Water Level - Low	1, 2	D	M	E[h]
5. Main Steam Line Isolation Valve - Closure	1	NA	M	E
6. Main Steam Line Radiation - High	1, 2[i]	S	M	$E^{(p)}$
7. Drywell Pressure - High	1, 2[n]	NA	M	Q

TABLE 4.1.A-1 (Continued)

REACTOR PROTECTION SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

TABLE NOTATION

(a) Neutron detectors may be excluded from the CHANNEL CALIBRATION.

(b) The IRM and SRM channels shall be determined to overlap for at least (½) decades during each startup after entering OPERATIONAL MODE 2 and the IRM and APRM channels shall be determined to overlap for at least (½) decades during each controlled shutdown, if not performed within the previous 7 days.

(c) Within 24 hours prior to startup, if not performed within the previous 7 days. The weekly CHANNEL FUNCTIONAL TEST may be used to fulfill this requirement.

(d) This calibration shall consist of the adjustment of the APRM CHANNEL to conform, within 2% of RATED THERMAL POWER, to the power values calculated by a heat balance during OPERATIONAL MODE 1 when THERMAL POWER is ≥25% of RATED THERMAL POWER. This adjustment must be accomplished: a) within 2 hours if the APRM CHANNEL is indicating lower power values than the heat balance, or b) within 12 hours if the APRM CHANNEL is indicating higher power values than the heat balance. Until any required APRM adjustment has been accomplished, notification shall be posted on the reactor control panel.

Any APRM CHANNEL gain adjustment made in compliance with Specification 3.11.B shall not be included in determining the above difference. This calibration is not required when THERMAL POWER is <25% of RATED THERMAL POWER. The provisions of Specification 4.0.D are not applicable.

(e) This calibration shall consist of the adjustment of the APRM flow biased channel to conform to a calibrated flow signal.

(f) The LPRMs shall be calibrated at least once per ~~1000~~ 2000 effective full power hours (EFPH).

(g) Verify measured recirculation loop flow to be greater than or equal to established recirculation loop flow at the existing pump speed.

(h) Trip units are calibrated at least once per 31 days and transmitters are calibrated at the frequency identified in the table.

(i) This function is not required to be OPERABLE when the reactor pressure vessel head is unbolted or removed per Specification 3.12.A.

(j) With any control rod withdrawn. Not applicable to control rods removed per Specification 3.10.I or 3.10.J.

(k) This function may be bypassed, provided a control rod block is actuated, for reactor protection system reset in Refuel and Shutdown positions of the reactor mode switch.

TABLE 4.1.A-1 (Continued)

REACTOR PROTECTION SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

TABLE NOTATION

(a) Neutron detectors may be excluded from the CHANNEL CALIBRATION.

(b) The IRM and SRM channels shall be determined to overlap for at least (½) decades during each startup after entering OPERATIONAL MODE 2 and the IRM and APRM channels shall be determined to overlap for at least (½) decades during each controlled shutdown, if not performed within the previous 7 days.

(c) Within 24 hours prior to startup, if not performed within the previous 7 days. The weekly CHANNEL FUNCTIONAL TEST may be used to fulfill this requirement.

(d) This calibration shall consist of the adjustment of the APRM CHANNEL to conform, within 2% of RATED THERMAL POWER, to the power values calculated by a heat balance during OPERATIONAL MODE 1 when THERMAL POWER is ≥25% of RATED THERMAL POWER. This adjustment must be accomplished: a) within 2 hours if the APRM CHANNEL is indicating lower power values than the heat balance, or b) within 12 hours if the APRM CHANNEL is indicating higher power values than the heat balance. Until any required APRM adjustment has been accomplished, notification shall be posted on the reactor control panel.

Any APRM CHANNEL gain adjustment made in compliance with Specification 3.11.B shall not be included in determining the above difference. This calibration is not required when THERMAL POWER is <25% of RATED THERMAL POWER. The provisions of Specification 4.0.D are not applicable.

(e) This calibration shall consist of the adjustment of the APRM flow biased channel to conform to a calibrated flow signal.

(f) The LPRMs shall be calibrated at least once per ~~1000~~ 2000 effective full power hours (EFPH).

(g) Verify measured recirculation loop flow to be greater than or equal to established recirculation loop flow at the existing pump speed.

(h) Trip units are calibrated at least once per 31 days and transmitters are calibrated at the frequency identified in the table.

(i) This function is not required to be OPERABLE when the reactor pressure vessel head is unbolted or removed per Specification 3.12.A.

(j) With any control rod withdrawn. Not applicable to control rods removed per Specification 3.10.I or 3.10.J.

(k) This function may be bypassed, provided a control rod block is actuated, for reactor protection system reset in Refuel and Shutdown positions of the reactor mode switch.

TABLE 4.1.A-1 (Continued)

REACTOR PROTECTION SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

(l) This function not required to be OPERABLE when THERMAL POWER is less than 45% of RATED THERMAL POWER.

(m) Required to be OPERABLE only prior to and during required SHUTDOWN MARGIN demonstrations performed per Specification 3.12.B.

(n) This function is not required to be OPERABLE when PRIMARY CONTAINMENT INTEGRITY is not required.

(o) The provisions of Specification 4.0.D are not applicable to the CHANNEL FUNCTIONAL TEST and CHANNEL CALIBRATION surveillances ~~for entry into their OPERATIONAL MODE(s) from OPERATIONAL MODE 1 provided the surveillances are performed within 12 hours after such entry.~~ for a period of 24 hours after entering OPERATIONAL MODE 2 or 3 when shutting down from OPERATIONAL MODE 1.

(p) This function is not required to be OPERABLE when reactor pressure is less than 600 psig.

~~* The frequency of this calibration has been left as an open item.~~

(q) A current source provides an instrument channel alignment every 3 months.

(r) "INSERT"

INSERT

The CHANNEL CALIBRATION surveillance requirements shall be performed if not performed within the previous seven days.

TABLE 4.1.A-1 (Continued)

REACTOR PROTECTION SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

(l) This function not required to be OPERABLE when THERMAL POWER is less than 45% of RATED THERMAL POWER.

(m) Required to be OPERABLE only prior to and during required SHUTDOWN MARGIN demonstrations performed per Specification 3.12.B.

(n) This function is not required to be OPERABLE when PRIMARY CONTAINMENT INTEGRITY is not required.

(o) The provisions of Specification 4.0.D are not applicable to the CHANNEL FUNCTIONAL TEST and CHANNEL CALIBRATION surveillances ~~for entry into their OPERATIONAL MODE(s) from OPERATIONAL MODE 1 provided the surveillances are performed within 12 hours after such entry.~~ for a period of 24 hours after entering OPERATIONAL MODE 2 or 3 when shutting down from OPERATIONAL MODE 1.

~~* The frequency of this calibration has been left as an open item.~~

(p) A current source provides an instrument channel alignment every 3 months.

(q) 'INSERT'

INSERT

The CHANNEL CALIBRATION surveillance requirements shall be performed if not performed within the previous seven days.

3.2 - LIMITING CONDITIONS FOR OPERATION

A. Isolation Actuation

The isolation actuation instrumentation CHANNEL(s) shown in Table 3.2.A-1 shall be OPERABLE with their trip setpoints set consistent with the values shown in the Trip Setpoint column.

APPLICABILITY:

As shown in Table 3.2.A-1.

ACTION:

1. With an isolation actuation instrumentation CHANNEL trip setpoint less conservative than the value shown in the Trip Setpoint column of Table 3.2.A-1, declare the CHANNEL inoperable until the CHANNEL is restored to OPERABLE status with its trip setpoint adjusted consistent with the Trip Setpoint value.
2. With the number of OPERABLE CHANNEL(s) less than required by the Minimum CHANNEL(s) per TRIP SYSTEM requirement for one TRIP SYSTEM, place the inoperable CHANNEL(s) and/or TRIP SYSTEM in the tripped condition[a] within one hour.

4.2 - SURVEILLANCE REQUIREMENTS

A. Isolation Actuation

1. Each isolation actuation instrumentation CHANNEL shall be demonstrated OPERABLE by the performance of the CHANNEL CHECK, CHANNEL FUNCTIONAL TEST and CHANNEL CALIBRATION operations for the OPERATIONAL MODE(s) and at the frequencies shown in Table 4.2.A-1.
2. LOGIC SYSTEM FUNCTIONAL TEST(s) ~~and simulated automatic operation~~ of all CHANNEL(s) shall be performed at least once per 18 months.

---

a An inoperable CHANNEL need not be placed in the tripped condition where this would cause the trip function to occur. In these cases, the inoperable CHANNEL shall be restored to OPERABLE status within 2 hours or the ACTION required by Table 3.2.A-1 for that trip function shall be taken.

3.2 - LIMITING CONDITIONS FOR OPERATION

A. Isolation Actuation

The isolation actuation instrumentation CHANNEL(s) shown in Table 3.2.A-1 shall be OPERABLE with their trip setpoints set consistent with the values shown in the Trip Setpoint column.

APPLICABILITY:

As shown in Table 3.2.A-1.

ACTION:

1. With an isolation actuation instrumentation CHANNEL trip setpoint less conservative than the value shown in the Trip Setpoint column of Table 3.2.A-1, declare the CHANNEL inoperable until the CHANNEL is restored to OPERABLE status with its trip setpoint adjusted consistent with the Trip Setpoint value.
2. With the number of OPERABLE CHANNEL(s) less than required by the Minimum CHANNEL(s) per TRIP SYSTEM requirement for one TRIP SYSTEM, place the inoperable CHANNEL(s) and/or TRIP SYSTEM in the tripped condition[a] within one hour.

4.2 - SURVEILLANCE REQUIREMENTS

A. Isolation Actuation

1. Each isolation actuation instrumentation CHANNEL shall be demonstrated OPERABLE by the performance of the CHANNEL CHECK, CHANNEL FUNCTIONAL TEST and CHANNEL CALIBRATION operations for the OPERATIONAL MODE(s) and at the frequencies shown in Table 4.2.A-1.
2. LOGIC SYSTEM FUNCTIONAL TEST(s) ~~and simulated automatic operation~~ of all CHANNEL(s) shall be performed at least once per 18 months.

a An inoperable CHANNEL need not be placed in the tripped condition where this would cause the trip function to occur. In these cases, the inoperable CHANNEL shall be restored to OPERABLE status within 2 hours or the ACTION required by Table 3.2.A-1 for that trip function shall be taken.

TABLE 3.2.A-1

ISOLATION ACTUATION INSTRUMENTATION

Functional Unit	Trip Setpoint[f]	Minimum CHANNEL(s) per TRIP SYSTEM[a]	Applicable OPERATIONAL MODE(s)	ACTION
1. PRIMARY CONTAINMENT ISOLATION				
a. Reactor Vessel Water Level - Low	≥144 inches	2	1, 2, 3	20
b. Drywell Pressure - High[d]	≤2 psig	2	1, 2, 3	20
c. Drywell Radiation - High	≤100 R/hr	1	1, 2, 3	23
2. SECONDARY CONTAINMENT ISOLATION				
a. Reactor Vessel Water Level - Low[c]	≥144 inches	2	1, 2, 3 & *	24
b. Drywell Pressure - High[c,d]	≤2 psig	2	1, 2, 3	24
c. Reactor Building Ventilation Exhaust Radiation - High[c]	≤4 mR/hr	2	1, 2, 3 & * *	24
d. Refueling Floor Radiation - High[c]	≤100 mR/hr	2	1, 2, 3 & * *	24
3. MAIN STEAM LINE (MSL) ISOLATION				
a. Reactor Vessel Water Level - Low Low	≥84 inches	2	1, 2, 3	21
b. MSL Tunnel Radiation - High[b]	≤3[g] x normal background	2	1, 2, 3	21
c. MSL Pressure - Low	≥825 psig	2	1	22
d. MSL Flow - High	≤120% of rated	2/line	1, 2, 3	21
e. MSL Tunnel Temperature - High	≤200°F	4	1, 2, 3	21

2 of 4 in each of 2 sets

TABLE 3.2.A-1

ISOLATION ACTUATION INSTRUMENTATION

Functional Unit	Trip Setpoint[j]	Minimum CHANNEL(s) per TRIP SYSTEM[a]	Applicable OPERATIONAL MODE(s)	ACTION
1. PRIMARY CONTAINMENT ISOLATION				
a. Reactor Vessel Water Level - Low	≥144 inches	2	1, 2, 3	20
b. Drywell Pressure - High[d]	≤2.5 psig	2	1, 2, 3	20
c. Drywell Radiation - High	≤100 R/hr	1	1, 2, 3	23
2. SECONDARY CONTAINMENT ISOLATION				
a. Reactor Vessel Water Level - Low[c,k]	≥144 inches	2	1, 2, 3 & *	24
b. Drywell Pressure - High[c,d,k]	≤2.5 psig	2	1, 2, 3	24
c. Reactor Building Ventilation Exhaust Radiation - High[c,k]	≤3 mR/hr	2	1, 2, 3 & **	24
d. Refueling Floor Radiation - High[c,k]	≤100 mR/hr	2	1, 2, 3 & **	24
3. MAIN STEAM LINE (MSL) ISOLATION				
a. Reactor Vessel Water Level - Low Low	≥84 inches	2	1, 2, 3	21
b. MSL Tunnel Radiation - High[b]	≤15[h] x normal background	2	1, 2, 3	21
c. MSL Pressure - Low	≥825 psig	2	1	22
d. MSL Flow - High[k]	≤140% of rated	2/line	1, 2, 3	21
e. MSL Tunnel Temperature - High	≤200°F	~~8~~ 2 of 4 in each of 2 sets	1, 2, 3	21

TABLE 3.2.A-1 (Continued)

ISOLATION ACTUATION INSTRUMENTATION

Functional Unit	Trip Setpoint[i]	Minimum CHANNEL(s) per TRIP SYSTEM[a]	Applicable OPERATIONAL MODE(s)	ACTION
4. REACTOR WATER CLEANUP SYSTEM ISOLATION				
a. Standby Liquid Control System Initiation[f]	NA	NA	1, 2, 3	23
b. Reactor Vessel Water Level - Low	≥144 inches	2	1, 2, 3	23
5. ISOLATION CONDENSER ISOLATION				
a. Steam Flow - High	≤300% of rated steam flow	1	1, 2, 3	23
b. Return Flow - High	≤32 (Unit 2)/ ≤14.8 (Unit 3) inches water diff.	1	1, 2, 3	23
6. HIGH PRESSURE COOLANT INJECTION ISOLATION				
a. Steam Flow - High	≤300% of rated steam flow [h]	1	1, 2, 3	23
b. Reactor Vessel Pressure - Low	≥80 psig	2	1, 2, 3	23
c. Area Temperature - High	≤200°F	~~8~~ 4[j]	1, 2, 3	23

TABLE 3.2.A-1 (Continued)

ISOLATION ACTUATION INSTRUMENTATION

TABLE NOTATION

* During CORE ALTERATIONS or operations with a potential for draining the reactor vessel.

** When handling irradiated fuel in the secondary containment.

(a) A CHANNEL may be placed in an inoperable status for up to 2 hours for required surveillance without placing the CHANNEL in the tripped condition provided the Functional Unit maintains isolation actuation capability.

(b) Also trips the mechanical vacuum pump and isolates the steam jet air ejectors.

(c) Isolates the reactor building ventilation system and actuates the standby gas treatment system.

(d) This function is not required to be OPERABLE when PRIMARY CONTAINMENT INTEGRITY is not required.

(e) Only one TRIP SYSTEM.

(f) Closes only reactor water cleanup system isolation valves.

(g) Normal background is as measured during full power operation without hydrogen being injected. With Unit 2 operating above 20% RATED THERMAL POWER and hydrogen being injected into the feedwater, this Unit 2 setting may be as measured during full power operation with hydrogen being injected.

(h) Includes a time delay of $3 \leq t \leq 9$ seconds.

(i) Reactor vessel water level settings are expressed in inches above the top of active fuel (which is 360 inches above vessel zero).

(j) All four switches in either of 2 groups for each trip system

TABLE 4.2.A-1

ISOLATION ACTUATION INSTRUMENTATION SURVEILLANCE REQUIREMENTS

Functional Unit	CHANNEL CHECK	CHANNEL FUNCTIONAL TEST	CHANNEL CALIBRATION	Applicable OPERATIONAL MODE(s)
1. PRIMARY CONTAINMENT ISOLATION				
a. Reactor Vessel Water Level - Low	S	M	E[a]	1, 2, 3
b. Drywell Pressure - High[b]	NA	M	Q	1, 2, 3
c. Drywell Radiation - High	S	M	E	1, 2, 3
2. SECONDARY CONTAINMENT ISOLATION				
a. Reactor Vessel Water Level - Low[c]	S	M	E[a]	1, 2, 3 & *
b. Drywell Pressure - High[b,c]	NA	M	Q	1, 2, 3
c. Reactor Building Ventilation Exhaust Radiation - High[c]	S	M	E	1, 2, 3 & * *
d. Refueling Floor Radiation - High[c]	S	M	E e Q	1, 2, 3 & * *
3. MAIN STEAM LINE (MSL) ISOLATION				
a. Reactor Vessel Water Level - Low Low	S	M	E[a]	1, 2, 3
b. MSL Tunnel Radiation - High	S	M	E (d)	1, 2, 3
c. MSL Pressure - Low	NA	M	Q	1
d. MSL Flow - High	S	M	E	1, 2, 3
e. MSL Tunnel Temperature - High	NA	E	E	1, 2, 3

TABLE 4.2.A-1

ISOLATION ACTUATION INSTRUMENTATION SURVEILLANCE REQUIREMENTS

Functional Unit	CHANNEL CHECK	CHANNEL FUNCTIONAL TEST	CHANNEL CALIBRATION	Applicable OPERATIONAL MODE(s)
1. PRIMARY CONTAINMENT ISOLATION				
a. Reactor Vessel Water Level - Low	S	M	E[a]	1, 2, 3
b. Drywell Pressure - High[b]	NA	M	Q	1, 2, 3
c. Drywell Radiation - High	S	M	E	1, 2, 3
2. SECONDARY CONTAINMENT ISOLATION				
a. Reactor Vessel Water Level - Low[c,d]	S	M	E[a]	1, 2, 3 & *
b. Drywell Pressure - High[b,c,d]	NA	M	Q	1, 2, 3
c. Reactor Building Ventilation Exhaust Radiation - High[c,d]	S	M	E	1, 2, 3 & **
d. Refueling Floor Radiation - High[c,d]	S	M	E	1, 2, 3 & **
3. MAIN STEAM LINE (MSL) ISOLATION				
a. Reactor Vessel Water Level - Low Low	S	M	E[a]	1, 2, 3
b. MSL Tunnel Radiation - High	S	M	E (e)	1, 2, 3
c. MSL Pressure - Low	NA	M	Q	1
d. MSL Flow - High[d]	S	M	E	1, 2, 3
e. MSL Tunnel Temperature - High	NA	E	E	1, 2, 3

TABLE 4.2.A-1 (Continued)

ISOLATION ACTUATION INSTRUMENTATION SURVEILLANCE REQUIREMENTS

Functional Unit	CHANNEL CHECK	CHANNEL FUNCTIONAL TEST	CHANNEL CALIBRATION	Applicable OPERATIONAL MODE(s)
4. REACTOR WATER CLEANUP SYSTEM ISOLATION				
a. Standby Liquid Control System Initiation	NA	E	NA	1, 2, 3
b. Reactor Vessel Water Level - Low	S	M	E[a]	1, 2, 3
5. ISOLATION CONDENSER				
a. Steam Flow - High	NA	M	Q	1, 2, 3
b. ~~Condensate Flow~~ Return Flow - High	NA	M	Q	1, 2, 3
6. HIGH PRESSURE COOLANT INJECTION ISOLATION				
a. Steam Flow - High	NA	M	E ~~Q~~[a]	1, 2, 3
b. Reactor Vessel Pressure - Low	NA	M	~~Q~~[a]	1, 2, 3
c. Area Temperature - High	NA	E	E	1, 2, 3
7. SHUTDOWN COOLING ISOLATION				
a. Reactor Vessel Water Level - Low	S	M	E[a]	3, 4, 5
b. Recirculation Line Water Temperature - High (Cut-in Permissive)	NA	M	Q	1, 2, 3

TABLE 4.2.A-1 (Continued)

ISOLATION ACTUATION INSTRUMENTATION SURVEILLANCE REQUIREMENTS

Functional Unit	CHANNEL CHECK	CHANNEL FUNCTIONAL TEST	CHANNEL CALIBRATION	Applicable OPERATIONAL MODE(s)
4. REACTOR WATER CLEANUP SYSTEM ISOLATION				
a. Standby Liquid Control System Initiation	NA	E	NA	1, 2, 3
b. Reactor Vessel Water Level - Low	S	M	E[a]	1, 2, 3
5. REACTOR CORE ISOLATION COOLING ISOLATION				
a. Steam Flow - High	NA	M	Q	1, 2, 3
b. Reactor Vessel Pressure - Low	NA	M	Q	1, 2, 3
c. Area Temperature - High	NA	E	E	1, 2, 3
6. HIGH PRESSURE COOLANT INJECTION ISOLATION				
a. Steam Flow - High	NA	M	E ~~Q~~[a]	1, 2, 3
b. Reactor Vessel Pressure - Low	NA	M	~~Q~~[a]	1, 2, 3
c. Area Temperature - High	NA	E	E	1, 2, 3
7. RHR SHUTDOWN COOLING MODE ISOLATION				
a. Reactor Vessel Water Level - Low	S	M	E[a]	3, 4, 5
b. Reactor Vessel Pressure - High (Cut-in Permissive)	NA	M	Q	1, 2, 3

TABLE 4.2.A-1 (Continued)

ISOLATION ACTUATION INSTRUMENTATION
SURVEILLANCE REQUIREMENTS

TABLE NOTATION

* During CORE ALTERATIONS or operations with a potential for draining the reactor vessel.

** When handling irradiated fuel in the secondary containment.

(a) Trip units are calibrated at least once per 31 days and transmitters are calibrated at the frequency identified in the table.

(b) This function is not required to be OPERABLE when PRIMARY CONTAINMENT INTEGRITY is not required.

(c) Isolates the reactor building ventilation system and actuates the standby gas treatment system.

(d) These instrument channels will be calibrated using simulated electrical signals once every three months. In addition calibration including the sensors will be performed every 18 months.

TABLE 4.2.A-1 (Continued)

ISOLATION ACTUATION INSTRUMENTATION
SURVEILLANCE REQUIREMENTS

TABLE NOTATION

* During CORE ALTERATIONS or operations with a potential for draining the reactor vessel.

** When handling irradiated fuel in the secondary containment.

(a) Trip units are calibrated at least once per 31 days and transmitters are calibrated at the frequency identified in the table.

(b) This function is not required to be OPERABLE when PRIMARY CONTAINMENT INTEGRITY is not required.

(c) Isolates the reactor building ventilation system and actuates the standby gas treatment system.

(d) Also isolates the control room ventilation system.

(e) These instrument channels will be calibrated using simulated electrical signals once every three months. In addition, calibration including the sensors will be performed every 18 months.

3.2 - LIMITING CONDITIONS FOR OPERATION

B. Emergency Core Cooling Systems (ECCS) Actuation

The ECCS actuation instrumentation CHANNEL(s) shown in Table 3.2.B-1 shall be OPERABLE with their trip setpoints set consistent with the values shown in the Trip Setpoint column.

APPLICABILITY:

As shown in Table 3.2.B-1.

ACTION:

1. With an ECCS actuation instrumentation CHANNEL trip setpoint less conservative than the value shown in the Trip Setpoint column of Table 3.2.B-1, declare the CHANNEL inoperable until the CHANNEL is restored to OPERABLE status with its trip setpoint adjusted consistent with the Trip Setpoint value.
2. With one or more ECCS actuation instrumentation CHANNEL(s) inoperable, take the ACTION required by Table 3.2.B-1.
3. With either ADS TRIP SYSTEM inoperable, restore the inoperable TRIP SYSTEM to OPERABLE status within:
   a. 7 days provided that both the HPCI and IC are OPERABLE, or
   b. 72 hours.

   With the above provisions of this ACTION not met, be in at least HOT

4.2 - SURVEILLANCE REQUIREMENTS

B. ECCS Actuation

1. Each ECCS actuation instrumentation CHANNEL shall be demonstrated OPERABLE by the performance of the CHANNEL CHECK, CHANNEL FUNCTIONAL TEST and CHANNEL CALIBRATION operations for the OPERATIONAL MODE(s) and at the frequencies shown in Table 4.2.B-1.
2. LOGIC SYSTEM FUNCTIONAL TEST(s) ~~and simulated automatic operation~~ of all CHANNEL(s) shall be performed at least once per 18 months.

3.2 - LIMITING CONDITIONS FOR OPERATION

B. Emergency Core Cooling Systems (ECCS) Actuation

The ECCS actuation instrumentation CHANNEL(s) shown in Table 3.2.B-1 shall be OPERABLE with their trip setpoints set consistent with the values shown in the Trip Setpoint column.

APPLICABILITY:

As shown in Table 3.2.B-1.

ACTION:

1. With an ECCS actuation instrumentation CHANNEL trip setpoint less conservative than the value shown in the Trip Setpoint column of Table 3.2.B-1, declare the CHANNEL inoperable until the CHANNEL is restored to OPERABLE status with its trip setpoint adjusted consistent with the Trip Setpoint value.

2. With one or more ECCS actuation instrumentation CHANNEL(s) inoperable, take the ACTION required by Table 3.2.B-1.

3. With either ADS TRIP SYSTEM inoperable, restore the inoperable TRIP SYSTEM to OPERABLE status within:

   a. 7 days provided that both the HPCI and RCIC systems are OPERABLE, or

   b. 72 hours.

   With the above provisions of this ACTION not met, be in at least HOT

4.2 - SURVEILLANCE REQUIREMENTS

B. ECCS Actuation

1. Each ECCS actuation instrumentation CHANNEL shall be demonstrated OPERABLE by the performance of the CHANNEL CHECK, CHANNEL FUNCTIONAL TEST and CHANNEL CALIBRATION operations for the OPERATIONAL MODE(s) and at the frequencies shown in Table 4.2.B-1.

2. LOGIC SYSTEM FUNCTIONAL TESTS (s) ~~and simulated automatic operation~~ of all CHANNEL(s) shall be performed at least once per 18 months.

TABLE 3.2.B-1 (Continued)

ECCS ACTUATION INSTRUMENTATION

Functional Unit	Trip Setpoint[h]	Minimum CHANNEL(s) per Trip Function[a]	Applicable OPERATIONAL MODE(s)	ACTION
3. HIGH PRESSURE COOLANT INJECTION (HPCI) SYSTEM[d]				
a. Reactor Vessel Water Level - Low Low	≥84 inches	4	1, 2, 3	35
b. Drywell Pressure - High[f]	≤2 psig	4	1, 2, 3	35
c. Condensate Storage Tank Level - Low[i]	≥10,000 gal	2	1, 2, 3	35
d. Suppression Chamber Water Level - High[i]	≤15' 5" above bottom of chamber	2	1, 2, 3	35
e. Reactor Vessel Water Level - High (Trip)	≤194 inches	1	1, 2, 3	31
f. HPCI Pump Discharge Flow - Low (Bypass)	≥600 gpm	1	1, 2, 3	33
g. Manual Initiation	NA	1/system	1, 2, 3	34
4. AUTOMATIC DEPRESSURIZATION SYSTEM - TRIP SYSTEM 'A' [d]				
a. Reactor Vessel Water Level - Low Low	≥84 inches	2	1, 2, 3	30
b. Drywell Pressure - High[f]	≤2 psig	2	1, 2, 3	30
c. Initiation Timer	≤120 sec	1	1, 2, 3	31
d. Low Low Level Timer	≤~~8.5~~ 10 min	1	1, 2, 3	31
e. CS Pump Discharge Pressure - High (Permissive)	≥100 psig & ≤150 psig	1/pump	1, 2, 3	31
f. LPCI Pump Discharge Pressure - High (Permissive)	≥100 psig & ≤150 psig	1/pump	1, 2, 3	31

TABLE 3.2.B-1 (Continued)

ECCS ACTUATION INSTRUMENTATION

Functional Unit		Trip Setpoint[h]	Minimum CHANNEL(s) per Trip Function[a]	Applicable OPERATIONAL MODE(s)	ACTION
5.	AUTOMATIC DEPRESSURIZATION SYSTEM - TRIP SYSTEM 'B' [d]				
a.	Reactor Vessel Water Level - Low Low	≥84 inches	2	1, 2, 3	30
b.	Drywell Pressure - High[f]	≤2 psig	2	1, 2, 3	30
c.	Initiation Timer	≤120 sec	1	1, 2, 3	31
d.	Low Low Level Timer	≤8.5 min 10	1	1, 2, 3	31
e.	CS Pump Discharge Pressure - High (Permissive)	≥100 psig & ≤150 psig	1/pump	1, 2, 3	31
f.	LPCI Pump Discharge Pressure - High (Permissive)	≥100 psig & ≤150 psig	1/pump	1, 2, 3	31
6.	LOSS OF POWER				
a.	4.16 kv Emergency Bus Undervoltage (Loss of Voltage)	2930 ± 146 volts decreasing voltage	2/bus	1, 2, 3, 4[e], 5[e]	36
b.	4.16 kv Emergency Bus Undervoltage (Degraded Voltage)	≥ 3784 volts (Unit 2)[g][i] ≥ 3832 volts (Unit 3)[g][i]	2/bus	1, 2, 3, 4[e], 5[e]	36

TABLE 4.2.B-1

ECCS ACTUATION INSTRUMENTATION SURVEILLANCE REQUIREMENTS

Functional Unit	CHANNEL CHECK	CHANNEL FUNCTIONAL TEST	CHANNEL CALIBRATION	Applicable OPERATIONAL MODE(s)
1. CORE SPRAY (CS) SYSTEM				
a. Reactor Vessel Water Level - Low Low	S	M	Q	1, 2, 3, 4[b], 5[b]
b. Drywell Pressure - High[d]	NA	M	Q	1, 2, 3
c. Reactor Vessel Pressure - Low (Permissive)	NA	M	Q	1, 2, 3, 4[b], 5[b]
d. CS Pump Discharge Flow - Low (Bypass)	NA	M	Q E[e]	1, 2, 3, 4[b], 5[b]
2. LOW PRESSURE COOLANT INJECTION (LPCI) SUBSYSTEM				
a. Reactor Vessel Water Level - Low Low	S	M	Q	1, 2, 3, 4[b], 5[b]
b. Drywell Pressure - High[d]	NA	M	Q	1, 2, 3
c. Reactor Vessel Pressure - Low (Permissive)	NA	M	Q	1, 2, 3, 4[b], 5[b]
d. LPCI Pump Discharge Flow - Low (Bypass)	NA	M	Q E[e]	1, 2, 3, 4[b], 5[b]
3. HIGH PRESSURE COOLANT INJECTION (HPCI) SYSTEM[a]				
a. Reactor Vessel Water Level - Low Low	S	M	Q	1, 2, 3
b. Drywell Pressure - High[d]	NA	M	Q	1, 2, 3
c. Condensate Storage Tank Level - Low	NA	M	NA	1, 2, 3
d. Suppression Chamber Water Level - High	NA	M	NA	1, 2, 3
e. Reactor Vessel Water Level - High (Trip)	NA	M	E	1, 2, 3
f. HPCI Pump Discharge Flow - Low (Bypass)	NA	M	Q E	1, 2, 3
g. Manual Initiation	NA	E	NA	1, 2, 3

3.2 - LIMITING CONDITIONS FOR OPERATION

C. ATWS - RPT

The anticipated transient without scram recirculation pump trip (ATWS - RPT) instrumentation CHANNEL(s) shown in Table 3.2.C-1 shall be OPERABLE with their trip setpoints set consistent with the values shown in the Trip Setpoint column.

APPLICABILITY:

OPERATIONAL MODE 1.

ACTION:

1. With an ATWS - RPT instrumentation CHANNEL trip setpoint less conservative than the value shown in the Trip Setpoint column of Table 3.2.C-1, declare the CHANNEL inoperable until the CHANNEL is restored to OPERABLE status with its trip setpoint adjusted consistent with the Trip Setpoint value.
2. With the number of OPERABLE CHANNEL(s) one less than required by the Minimum OPERABLE CHANNEL(s) per TRIP SYSTEM requirement for one or both TRIP SYSTEM(s), restore the inoperable CHANNEL(s) to OPERABLE status within 14 days or be in at least STARTUP within the next 8 hours.
3. With the number of OPERABLE CHANNEL(s) two or more less than required by the Minimum OPERABLE CHANNEL(s) per TRIP SYSTEM requirement for one TRIP SYSTEM and:

e

'INSERT'

INSERT footnote (a) from next page

4.2 - SURVEILLANCE REQUIREMENTS

C. ATWS - RPT

1. Each ATWS - RPT instrumentation CHANNEL shall be demonstrated OPERABLE by the performance of the CHANNEL CHECK, CHANNEL FUNCTIONAL TEST and CHANNEL CALIBRATION operations at the frequencies shown in Table 4.2.C-1.
2. LOGIC SYSTEM FUNCTIONAL TEST(s) ~~and simulated automatic operation~~ of all CHANNEL(s) shall be performed at least once per 18 months.

e

3.2 - LIMITING CONDITIONS FOR OPERATION

a. If the inoperable CHANNEL(s) consist of one reactor vessel water level CHANNEL and one reactor vessel pressure CHANNEL, place both inoperable CHANNEL(s) in the tripped[a] condition within one hour or declare the TRIP SYSTEM inoperable.

b. If the inoperable CHANNEL(s) include two reactor vessel water level CHANNEL(s) or two reactor vessel pressure CHANNEL(s), declare the TRIP SYSTEM inoperable.

5. ~~4.~~ With one TRIP SYSTEM inoperable, restore the inoperable TRIP SYSTEM to OPERABLE status within 72 hours or be in at least STARTUP within the next 6 ~~8~~ hours.

6. ~~5.~~ With both TRIP SYSTEM(s) inoperable, restore at least one TRIP SYSTEM to OPERABLE status within one hour or be in at least STARTUP within the next 6 ~~8~~ hours.

4.2 - SURVEILLANCE REQUIREMENTS

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a The inoperable CHANNEL(s) need not be placed in the tripped condition where this would cause the Trip Function to occur.

INSERT

2. With one level CHANNEL or one pressure CHANNEL inoperable in one or both TRIP SYSTEM(s), within 14 days, either restore the inoperable CHANNEL to OPERABLE status or place the inoperable CHANNEL in the tripped[a] condition. Otherwise, be in STARTUP within the next 6 hours.

3. With two level CHANNELS or two pressure CHANNELS inoperable in one or both TRIP SYSTEM(s), declare the TRIP SYSTEM(s) inoperable.

4. With one level CHANNEL and one pressure CHANNEL inoperable in one or both TRIP SYSTEM(s), restore at least one inoperable CHANNEL to OPERABLE status within 14 days or be in STARTUP within the next 6 hours.

3.2 - LIMITING CONDITIONS FOR OPERATION

C. ATWS - RPT

The anticipated transient without scram recirculation pump trip (ATWS - RPT) instrumentation CHANNEL(s) shown in Table 3.2.C-1 shall be OPERABLE with their trip setpoints set consistent with the values shown in the Trip Setpoint column.

APPLICABILITY:

OPERATIONAL MODE 1.

ACTION:

1. With an ATWS - RPT instrumentation CHANNEL trip setpoint less conservative than the value shown in the Trip Setpoint column of Table 3.2.C-1, declare the CHANNEL inoperable until the CHANNEL is restored to OPERABLE status with the CHANNEL trip setpoint adjusted consistent with the Trip Setpoint value.
2. With the number of OPERABLE CHANNEL(s) one less than required by the Minimum OPERABLE CHANNEL(s) per TRIP SYSTEM requirement for one or both TRIP SYSTEM(s), restore the inoperable CHANNEL(s) to OPERABLE status within 14 days or be in at least STARTUP within the next 8 hours.
3. With the number of OPERABLE CHANNEL(s) two or more less than required by the Minimum OPERABLE CHANNEL(s) per TRIP SYSTEM requirement for one TRIP SYSTEM and:

'INSERT'

INSERT footnote (a) from next page

4.2 - SURVEILLANCE REQUIREMENTS

C. ATWS - RPT

1. Each ATWS - RPT instrumentation CHANNEL shall be demonstrated OPERABLE by the performance of the CHANNEL CHECK, CHANNEL FUNCTIONAL TEST and CHANNEL CALIBRATION operations at the frequencies shown in Table 4.2.C-1.
2. LOGIC SYSTEM FUNCTIONAL TEST(s) ~~and simulated automatic operation~~ of all CHANNEL(s) shall be performed at least once per 18 months.

3.2 - LIMITING CONDITIONS FOR OPERATION

4.2 - SURVEILLANCE REQUIREMENTS

a. If the inoperable CHANNEL(s) consist of one reactor vessel water level CHANNEL and one reactor vessel pressure CHANNEL, place both inoperable CHANNEL(s) in the tripped condition[a] within one hour or declare the TRIP SYSTEM inoperable.

b. If the inoperable CHANNEL(s) include two reactor vessel water level CHANNEL(s) or two reactor vessel pressure CHANNEL(s), declare the TRIP SYSTEM inioperable.

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5. ~~4.~~ With one TRIP SYSTEM inoperable, restore the inoperable TRIP SYSTEM to OPERABLE status within 72 hours or be in at least STARTUP within the next ~~8~~ 6 hours.

6. ~~5.~~ With both TRIP SYSTEM(s) inoperable, restore at least one TRIP SYSTEM to OPERABLE status within one hour or be in at least STARTUP within the next ~~8~~ 6 hours.

MOVE TO PREVIOUS PAGE

a The inoperable CHANNEL(s) need not be placed in the tripped condition where this would cause the Trip Function to occur.

INSERT

2. With one level CHANNEL or one pressure CHANNEL inoperable in one or both TRIP SYSTEM(s), within 14 days, either restore the inoperable CHANNEL to OPERABLE status or place the inoperable CHANNEL in the tripped[a] condition. Otherwise, be in STARTUP within the next 6 hours.

3. With two level CHANNELS or two pressure CHANNELS inoperable in one or both TRIP SYSTEM(s), declare the TRIP SYSTEM(s) inoperable.

4. With one level CHANNEL and one pressure CHANNEL inoperable in one or both TRIP SYSTEM(s), restore at least one inoperable CHANNEL to OPERABLE status within 14 days or be in STARTUP within the next 6 hours.

3.2 - LIMITING CONDITIONS FOR OPERATION

D. Isolation Condenser Actuation

The isolation condenser actuation instrumentation CHANNEL(s) shown in Table 3.2.D-1 shall be OPERABLE with their trip setpoints set consistent with the values shown in the Trip Setpoint column.

APPLICABILITY:

OPERATIONAL MODE(s) 1, 2 and 3 with the reactor steam dome pressure >150 psig.

ACTION:

1. With an isolation condenser actuation instrumentation CHANNEL trip setpoint less conservative than the value shown in the Trip Setpoint column of Table 3.2.D-1, declare the CHANNEL inoperable until the CHANNEL is restored to OPERABLE status with its trip setpoint adjusted consistent with the Trip Setpoint value.
2. With one or more isolation condenser system actuation instrumentation CHANNEL(s) inoperable, take the ACTION required by Table 3.2.D-1.

4.2 - SURVEILLANCE REQUIREMENTS

D. Isolation Condenser Actuation

1. Each isolation condenser actuation instrumentation CHANNEL shall be demonstrated OPERABLE by the performance of the CHANNEL CHECK, CHANNEL FUNCTIONAL TEST and CHANNEL CALIBRATION operations at the frequencies shown in Table 4.2.D-1.
2. LOGIC SYSTEM FUNCTIONAL TEST(s) ~~and simulated automatic operation~~ of all CHANNEL(s) shall be performed at least once per 18 months.

3.2 - LIMITING CONDITIONS FOR OPERATION

D. Reactor Core Isolation Cooling Actuation

The reactor core isolation cooling (RCIC) system actuation instrumentation CHANNEL(s) shown in Table 3.2.D-1 shall be OPERABLE with their trip setpoints set consistent with the values shown in the Trip Setpoint column.

APPLICABILITY:

OPERATIONAL MODE(s) 1, 2 and 3 with the reactor steam dome pressure >150 psig.

ACTION:

1. With a RCIC system actuation instrumentation CHANNEL trip setpoint less conservative than the value shown in the Trip Setpoint column of Table 3.2.D-1, declare the CHANNEL inoperable until the CHANNEL is restored to OPERABLE status with its trip setpoint adjusted consistent with the Trip Setpoint value.
2. With one or more RCIC system actuation instrumentation CHANNEL(s) inoperable, take the ACTION required by Table 3.2.D-1.

4.2 - SURVEILLANCE REQUIREMENTS

D. Reactor Core Isolation Cooling Actuation

1. Each RCIC system actuation instrumentation CHANNEL shall be demonstrated OPERABLE by the performance of the CHANNEL CHECK, CHANNEL FUNCTIONAL TEST and CHANNEL CALIBRATION operations at the frequencies shown in Table 4.2.D-1.
2. LOGIC SYSTEM FUNCTIONAL TEST(s) ~~and simulated automatic operation~~ of all CHANNEL(s) shall be performed at least once per 18 months.

TABLE 3.2.D-1

ISOLATION CONDENSER ACTUATION INSTRUMENTATION

Functional Unit	Trip Setpoint	Minimum CHANNEL(s) per TRIP SYSTEM[a]	ACTION
Reactor Vessel Pressure - High	≤1070 psig for ≤15 seconds	2	40

ACTION

ACTION 40 - With the number of OPERABLE CHANNEL(s) less than required by the Minimum CHANNEL(s) per TRIP SYSTEM requirement:

a. With one CHANNEL inoperable, place the inoperable CHANNEL in the tripped condition within one hour or declare the isolation condenser system inoperable.

b. With more than one CHANNEL inoperable, declare the isolation condenser system inoperable.

(a) A CHANNEL may be placed in an inoperable status for up to 2 hours for required surveillance without placing the TRIP SYSTEM in the tripped condition provided at least one OPERABLE CHANNEL in the same TRIP SYSTEM is monitoring that parameter.

TABLE 4.2.D-1

ISOLATION CONDENSER ACTUATION INSTRUMENTATION SURVEILLANCE REQUIREMENTS

Functional Unit	CHANNEL CHECK	CHANNEL FUNCTIONAL TEST	CHANNEL CALIBRATION
Reactor Vessel Pressure - High	NA	M	~~E~~ Q

3

TABLE ~~4~~.2.D-1 (Continued)

REACTOR CORE ISOLATION COOLING ACTUATION INSTRUMENTATION
~~SURVEILLANCE REQUIREMENTS~~

ACTION

ACTION 40 - With the number of OPERABLE CHANNEL(s) less than required by the Minimum CHANNEL(s) per TRIP SYSTEM requirement:

a. With one CHANNEL inoperable, place the inoperable CHANNEL in the tripped condition within one hour or declare the RCIC system inoperable.

b. With more than one CHANNEL inoperable, declare the RCIC system inoperable.

ACTION 41 - With the number of OPERABLE CHANNEL(s) less than required by the Minimum CHANNEL(s) per TRIP SYSTEM requirement, declare the RCIC system inoperable.

ACTION 42 - With the number of OPERABLE CHANNEL(s) less than required by the Minimum CHANNEL(s) per TRIP SYSTEM requirement, place at least one inoperable CHANNEL in the tripped condition within one hour or declare the RCIC system inoperable.

ACTION 43 - With the number of OPERABLE CHANNEL(s) less than required by the Minimum OPERABLE CHANNEL(s) per TRIP SYSTEM requirement, restore the inoperable CHANNEL to OPERABLE status within 8 hours or declare the RCIC system inoperable.

TABLE 3.2.E-1

CONTROL ROD BLOCK INSTRUMENTATION

Functional Unit	Trip Setpoint	Minimum CHANNEL(s) per Trip Function (i)	Applicable OPERATIONAL MODE(s)	ACTION
1. ROD BLOCK MONITORS[a]				
a. Upscale	As specified in the COLR	2	1 (e)	50
b. Inoperative	NA	2	1 (e)	50
c. Downscale	≥5/125 of full scale	2	1 (e)	50
2. AVERAGE POWER RANGE MONITORS				
a. Flow Biased Neutron Flux - High				
1. Dual Recirculation Loop Operation	≤(0.58W+50) (g)	4	1	51
2. Single Recirculation Loop Operation	≤(0.58W+46.5) (g)	4	1	51
b. Inoperative	NA	4	1, 2, 5 (h)	51
c. Downscale	≥3/125 of full scale	4	1	51
d. Startup Neutron Flux - High	≤12/125 of full scale	4	2, 5 (h)	51

TABLE 3.2.E-1 (Continued)

CONTROL ROD BLOCK INSTRUMENTATION

Functional Unit	Trip Setpoint	Minimum CHANNEL(s) per Trip Function (i)	Applicable OPERATIONAL MODE(s)	ACTION
3. SOURCE RANGE MONITORS				
a. Detector not full in[b]	NA	3	2	51
		2	5	51
b. Upscale[c]	$\leq 1 \times 10^5$ cps	3	2	51
		2	5	51
c. Inoperative[c]	NA	3	2	51
		2	5	51
d. Downscale[d]	~~≥3 cps[f]~~	~~3~~	~~2~~	~~51~~
		~~2~~	~~5~~	~~51~~
4. INTERMEDIATE RANGE MONITORS				
a. Detector not full in~~[e]~~ (d)	NA	6	2, 5	51
b. Upscale	≤108/125 of full scale	6	2, 5	51
c. Inoperative	NA	6	2, 5	51
d. Downscale~~[e]~~ (d)	≥5/125 of full scale	6	2, 5	51

TABLE 3.2.E-1

CONTROL ROD BLOCK INSTRUMENTATION

Functional Unit	Trip Setpoint	Minimum CHANNEL(s) per Trip Function (j)	Applicable OPERATIONAL MODE(s)	ACTION
1. ROD BLOCK MONITORS[a]				
a. Upscale	As specified in COLR	2	1 (e)	50
b. Inoperative	NA	2	1 (e)	50
c. Downscale	≥3/125 of full scale	2	1 (e)	50
2. AVERAGE POWER RANGE MONITORS				
a. Flow Biased Neutron Flux - High				
1. Dual Recirculation Loop Operation	≤(0.58W+50) (g)	4	1	51
2. Single Recirculation Loop Operation	≤(0.58W+46.5) (g)	4	1	51
b. Inoperative	NA	4	1, 2, 5 (h)	51
c. Downscale	≥3/125 of full scale	4	1	51
d. Startup Neutron Flux - High	≤12/125 of full scale	4	2, 5 (h)	51

TABLE 3.2.E-1 (Continued)

CONTROL ROD BLOCK INSTRUMENTATION

(i)

Functional Unit	Trip Setpoint	Minimum CHANNEL(s) per Trip Function	Applicable OPERATIONAL MODE(s)	ACTION
5. SCRAM DISCHARGE VOLUME (SDV)				
a. Water Level - High	~~≤25 gal~~ (Unit 2) ≤29 gal (Unit 3) ≤25 gal	1 per bank	1, 2, 5[a] (f)	52
b. SDV Switch in Bypass	NA	1	5[a] (f)	52

TABLE 3.2.E-1 (Continued)

CONTROL ROD BLOCK INSTRUMENTATION

(i)

Functional Unit	Trip Setpoint	Minimum CHANNEL(s) per Trip Function	Applicable OPERATIONAL MODE(s)	ACTION
3. SOURCE RANGE MONITORS				
a. Detector not full in[b]	NA	3 2	2 5	51 51
b. Upscale[c]	≤1 x 10^5 cps	3 2	2 5	51 51
c. Inoperative[c]	NA	3 2	2 5	51 51
~~d. Downscale[d]~~	~~≥3 cps[f]~~	~~3~~ ~~2~~	~~2~~ ~~5~~	~~51~~ ~~51~~
4. INTERMEDIATE RANGE MONITORS				
a. Detector not full in (d)	NA	6	2, 5	51
b. Upscale	≤108/125 of full scale	6	2, 5	51
c. Inoperative	NA	6	2, 5	51
d. Downscale~~[e]~~	≥3/125 of full scale	6	2, 5	51

TABLE 3.2.E-1 (Continued)

CONTROL ROD BLOCK INSTRUMENTATION

(i)

Functional Unit	Trip Setpoint	Minimum CHANNEL(s) per Trip Function	Applicable OPERATIONAL MODE(s)	ACTION
5. SCRAM DISCHARGE VOLUME (SDV)				
a. Water Level - High	≤25 gal	1 per bank	1, 2, 5~~(a)~~	52
b. SDV Switch in Bypass	NA	1	5~~(b)~~ (f)	52

TABLE 3.2.E-1 (Continued)

CONTROL ROD BLOCK INSTRUMENTATION

TABLE NOTATION

(a) The RBM shall be automatically bypassed when a peripheral control rod is selected or the reference APRM channel indicates less than 30% of RATED THERMAL POWER.

(b) This function shall be automatically bypassed if detector count rate is > 100 cps or the IRM channels are on range 3 or higher.

(c) This function shall be automatically bypassed when the associated IRM channels are on range 8 or higher.

~~(d) This function shall be automatically bypassed when the IRM channels are on range 3 or higher.~~

(d) ~~(e)~~ This function shall be automatically bypassed when the IRM channels are on range 1.

(e) ~~(f)~~ With THERMAL POWER ≥30% of RATED THERMAL POWER.

(f) ~~(g)~~ With more than one control rod withdrawn. Not applicable to control rods removed per Specification 3.10.I or 3.10.J.

(g) ~~(h)~~ The Average Power Range Monitor rod block function is varied as a function of recirculation drive flow (W). The trip setting of this function must be maintained in accordance with Specification 3.11.B. W is equal to the percentage of the drive flow required to produce a rated core flow of 98×10^6 lbs/hr.

(h) ~~(i)~~ ~~Shall be ≥0.7 cps provided signal-to-noise ratio is ≥2.0.~~

(i) ~~(j)~~ Required to be OPERABLE only during SHUTDOWN MARGIN demonstrations performed per Specification 3.12.B.

INSERT

INSERT

A CHANNEL may be placed in an inoperable status for up to 2 hours for required surveillance without placing the CHANNEL in the tripped condition provided the Functional Unit maintains control rod block capability.

TABLE 4.2.E-1

CONTROL ROD BLOCK INSTRUMENTATION SURVEILLANCE REQUIREMENTS

Functional Unit	CHANNEL CHECK	CHANNEL FUNCTIONAL TEST	CHANNEL CALIBRATION[a]	Applicable OPERATIONAL MODE(s)
1. ROD BLOCK MONITORS				
a. Upscale	NA	S/U[b,c], M[c]	Q	1[d]
b. Inoperative	NA	S/U[b,c], M[c]	NA	1[d]
c. Downscale	NA	S/U[b,c], M[c]	Q	1[d]
2. AVERAGE POWER RANGE MONITORS				
a. Flow Biased Neutron Flux - High				
1. Dual Recirculation Loop Operation	NA	S/U[b], M	SA	1
2. Single Recirculation Loop Operation	NA	S/U[b], M	SA	1
b. Inoperative	NA	S/U[b], M	NA	1, 2, 5[e]
c. Downscale	NA	S/U[b], M	SA (e)	1
d. Startup Neutron Flux - High	NA	S/U[b], M	SA (k)	2, 5[e] (j)
3. SOURCE RANGE MONITORS				
a. Detector not full in[f]	NA	S/U[b], W	NA E	2,[i] 5 (i)
b. Upscale[g]	NA	S/U[b], W	E	2,[i] 5
c. Inoperative[g]	NA	S/U[b], W	NA	2,[i] 5
d. Downscale[h]	NA	S/U[b], W	E	2,[i] 5

TABLE 3.2.E-1 (Continued)

CONTROL ROD BLOCK INSTRUMENTATION

TABLE NOTATION

(a) The RBM shall be automatically bypassed when a peripheral control rod is selected or the reference APRM channel indicates less than 30% of RATED THERMAL POWER.

(b) This function shall be automatically bypassed if detector count rate is > 100 cps or the IRM channels are on range 3 or higher.

(c) This function shall be automatically bypassed when the associated IRM channels are on range 8 or higher.

~~(d) This function shall be automatically bypassed when the IRM channels are on range 3 or higher.~~

(d) → ~~(e)~~ This function shall be automatically bypassed when the IRM channels are on range 1.

(e) → ~~(f)~~ With THERMAL POWER ≥ 30% of RATED THERMAL POWER.

(f) → ~~(g)~~ With more than one contr̂ol rod withdrawn. Not applicable to control rods removed per Specification 3.10.I or 3.10.J.

(g) → ~~(h)~~ The Average Power Range Monitor rod block function is varied as a function of recirculation loop flow (W). The trip setting of this function must be maintained in accordance with Specification 3.11.B. W is equal to the percentage of the drive flow required to produce a rated core flow of 98 x 10^6 lbs/hr.

(h) → ~~(i) May be ≥ 0.7 cps provided signal-to-noise ratio is ≥ 2.0.~~

(i) → ~~(j)~~ Required to be OPERABLE only during SHUTDOWN MARGIN demonstrations performed per Specification 3.12.B.

INSERT

INSERT

A CHANNEL may be placed in an inoperable status for up to 2 hours for required surveillance without placing the CHANNEL in the tripped condition provided the Functional Unit maintains control rod block capability.

TABLE 4.2.E-1 (Continued)

CONTROL ROD BLOCK INSTRUMENTATION SURVEILLANCE REQUIREMENTS

Functional Unit	CHANNEL CHECK	CHANNEL FUNCTIONAL TEST	CHANNEL CALIBRATION[a]	Applicable OPERATIONAL MODE(s)
4. INTERMEDIATE RANGE MONITORS				
a. Detector not full in (h)	NA	S/U[b], W	~~NA~~ E	2 (i), 5
b. Upscale	NA	S/U[b], W	E (k)	2 (i), 5
c. Inoperative	NA	S/U[b], W	NA	2 (i), 5
d. Downscale (h)	NA	S/U[b], W	E (k)	2 (i), 5
5. SCRAM DISCHARGE VOLUME (SDV)				
a. Water Level - High	NA	Q	NA	1, 2, 5[e]
b. SDV Switch in Bypass	NA	~~M~~ E	NA	5[e]

TABLE 4.2.E-1

CONTROL ROD BLOCK INSTRUMENTATION
SURVEILLANCE REQUIREMENTS

Functional Unit	CHANNEL CHECK	CHANNEL FUNCTIONAL TEST	CHANNEL CALIBRATION[a]	Applicable OPERATIONAL MODE(s)
1. ROD BLOCK MONITORS				
a. Upscale	NA	S/U[b,c], M[c]	Q	1[d]
b. Inoperative	NA	S/U[b,c], M[c]	NA	1[d]
c. Downscale	NA	S/U[b,c], M[c]	Q	1[d]
2. AVERAGE POWER RANGE MONITORS				
a. Flow Biased Neutron Flux - High				
1. Dual Recirculation Loop Operation	NA	S/U[b], M	SA	1
2. Single Recirculation Loop Operation	NA	S/U[b], M	SA	1
b. Inoperative	NA	S/U[b], M	NA	1, 2, 5[e] e (j)
c. Downscale	NA	S/U[b], M	SA	1
d. Startup Neutron Flux - High	NA	S/U[b], M	SA (k)	2, 5[e] e
3. SOURCE RANGE MONITORS				
a. Detector not full in[f]	NA	S/U[b], W	NA E	2[i] e, 5 (l)
b. Upscale[g]	NA	S/U[b], W	E	2[i] e, 5
c. Inoperative[g]	NA	S/U[b], W	NA	2[i] e, 5
d. Downscale[h]	NA	S/U[b], W	E	2[i], 5 e

TABLE 4.2.E-1 (Continued)

CONTROL ROD BLOCK INSTRUMENTATION
SURVEILLANCE REQUIREMENTS

TABLE NOTATION

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

(b) Within 7 days prior to startup.

(c) Includes reactor manual control "relay select matrix" system input.

(d) With THERMAL POWER ≥30% of RATED THERMAL POWER.

(e) With more than one control rod withdrawn. Not applicable to control rods removed per Specification 3.10.I or 3.10.J.

(f) This function shall be automatically bypassed if detector count rate is >100 cps or the IRM channels are on range 3 or higher.

(g) This function shall be automatically bypassed when the associated IRM channels are on range 8 or higher.

(h) This function shall be automatically bypassed when the IRM channels are on range 3 or higher.

(h) (i) This function shall be automatically bypassed when the IRM channels are on range 1.

(i) (j) The provisions of Specification 4.0.D are not applicable to the CHANNEL FUNCTIONAL TEST and CHANNEL CALIBRATION surveillances for entry into the applicable OPERATIONAL MODE(s) from OPERATIONAL MODE 1 provided the surveillances are performed within 12 hours after such entry

(j) (k) Required to be OPERABLE only during SHUTDOWN MARGIN demonstrations performed per Specification 3.12.B.

(K) Insert

INSERT

The CHANNEL CALIBRATION surveillance requirements shall be performed within 12 hours upon each entry into any OPERATIONAL MODE(s) from OPERATIONAL MODE 1 if not performed within the previous seven days.

TABLE 4.2.E-1 (Continued)

CONTROL ROD BLOCK INSTRUMENTATION SURVEILLANCE REQUIREMENTS

Functional Unit	CHANNEL CHECK	CHANNEL FUNCTIONAL TEST	CHANNEL CALIBRATION[a]	Applicable OPERATIONAL MODE(s)
4. INTERMEDIATE RANGE MONITORS				
a. Detector not full in (h)	NA	S/U[b], W	~~NA~~ E	2, 5 (i)
b. Upscale	NA	S/U[b], W	E (k)	2, 5 (i)
c. Inoperative	NA	S/U[b], W	NA	2, 5 (i)
d. Downscale (h)	NA	S/U[b], W	E (k)	2, 5 (i)
5. SCRAM DISCHARGE VOLUME (SDV)				
a. Water Level - High	NA	Q	NA	1, 2, 5[e]
b. SDV Switch in Bypass	NA	~~M~~ E	NA	5[e]

TABLE 4.2.E-1 (Continued)

CONTROL ROD BLOCK INSTRUMENTATION
SURVEILLANCE REQUIREMENTS

TABLE NOTATION

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

(b) Within 7 days prior to startup.

(c) Includes reactor manual control "relay select matrix" system input.

(d) With THERMAL POWER ≥30% of RATED THERMAL POWER.

(e) With more than one contol rod withdrawn. Not applicable to control rods removed per Specification 3.10.I or 3.10.J.

(f) This function shall be automatically bypassed if detector count rate is >100 cps or the IRM channels are on range 3 or higher.

(g) This function shall be automatically bypassed when the associated IRM channels are on range 8 or higher.

~~(h) This function shall be automatically bypassed when the IRM channels are on range 3 or higher.~~

(h) (i) This function shall be automatically bypassed when the IRM channels are on range 1.

(i) (j) The provisions of Specification 4.0.D are not applicable to the CHANNEL FUNCTIONAL TEST and CHANNEL CALIBRATION surveillances for entry into the applicable OPERATIONAL MODE(s) from OPERATIONAL MODE 1 provided the surveillances are performed within 12 hours after such entry.

(j) (k) Required to be OPERABLE only during SHUTDOWN MARGIN demonstrations performed per Specification 3.12.B.

(k) insert

INSERT

The CHANNEL CALIBRATION surveillance requirements shall be performed within 12 hours upon each entry into any OPERATIONAL MODE(s) from OPERATIONAL MODE 1 if not performed within the previous seven days.

TABLE 3.2.F-1

ACCIDENT MONITORING INSTRUMENTATION

INSTRUMENTATION	Required CHANNEL(s)	Minimum CHANNEL(s)	Applicable OPERATIONAL MODE(s)	ACTION
1. Reactor Vessel Pressure	2	1	1, 2	60
2. Reactor Vessel Water Level	2	1	1, 2	60
3 Torus Water Level ~~Wide Range~~	2	1	1, 2	60
4. Torus Water Temperature	2	1	1, 2	60
5. Drywell Pressure - Wide Range	2	1	1, 2	60
6. Drywell Pressure - Narrow Range	2	1	1, 2	60
7. Drywell Air Temperature	2	1	1, 2	60
8. Drywell Oxygen Concentration - Analyzer and Monitor	2	1	1, 2	62
9. Drywell Hydrogen Concentration - Analyzer and Monitor	2	1	1, 2	62
10 Safety & Relief Valve Position Indicators - Acoustic & Temperature	2/valve (1 each)	1/valve	1, 2	63
11. (Source Range) Neutron Monitors	2	2	1,2	60
12. Drywell Radiation Monitors	2	2	1, 2, 3	61
13. Torus Pressure	2 (a)	1	1, 2	60

(a) This function is shared with Drywell Pressure - Wide Range and Drywell Pressure - Narrow Range

TABLE 3.2.F-1 (Continued)

ACCIDENT MONITORING INSTRUMENTATION

ACTION

ACTION 60 -

a. With the number of OPERABLE accident monitoring instrumentation CHANNEL(s) less than the Required CHANNEL(s) shown in Table 3.2.F-1, restore the inoperable CHANNEL(s) to OPERABLE status within 30 days or be in at least HOT SHUTDOWN within the next 12 hours.

b. With the number of OPERABLE accident monitoring instrumentation CHANNEL(s) less than the Minimum CHANNEL(s) shown in Table 3.2.F-1, restore the inoperable CHANNEL(s) to OPERABLE status within 48 hours or be in at least HOT SHUTDOWN within the next 12 hours.

ACTION 61-

With the number of OPERABLE accident monitoring instrumentation CHANNEL(s) less than the Minimum CHANNEL(s) shown in Table 3.2.F-1, initiate the preplanned alternate method of monitoring the appropriate parameter(s) within 72 hours, and:

a. Either restore the inoperable CHANNEL(s) to OPERABLE status within 7 days of the event, or

b. Prepare and submit a Special Report to the Commission pursuant to Specification ~~6.6.C.3~~ 6.9.B within 30 days following the event outlining the action taken, the cause of the inoperability and the plans and schedule for restoring the system to OPERABLE status.

ACTION 62-

a. With the number of OPERABLE accident monitoring instrumentation CHANNEL(s) one less than the Required CHANNEL(s) shown in Table 3.2.F-1, restore the inoperable CHANNEL(s) to OPERABLE status within 30 days or be in at least HOT SHUTDOWN within the next 12 hours.

b. With the number of OPERABLE accident monitoring instrumentation CHANNEL(s) less than the Minimum CHANNEL(s) shown in Table 3.2.F-1; and provided the high radiation sampling system (HRSS) combustible gas monitoring capability for the drywell is OPERABLE; restore the inoperable CHANNEL(s) to OPERABLE status within 30 days or be in at least HOT SHUTDOWN within the next 12 hours.

c. With the number of OPERABLE accident monitoring instrumentation CHANNEL(s) less than the Minimum CHANNEL(s) shown in Table 3.2.F-1; and the HRSS combustible gas monitoring capability for the drywell inoperable; restore at least one inoperable CHANNEL to OPERABLE status within 7 days or be in at least HOT SHUTDOWN within the next 12 hours.

TABLE 3.2.F-1

ACCIDENT MONITORING INSTRUMENTATION

INSTRUMENTATION	Required CHANNEL(s)	Minimum CHANNEL(s)	Applicable OPERATIONAL MODE(s)	ACTION
1. Reactor Vessel Pressure	2	1	1, 2	60
2. Reactor Vessel Water Level	2	1	1, 2	60
3 Torus Water Level ~~Wide Range~~	2	1	1, 2	60
4. Torus Water Temperature	2	1	1, 2	60
5. Drywell Pressure - Wide Range	2	1	1, 2	60
6. Drywell Pressure - Narrow Range	2	1	1, 2	60
7. Drywell Air Temperature	2	1	1, 2	60
8. Drywell Oxygen Concentration - Analyzer and Monitor	2	1	1, 2	62
9. Drywell Hydrogen Concentration - Analyzer and Monitor	2	1	1, 2	62
10. Safety & Relief Valve Position Indicators - Acoustic & Temperature	2/valve (1 each)	1/valve	1, 2	63
11. (Source Range) Neutron Monitors	2	2	1, 2	60
12. Drywell Radiation Monitors	2	2	1, 2, 3	61
13. Torus Air Temperature	2	1	1, 2	60
14. Torus Pressure	2[a]	1	1, 2	60

(a) this function is shared with Drywell Pressure - Wide Range and Drywell Pressure - Narrow Range.

TABLE 3.2.F-1 (Continued)

ACCIDENT MONITORING INSTRUMENTATION

ACTION 63 -

a. With the number of OPERABLE accident monitoring instrumentation CHANNEL(s) less than the Required CHANNEL(s) shown in Table 3.2.F-1, restore the inoperable CHANNEL(s) to OPERABLE status prior to startup from a COLD SHUTDOWN of longer than 72 hours.

b. With the number of OPERABLE accident monitoring instrumentation CHANNEL(s) less than the Minimum CHANNEL(s) shown in Table 3.2.F-1, restore the inoperable CHANNEL(s) to OPERABLE status within 30 days or be in at least HOT SHUTDOWN within the next 12 hours.

at least one of

TABLE 3.2.F-1 (Continued)

ACCIDENT MONITORING INSTRUMENTATION

ACTION

ACTION 60 - a. With the number of OPERABLE accident monitoring instrumentation CHANNEL(s) less than the Required CHANNEL(s) shown in Table 3.2.F-1, restore the inoperable CHANNEL(s) to OPERABLE status within 30 days or be in at least HOT SHUTDOWN within the next 12 hours.

b. With the number of OPERABLE accident monitoring instrumentation CHANNEL(s) less than the Minimum CHANNEL(s) shown in Table 3.2.F-1, restore the inoperable CHANNEL(s) to OPERABLE status within 48 hours or be in at least HOT SHUTDOWN within the next 12 hours.

ACTION 61- With the number of OPERABLE accident monitoring instrumentation CHANNEL(s) less than the Minimum CHANNEL(s) shown in Table 3.2.F-1, initiate the preplanned alternate method of monitoring the appropriate parameter(s) within 72 hours, and:

a. Either restore the inoperable CHANNEL(s) to OPERABLE status within 7 days of the event, or

b. Prepare and submit a Special Report to the Commission pursuant to Specification ~~6.6.C.3~~ 6.9.B within 30 days following the event outlining the action taken, the cause of the inoperability and the plans and schedule for restoring the system to OPERABLE status.

ACTION 62- a. With the number of OPERABLE accident monitoring instrumentation CHANNEL(s) one less than the Required CHANNEL(s) shown in Table 3.2.F-1, restore the inoperable CHANNEL(s) to OPERABLE status within 30 days or be in at least HOT SHUTDOWN within the next 12 hours.

b. With the number of OPERABLE accident monitoring instrumentation CHANNEL(s) less than the Minimum CHANNEL(s) shown in Table 3.2.F-1; and provided the high radiation sampling system (HRSS) combustible gas monitoring capability for the drywell is OPERABLE; restore the inoperable CHANNEL(s) to OPERABLE status within 30 days or be in at least HOT SHUTDOWN within the next 12 hours.

c. With the number of OPERABLE accident monitoring instrumentation CHANNEL(s) less than the Minimum CHANNEL(s) shown in Table 3.2.F-1; and the HRSS combustible gas monitoring capability for the drywell inoperable; restore at least one inoperable CHANNEL to OPERABLE status within 7 days or be in at least HOT SHUTDOWN within the next 12 hours.

TABLE 4.2.F-1

ACCIDENT MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

INSTRUMENTATION	CHANNEL CHECK	CHANNEL CALIBRATION	Applicable OPERATIONAL MODE(s)
1. Reactor Vessel Pressure	M	SA E	1, 2
2. Reactor Vessel Water Level	M	E	1, 2
3 Torus Water Level	M	A E	1, 2
4. Torus Water Temperature	M	E	1, 2
5. Drywell Pressure - Wide Range	M	E	1, 2
6. Drywell Pressure - Narrow Range	M	Q E	1, 2
7. Drywell Air Temperature	M	E	1, 2
~~8. Drywell Oxygen Concentration - Analyzer and Monitor~~	~~M~~	~~E~~	~~1, 2~~
8 ~~9.~~ Drywell Hydrogen /Oxygen Concentration - Analyzer and Monitor	M	Q(a)	1, 2
9 ~~10.~~ Safety/Relief Valve Position Indicators - Acoustic & Temperature	M (c)	E	1, 2
10 ~~11.~~ (Source Range) Neutron Monitors	M	E(a) Q(b)	1, 2
11 ~~12.~~ Drywell Radiation Monitors	M	E(b) (a)	1, 2, 3
12. Torus Pressure	M	Q	1, 2

TABLE 3.2.F-1 (Continued)

ACCIDENT MONITORING INSTRUMENTATION

ACTION 63 -

a. With the number of OPERABLE accident monitoring instrumentation CHANNEL(s) less than the Required CHANNEL(s) shown in Table 3.2.F-1, restore the inoperable CHANNEL(s) to OPERABLE status prior to startup from a COLD SHUTDOWN of longer than 72 hours.

b. With the number of OPERABLE accident monitoring instrumentation CHANNEL(s) less than the Minimum CHANNEL(s) shown in Table 3.2.F-1, restore the inoperable CHANNEL(s) to OPERABLE status within 30 days or be in at least HOT SHUTDOWN within the next 12 hours.

at least one of

TABLE 4.2.F-1 (Continued)

ACCIDENT MONITORING INSTRUMENTATION
SURVEILLANCE REQUIREMENTS

TABLE NOTATION

~~(a) Using sample gas containing:~~

~~a. One volume percent hydrogen, balance nitrogen.~~
~~b. Four volume percent hydrogen, balance nitrogen.~~

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

(b) ~~(c)~~ Neutron detectors may be excluded from the CHANNEL CALIBRATION.

(c) CHANNEL CHECK of the Acoustic Monitors shall consist of verifying the instrument threshold levels.

TABLE 4.2.F-1

ACCIDENT MONITORING INSTRUMENTATION
SURVEILLANCE REQUIREMENTS

INSTRUMENTATION	CHANNEL CHECK	CHANNEL CALIBRATION	Applicable OPERATIONAL MODE(s)
1. Reactor Vessel Pressure	M	E	1, 2
2. Reactor Vessel Water Level	M	E	1, 2
3 Torus Water Level ~~Wide Range~~	M	E	1, 2
4. Torus Water Temperature	M	E	1, 2
5. Drywell Pressure - Wide Range	M	E	1, 2
6. Drywell Pressure - Narrow Range	M	E	1, 2
7. Drywell Air Temperature	M	E	1, 2
~~8. Drywell Oxygen Concentration - Analyzer and Monitor~~	~~M~~	~~E~~	~~1, 2~~
8. ~~9.~~ Drywell Hydrogen /Oxygen Concentration - Analyzer and Monitor	M	Q	1, 2
9. ~~10.~~ Safety & Relief Valve Position Indicators - Acoustic & Temperature	M	E	1, 2
10. ~~11.~~ (Source Range) Neutron Monitors	M	E (b)	1, 2
11. ~~12.~~ Drywell Radiation Monitors	M	E (a)	1, 2, 3
12. ~~13.~~ Torus Air Temperature	M	E	1, 2
13. Torus Pressure	M	E	1, 2

3.2 - LIMITING CONDITIONS FOR OPERATION

G. Source Range Monitoring

At least the following source range monitor (SRM) channels shall be OPERABLE:

a. In OPERATIONAL MODE 2[a], three.

b. In OPERATIONAL MODE 3 and 4, two.

APPLICABILITY:

OPERATIONAL MODE(s) 2[a], 3, and 4.

ACTION:

1. In OPERATIONAL MODE 2[a] with one of the above required source range monitor CHANNEL(s) inoperable, at least 3 source range monitor CHANNEL(s) shall be restored to OPERABLE status within 4 hours or the reactor shall be in at least HOT SHUTDOWN within the next 12 hours.

2. In OPERATIONAL MODE(s) 3 or 4 with one or more of the above required source range monitor CHANNEL(s) inoperable, verify all insertable control rods to be fully inserted in the core and lock the reactor mode switch in the Shutdown position within one hour.

4.2 - SURVEILLANCE REQUIREMENTS

G. Source Range Monitoring

Each of the required source range monitor CHANNEL(s) shall be demonstrated OPERABLE by:

1. Verifying, prior to withdrawal of the control rods, that the SRM count rate is ≥3 cps[b] with the detector fully inserted.

2. Performance of a CHANNEL CHECK at least once per:

   a. 12 hours in OPERATIONAL MODE 2[a], and

   b. 24 hours in OPERATIONAL MODE(s) 3 or 4.

3. Performance of a CHANNEL FUNCTIONAL TEST:

   a. Within 7 days prior to startup, and

   b. At least once per 31 days[c]. (b)

4. Performance of a CHANNEL CALIBRATION[d] at least once per 18 months[c].
(c) (b)

---

a With IRM's on range 2 or below.

~~b May be reduced to ≥0.7 cps provided the signal to noise ratio is ≥2.0.~~

(b) c The provisions of Specification 4.0.D are not applicable for entry into the applicable OPERATIONAL MODE(s) from OPERATIONAL MODE 1, provided the surveillance is performed within 12 hours after such entry.

(c) d Neutron detectors may be excluded from the CHANNEL CALIBRATION.

TABLE 4.2.F-1 (Continued)

ACCIDENT MONITORING INSTRUMENTATION
SURVEILLANCE REQUIREMENTS

TABLE NOTATION

(a) Using sample gas containing:
  a. One volume percent hydrogen, balance nitrogen.
  b. Four volume percent hydrogen, balance nitrogen.

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

(b.) (c) Neutron detectors may be excluded from the CHANNEL CALIBRATION.

3.2 - LIMITING CONDITIONS FOR OPERATION

Gas

H. Explosive ~~Mixture~~ Monitoring

gas

The explosive monitoring instrumentation CHANNEL(s) shown in Table 3.2.H-1 shall be OPERABLE with their alarm/trip setpoints set to ensure that the limits of Specification 3.8.H are not exceeded.

APPLICABILITY:

During offgas holdup system operation.

ACTION:

1. With an explosive gas monitoring instrumentation CHANNEL alarm/trip setpoint less conservative than required by the above specification, declare the CHANNEL inoperable and take the ACTION shown in Table 3.2.H-1.
2. With less than the minimum number of explosive gas monitoring instrumentation CHANNEL(s) OPERABLE, take the ACTION shown in Table 3.2.H-1. Restore the inoperable instrumentation to OPERABLE status within 30 days and, if unsuccessful, prepare and submit a Special Report to the Commission pursuant to Specification ~~6.6.C.3~~ 6.9.B to explain why this inoperability was not corrected in a timely manner.
3. The provisions of Specification 3.0.C are not applicable.

4.2 - SURVEILLANCE REQUIREMENTS

H. Explosive ~~Mixture~~ Monitoring

Gas

Each explosive gas monitoring instrumentation CHANNEL shall be demonstrated OPERABLE by the performance of the CHANNEL CHECK, CHANNEL FUNCTIONAL TEST and CHANNEL CALIBRATION operations at the frequencies shown in Table 4.2.H-1.

3.2 - LIMITING CONDITIONS FOR OPERATION

G. Source Range Monitoring

At least the following source range monitor (SRM) channels shall be OPERABLE:

a. In OPERATIONAL MODE 2[a], three.

b. In OPERATIONAL MODE 3 and 4, two.

APPLICABILITY:

OPERATIONAL MODE(s) 2[a], 3, and 4.

ACTION:

1. In OPERATIONAL MODE 2[a] with one of the above required source range monitor CHANNEL(s) inoperable, at least 3 source range monitor CHANNEL(s) shall be restored to OPERABLE status within 4 hours or the reactor shall be in at least HOT SHUTDOWN within the next 12 hours.

2. In OPERATIONAL MODE(s) 3 or 4 with one or more of the above required source range monitor CHANNEL(s) inoperable, verify all insertable control rods to be fully inserted in the core and lock the reactor mode switch in the Shutdown position within one hour.

4.2 - SURVEILLANCE REQUIREMENTS

G. Source Range Monitoring

Each of the required source range monitor CHANNEL(s) shall be demonstrated OPERABLE by:

1. Verifying, prior to withdrawal of the control rods, that the SRM count rate is ≥ 3 cps[b] with the detector fully inserted.

2. Performance of a CHANNEL CHECK at least once per:

   a. 12 hours in OPERATIONAL MODE 2[a], and

   b. 24 hours in OPERATIONAL MODE(s) 3 or 4.

3. Performance of a CHANNEL FUNCTIONAL TEST:

   a. Within 7 days prior to startup, and

   b. At least once per 31 days[c]. (b)

4. Performance of a CHANNEL CALIBRATION[d] at least once per 18 months[c]. (c) (b)

---

a With IRM's on range 2 or below.

~~b May be reduced to ≥ 0.7 cps provided the signal to noise ratio is ≥ 2.0.~~

(b) c The provisions of Specification 4.0.D are not applicable for entry into the applicable OPERATIONAL MODE(s) from OPERATIONAL MODE 1, provided the surveillance is performed within 12 hours after such entry.

(c) d Neutron detectors may be excluded from the CHANNEL CALIBRATION.

3.2 - LIMITING CONDITIONS FOR OPERATION

H. Explosive Gas Monitoring

The explosive gas monitoring instrumentation CHANNEL(s) shown in Table 3.2.H-1 shall be OPERABLE with their alarm/trip setpoints set to ensure that the limits of Specification 3.8.H are not exceeded.

APPLICABILITY:

During offgas holdup system operation.

ACTION:

1. With an explosive gas monitoring instrumentation CHANNEL alarm/trip setpoint less conservative than required by the above specification, declare the CHANNEL inoperable and take the ACTION shown in Table 3.2.H-1.
2. With less than the minimum number of explosive gas monitoring instrumentation CHANNEL(s) OPERABLE, take the ACTION shown in Table 3.2.H-1. Restore the inoperable instrumentation to OPERABLE status within 30 days and, if unsuccessful, prepare and submit a Special Report to the Commision pursuant to Specification ~~6.6.C.3~~ 6.9.B to explain why this inoperability was not corrected in a timely manner.
3. The provisions of Specification 3.0.C are not applicable.

4.2 - SURVEILLANCE REQUIREMENTS

H. Explosive Gas Monitoring

Each explosive gas monitoring instrumentation CHANNEL shall be demonstrated OPERABLE by the performance of the CHANNEL CHECK, CHANNEL FUNCTIONAL TEST and CHANNEL CALIBRATION operations at the frequencies shown in Table 4.2.H-1.

TABLE 4.2.H-1

GAS

EXPLOSIVE MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

Functional Unit	CHANNEL CHECK	CHANNEL FUNCTIONAL TEST	CHANNEL CALIBRATION
MAIN CONDENSER OFFGAS TREATMENT SYSTEM EXPLOSIVE GAS MONITORING SYSTEM			
Hydrogen Monitor	D	M	Q[a]

~~a The CHANNEL CALIBRATION shall include the use of standard gas samples containing a nominal:~~
1. ~~One volume percent hydrogen, balance nitrogen, and~~
2. ~~Four volume percent hydrogen, balance nitrogen.~~

3.2 - LIMITING CONDITIONS FOR OPERATION

I. Suppression Chamber and Drywell Spray Actuation

The suppression chamber and drywell spray actuation instrumentation CHANNEL(s) shown in Table 3.2.I-1 shall be OPERABLE with their trip setpoints set consistent with the values shown in the Trip Setpoint column of Table 3.2.I-1.

APPLICABILITY:

OPERATIONAL MODE(s) 1, 2 and 3.

ACTION:

With a suppression chamber and drywell spray actuation instrumentation CHANNEL trip setpoint less conservative than the value shown in the Trip Setpoint column of Table 3.2.I-1, declare the CHANNEL inoperable and take the ACTION shown in Table 3.2.I-1.

4.2 - SURVEILLANCE REQUIREMENTS

I. Suppression Chamber and Drywell Spray Actuation

1. Each suppression chamber and drywell spray actuation instrumentation CHANNEL shall be demonstrated OPERABLE by the performance of the CHANNEL CHECK, CHANNEL FUNCTIONAL TEST and CHANNEL CALIBRATION operations at the frequencies shown in Table 4.2.I-1.

2. LOGIC SYSTEM FUNCTIONAL TEST(s) ~~and simulated automatic operation~~ of all CHANNEL(s) shall be performed at least once per 18 months.

TABLE 4.2.H-1

EXPLOSIVE GAS MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

Functional Unit	CHANNEL CHECK	CHANNEL FUNCTIONAL TEST	CHANNEL CALIBRATION
MAIN CONDENSER OFFGAS TREATMENT SYSTEM EXPLOSIVE GAS MONITORING SYSTEM			
Hydrogen Monitor	D	M	Q[a]

---

a The CHANNEL CALIBRATION shall include the use of standard gas samples containing a nominal:
1. One volume percent hydrogen, balance nitrogen, and
2. Four volume percent hydrogen, balance nitrogen.

TABLE 3.2.I-1

SUPPRESSION CHAMBER AND DRYWELL SPRAY ACTUATION INSTRUMENTATION

Functional Unit	Trip Setpoint[a]	Minimum CHANNEL(s) per TRIP SYSTEM (c)	ACTION
1. Drywell Pressure - High (Permissive)	$0.5 \leq \rho \leq 1.5$ psig	2	80
2. Reactor Vessel Water Level -Low (Permissive)	≥ -48 inches	1	80

ACTION

ACTION 80 -

a. With the number of OPERABLE CHANNEL(s) less than required by the Minimum OPERABLE CHANNEL(s) per TRIP SYSTEM requirement for one TRIP SYSTEM, place at least one inoperable CHANNEL in the tripped condition[b] within one hour or declare the suppression chamber and drywell sprays inoperable.

b. With the number of OPERABLE CHANNEL(s) less than required by the Minimum OPERABLE CHANNEL(s) per TRIP SYSTEM requirement for both TRIP SYSTEM(s), declare the suppression chamber and drywell sprays inoperable.

---

a Reactor vessel water level settings are expressed in inches above the top of active fuel (which is 360 inches above vessel zero).

b If an instrument is inoperable, it shall be placed (or simulated) in a tripped condition so that it will not prevent a containment spray.

INSERT

INSERT

c A CHANNEL may be placed in an inoperable status for up to 2 hours for required surveillance without placing the CHANNEL in the tripped condition provided the Functional Unit maintains Suppression Chamber and Drywell Spray Actuation capability.

3.2 - LIMITING CONDITIONS FOR OPERATION

I. Suppression Chamber and Drywell Spray Actuation

The Suppression Chamber and Drywell Spray Actuation instrumentation CHANNEL(s) shown in Table 3.2.I-1 shall be OPERABLE with their trip setpoints set consistent with the values shown in the Trip Setpoint column of Table 3.2.I-1.

APPLICABILITY:

OPERATIONAL MODE(s) 1, 2 & 3.

ACTION:

With a Suppression Chamber and Drywell Spray Actuation instrumentation CHANNEL trip setpoint less conservative than the value shown in the Trip Setpoint column of Table 3.2.I-1, declare the CHANNEL inoperable and take the ACTION shown in Table 3.2.I-1.

4.2 - SURVEILLANCE REQUIREMENTS

I. Suppression Chamber and Drywell Spray Actuation

1. Each Suppression Chamber and Drywell Spray Actuation instrumentation CHANNEL shall be demonstrated OPERABLE by the performance of the CHANNEL CHECK, CHANNEL FUNCTIONAL TEST and CHANNEL CALIBRATION operations at the frequencies shown in Table 4.2.I-1.

2. LOGIC SYSTEM FUNCTIONAL TEST(s) ~~and simulated automatic operation~~ of all CHANNEL(s) shall be performed at least once per 18 months.

TABLE 3.2.I-1

SUPPRESSION CHAMBER AND DRYWELL SPRAY ACTUATION INSTRUMENTATION

Functional Unit	Trip Setpoint[a]	Minimum CHANNEL(s) per TRIP SYSTEM (c)	ACTION
1. Drywell Pressure - High (Permissive)	$0.5 \leq p \leq 1.5$ psig	2	80
2. Reactor Vessel Water Level - Low (Permissive)	$\geq$ -48 inches	1	80

ACTION

ACTION 80 -

a. With the number of OPERABLE CHANNEL(s) less than required by the Minimum OPERABLE CHANNEL(s) per TRIP SYSTEM requirement for one TRIP SYSTEM, place at least one inoperable CHANNEL in the tripped condition[B] within one hour or declare the Suppression Chamber and Drywell Spray Actuation mode of the Residual Heat Removal system inoperable.

b. With the number of OPERABLE CHANNEL(s) less than required by the Minimum OPERABLE CHANNEL(s) per TRIP SYSTEM requirement for both TRIP SYSTEM(s), declare the Suppression Chamber and Drywell Spray Actuation mode of the Residual Heat Removal system inoperable.

---

a Reactor vessel water level settings are expressed in inches above the top of active fuel (which is 360 inches above vessel zero).

b If an instrument is inoperable, it shall be placed (or simulated) in a tripped condition so that it will not prevent a containment spray.

INSERT

INSERT

c A CHANNEL may be placed in an inoperable status for up to 2 hours for required surveillance without placing the CHANNEL in the tripped condition provided the Functional Unit maintains Suppression Chamber and Drywell Spray Actuation capability.

3.2 - LIMITING CONDITIONS FOR OPERATION

J. Feedwater Pump Trip

The feedwater pump trip instrumentation CHANNEL(s) shown in Table 3.2.J-1 shall be OPERABLE with their trip setpoints set consistent with the values shown in the Trip Setpoint column of Table 3.2.J-1.

APPLICABILITY:

OPERATIONAL MODE 1.

ACTION:

With a feedwater pump trip instrumentation CHANNEL trip setpoint less conservative than value shown in the Trip Setpoint column of Table 3.2.J-1, declare the CHANNEL inoperable and take the ACTION shown in Table 3.2.J-1.

4.2 - SURVEILLANCE REQUIREMENTS

J. Feedwater Pump Trip

1. Each feedwater pump trip instrumentation CHANNEL shall be demonstrated OPERABLE by the performance of the CHANNEL CHECK, CHANNEL FUNCTIONAL TEST and CHANNEL CALIBRATION operations at the frequencies shown in Table 4.2.J-1.

2. LOGIC SYSTEM FUNCTIONAL TEST(s) ~~and simulated automatic operation~~ of all CHANNEL(s) shall be performed at least once per 18 months.

TABLE 3.2.J-1

FEEDWATER PUMP TRIP INSTRUMENTATION

Functional Unit	Trip Setpoint[a]	Minimum CHANNEL(s) (b)	ACTION
Reactor Vessel Water Level -High	≤ 201 inches	2	90

ACTION

ACTION 90 -

a. With the number of OPERABLE CHANNEL(s) one less than required by the Minimum CHANNEL(s) requirement, restore the inoperable CHANNEL to OPERABLE status within 7 days or place the inoperable CHANNEL in the tripped condition within the next 8 hours.

b. With the number of OPERABLE CHANNEL(s) two less than required by the Minimum CHANNEL(s) requirement, restore at least one of the inoperable CHANNEL(s) to OPERABLE status within 72 hours or be in at least STARTUP within the next 8 hours.

---

a Reactor vessel water level settings are expressed in inches above the top of active fuel (which is 360 inches above vessel zero).

INSERT

INSERT

b A CHANNEL may be placed in an inoperable status for up to 2 hours for required surveillance without placing the CHANNEL in the tripped condition.

3.2 - LIMITING CONDITIONS FOR OPERATION

J. Feedwater Pump Trip

The feedwater pump trip instrumentation CHANNEL(s) shown in Table 3.2.J-1 shall be OPERABLE with their trip setpoints set consistent with the values shown in the Trip Setpoint column of Table 3.2.J-1.

APPLICABILITY:

OPERATIONAL MODE 1.

ACTION:

With a feedwater pump trip instrumentation CHANNEL trip setpoint less conservative than the value shown in the Trip Setpoint column of Table 3.2.J-1, declare the CHANNEL inoperable and take the ACTION shown in Table 3.2.J-1.

4.2 - SURVEILLANCE REQUIREMENTS

J. Feedwater Pump Trip

1. Each feedwater pump trip instrumentation CHANNEL shall be demonstrated OPERABLE by the performance of the CHANNEL CHECK, CHANNEL FUNCTIONAL TEST and CHANNEL CALIBRATION operations at the frequencies shown in Table 4.2.J-1.

2. LOGIC SYSTEM FUNCTIONAL TEST(s) ~~and simulated automatic operation~~ of all CHANNEL(s) shall be performed at least once per 18 months.

TABLE 3.2.J-1

FEEDWATER PUMP TRIP INSTRUMENTATION

Functional Unit	Trip Setpoint[a]	Minimum CHANNEL(s) (b)	ACTION
Reactor Vessel Water Level -High	≤ 201 inches	2	90

ACTION

ACTION 90 - a. With the number of OPERABLE CHANNEL(s) one less than required by the Minimum CHANNEL(s) requirement, restore the inoperable CHANNEL to OPERABLE status within 7 days or place the inoperable CHANNEL in the tripped condition within the next 8 hours.

b. With the number of OPERABLE CHANNEL(s) two less than required by the Minimum CHANNEL(s) requirement, restore at least one of the inoperable CHANNEL(s) to OPERABLE status within 72 hours or be in at least STARTUP within the next 8 hours.

---

a Reactor vessel water level settings are expressed in inches above the top of active fuel (which is 360 inches above vessel zero).

INSERT

INSERT

b A CHANNEL may be placed in an inoperable status for up to 2 hours for required surveillance without placing the CHANNEL in the tripped condition.

BASES

3/4.2.E Control Rod Block Actuation Instrumentation

The control rod block functions are provided to prevent excessive control rod withdrawal so that the MINIMUM CRITICAL POWER RATIO (MCPR) does not go below the MCPR fuel cladding integrity Safety Limit. During shutdown conditions, control rod block instrumentation initiates withdrawal blocks to ensure that all control rods remain inserted to prevent inadvertent criticality.

The trip logic for this function is one-out-of-n; e.g., any trip on one of the six average power range monitors (APRMs), eight intermediate range monitors (IRMs), or four source range monitors (SRMs), will result in a rod block. The minimum instrument CHANNEL requirements assure sufficient instrumentation to assure that the single failure criterion is met. The minimum instrument CHANNEL requirements for the rod block monitor may be reduced by one for a short period of time to allow for maintenance, testing, or calibration.

The APRM rod block function is flow-biased and prevents a significant reduction in MCPR, especially during operation at reduced flow. The APRM provides gross core protection, i.e., limits the gross withdrawal of control rods in the normal withdrawal sequence.

In the REFUEL and STARTUP/HOT STANDBY OPERATIONAL MODE(s), the APRM rod block function setpoint is significantly reduced to provide the same type of protection in the REFUEL and STARTUP/HOT STANDBY OPERATIONAL MODE(s) as the APRM flow-biased rod block does in the RUN OPERATIONAL MODE, i.e., prevents control rod withdrawal before a scram is reached.

The rod block monitor (RBM) function provides local protection of the core, i.e., the prevention of transition boiling in a local region of the core for a single rod withdrawal error. The trip setting is flow-biased. At low power, the worst-case withdrawal of a single control rod without rod block action will not violate the fuel cladding integrity Safety Limit. Thus the RBM rod block function is not required below the specified power level. The worst-case single control rod withdrawal error is analyzed for each reload to assure that, with the specific trip settings, rod withdrawal is blocked before the MCPR reaches the fuel cladding integrity Safety Limit. An RBM "inoperative" actuates on several inputs including: (1) nulling, (2) failure to null, (3) <50% assigned inputs, (4) card pulled, (5) no rod selected, (6) >1 rod selected and (7) switch not in operate. e

The IRM rod block function provides local as well as gross core protection. The scaling arrangement is such that the trip setting is less than a factor of ten above the indicated level. Analysis of the worst-case accident results in rod block action before MCPR approaches the MCPR fuel cladding integrity Safety Limit.

A downscale indication on an APRM is an indication that the instrument has failed or is not sensitive enough. In either case, the instrument will not respond to changes in control rod motion, and the control rod motion is thus prevented.

The SRM rod blocks of low count rate and the detector not fully inserted assure that the SRMs are not withdrawn from the core prior to commencing rod withdrawal for startup. The scram discharge volume, high water level rod block provides annunciation for operator action. The alarm

BASES

(SRMs), will result in a rod block. The minimum instrument CHANNEL requirements assure sufficient instrumentation to assure that the single failure criterion is met. The minimum instrument CHANNEL requirements for the rod block monitor may be reduced by one for a short period of time to allow for maintenance, testing, or calibration.

The APRM rod block function is flow-biased and prevents a significant reduction in MCPR, especially during operation at reduced flow. The APRM provides gross core protection, i.e., limits the gross withdrawal of control rods in the normal withdrawal sequence.

In the REFUEL MODE during SHUTDOWN MARGIN demonstrations and the STARTUP/HOT STANDBY OPERATIONAL MODE, the APRM rod block function setpoint is significantly reduced to provide the same type of protection in the REFUEL and STARTUP/HOT STANDBY OPERATIONAL MODE(s) as the APRM flow-biased rod block does in the RUN OPERATIONAL MODE, i.e., prevents control rod withdrawal before a scram is reached.

The rod block monitor (RBM) function provides local protection of the core, i.e., the prevention of transition boiling in a local region of the core for a single rod withdrawal error. The trip setting is flow-biased. At low power, the worst-case withdrawal of a single control rod without rod block action will not violate the fuel cladding integrity Safety Limit. Thus the RBM rod block function is not required below the specified power level. The worst-case single control rod withdrawal error is analyzed for each reload to assure that, with the specific trip settings, rod withdrawal is blocked before the MCPR reaches the fuel cladding integrity Safety Limit. RBM "inoperative" actuates on several inputs including: (1) nulling, (2) failure to null, (3) <50% assigned inputs, (4) card pulled, (5) no rod selected, (6) >1 rod selected and (7) switch not in operate.

The IRM rod block function provides local as well as gross core protection. The scaling arrangement is such that the trip setting is less than a factor of ten above the indicated level. Analysis of the worst-case accident results in rod block action before MCPR approaches the MCPR fuel cladding integrity Safety Limit.

A downscale indication on an APRM is an indication that the instrument has failed or is not sensitive enough. In either case, the instrument will not respond to changes in control rod motion, and the control rod motion is thus prevented.

The SRM rod blocks of low count rate and the detector not fully inserted assure that the SRMs are not withdrawn from the core prior to commencing rod withdrawal for startup. The scram discharge volume, high water level rod block provides annunciation for operator action. The alarm setpoint has been selected to provide adequate time to allow for the determination of the cause for the level increase and corrective action prior to automatic scram initiation.

3/4.2.F Accident Monitoring Instrumentation

Instrumentation is provided to monitor sufficient accident conditions to adequately assess important variables and provide operators with necessary information to complete the appropriate

BASES

setpoint has been selected to provide adequate time to allow for the determination of the cause for the level increase and corrective action prior to automatic scram initiation.

3/4.2.F Accident Monitoring Instrumentation

Instrumentation is provided to monitor sufficient accident conditions to adequately assess important variables and provide the operators with the necessary information to complete the appropriate mitigation actions. OPERABILITY of the instrumentation listed provides adequate monitoring of the containment following a loss-of-coolant accident. Information from this instrumentation will provide the operator with a detailed knowledge of the conditions resulting from the accident; based on this information, the operator can make logical decisions regarding post accident recovery. Allowable outage times are based on diverse instrumentation availability for guiding the operator should an accident occur, and on the low probability of an instrument being out-of-service concurrent with an accident. This instrumentation is identified in response to Generic Letter 82-33 and the associated NRC Safety Evaluation Report, and some instrumentation is included in accordance with the response to Generic Letter 83-36.

"INSERT"

3/4.2.G Source Range Monitoring Instrumentation

The source range monitors (SRM) provide the operator with the status of the neutron flux in the core at very low power levels during startup and shutdown. The consequences of reactivity accidents are functions of the initial neutron flux. Therefore, the requirements for a minimum count rate assures that any transient, should it occur, begins at or above the initial value used in the analyses of transients from cold conditions. Two OPERABLE SRM CHANNEL(s) are adequate to monitor the approach to criticality using homogeneous patterns of scattered control rod withdrawal. Three OPERABLE SRMs provide an added conservatism. When the intermediate range monitors are on scale, adequate information is available without the SRMs and they can be retracted.

3/4.2.H Explosive Gas Monitoring Instrumentation

Instrumentation is provided to monitor the concentrations of potentially explosive mixtures in the off-gas (waste) holdup system to prevent a possible uncontrolled release via this pathway. This instrumentation is included in accordance with Generic Letter 89-01.

3/4.2.I Suppression Chamber and Drywell Spray Actuation Instrumentation

Instrumentation is provided to monitor the parameters which are necessary to permit initiation of the suppression chamber and drywell spray mode of the low pressure coolant injection/ containment cooling system to condense steam in the containment atmosphere. The spray mode does not significantly affect the rise of drywell pressure following a loss of coolant accident, but does result in quicker depressurization following completion of the blowdown.

BASES

mitigation actions. OPERABILITY of the instrumentation listed provides adequate monitoring of the containment following a loss-of-coolant accident. Information from this instrumentation will provide the operator with a detailed knowledge of the conditions resulting from the accident; based on this information, the operator can make logical decisions regarding post accident recovery. Allowable outage times are based on diverse instrumentation availability for guiding the operator should an accident occur, and on the low probability of an instrument being out-of-service concurrent with an accident. ~~This instrumentation is identified in response to Generic Letter 82-33 and the associated NRC Safety Evaluation Report, and some instrumentation is included in accordance with the response to Generic Letter 83-36.~~ 'INSERT'

3/4.2.G Source Range Monitoring Instrumentation

The source range monitors (SRM) provide the operator with the status of the neutron flux in the core at very low power levels during startup and shutdown. The consequences of reactivity accidents are functions of the initial neutron flux. Therefore, the requirements for a minimum count rate assures that any transient, should it occur, begins at or above the initial value used in the analyses of transients from cold conditions. Two OPERABLE SRM CHANNEL(s) are adequate to monitor the approach to criticality using homogeneous patterns of scattered control rod withdrawal. Three OPERABLE SRMs provide an added conservatism. When the intermediate range monitors are on scale, adequate information is available without the SRMs and they can be retracted.

3/4.2.H Explosive Gas Monitoring Instrumentation

Instrumentation is provided to monitor the concentrations of potentially explosive mixtures in the off-gas holdup system to prevent a possible uncontrolled release via this pathway. This instrumentation is included in accordance with Generic Letter 89-01.

3/4.2.I Suppression Chamber and Drywell Spray Actuation Instrumentation

Instrumentation is provided to monitor the parameters which are necessary to permit initiation of the containment cooling mode of the residual heat removal system to condense steam in the containment atmosphere. The spray mode does not significantly affect the rise of drywell pressure following a loss of coolant accident, but does result in quicker depressurization following completion of the blowdown.

INSERT

As noted in the surveillance requirements, the instrumentation CHANNEL(s) associated with Torus Pressure provides a dual function and is shared in common with the Drywell Pressure (Narrow and Wide Ranges) instrumentation.

3.3 - LIMITING CONDITIONS FOR OPERATION

A. SHUTDOWN MARGIN (SDM)

The SHUTDOWN MARGIN (SDM) shall be equal to or greater than:

38

1. 0.~~35~~% Δk/k with the highest worth control rod analytically determined, or

28

2. 0.~~25~~% Δk/k with the highest worth control rod determined by test.

APPLICABILITY:

OPERATIONAL MODE(s) 1, 2, 3, 4, and 5.

ACTION:

With the SHUTDOWN MARGIN less than specified:

1. In OPERATIONAL MODE 1 or 2, restore the required SHUTDOWN MARGIN within 6 hours or be in at least HOT SHUTDOWN within the next 12 hours.
2. In OPERATIONAL MODE 3 or 4, immediately verify all insertable control rods to be fully inserted and suspend all activities that could reduce the SHUTDOWN MARGIN. In OPERATIONAL MODE 4, establish SECONDARY CONTAINMENT INTEGRITY within 8 hours.
3. In OPERATIONAL MODE 5, suspend CORE ALTERATION(s) and other activities that could reduce the SHUTDOWN MARGIN and fully insert all insertable control rods within 1 hour. Establish SECONDARY CONTAINMENT INTEGRITY within 8 hours.

4.3 - SURVEILLANCE REQUIREMENTS

A. SHUTDOWN MARGIN

The SHUTDOWN MARGIN shall be determined to be equal to or greater than that specified at any time during the operating cycle:

1. By demonstration, prior to or during the first startup after each refueling outage.
2. Within 24 hours after detection of a withdrawn control rod that is immovable, as a result of excessive friction or mechanical interference, or known to be unscrammable. The required SHUTDOWN MARGIN shall be verified acceptable with an increased allowance for the withdrawn worth of the immovable or unscrammable control rod.
3. By calculation, prior to each fuel movement during the fuel loading sequence.

3.3 - LIMITING CONDITIONS FOR OPERATION

A. SHUTDOWN MARGIN (SDM)

The SHUTDOWN MARGIN (SDM) shall be equal to or greater than:

38

1. 0.38% Δk/k with the highest worth control rod analytically determined, or

28

2. 0.28% Δk/k with the highest worth control rod determined by test.

APPLICABILITY:

OPERATIONAL MODE(s) 1, 2, 3, 4, and 5.

ACTION:

With the SHUTDOWN MARGIN less than specified:

1. In OPERATIONAL MODE 1 or 2, restore the required SHUTDOWN MARGIN within 6 hours or be in at least HOT SHUTDOWN within the next 12 hours.

2. In OPERATIONAL MODE 3 or 4, immediately verify all insertable control rods to be fully inserted and suspend all activities that could reduce the SHUTDOWN MARGIN. In OPERATIONAL MODE 4, establish SECONDARY CONTAINMENT INTEGRITY within 8 hours.

3. In OPERATIONAL MODE 5, suspend CORE ALTERATION(s) and other activities that could reduce the SHUTDOWN MARGIN and fully insert all insertable control rods within 1 hour. Establish SECONDARY CONTAINMENT INTEGRITY within 8 hours.

4.3 - SURVEILLANCE REQUIREMENTS

A. SHUTDOWN MARGIN

The SHUTDOWN MARGIN shall be determined to be equal to or greater than that specified at any time during the operating cycle:

1. By demonstration, prior to or during the first startup after each refueling outage.

2. Within 24 hours after detection of a withdrawn control rod that is immovable, as a result of excessive friction or mechanical interference, or known to be unscrammable. The required SHUTDOWN MARGIN shall be verified acceptable with an increased allowance for the withdrawn worth of the immovable or unscrammable control rod.

3. By calculation, prior to each fuel movement during the fuel loading sequence.

3.3 - LIMITING CONDITIONS FOR OPERATION

C. Control Rod OPERABILITY

All control rods shall be OPERABLE.

APPLICABILITY:

OPERATIONAL MODE(s) 1 and 2.

ACTION:

1. With one control rod inoperable due to being immovable as a result of excessive friction or mechanical interference, or known to be unscrammable:

   a. Within one hour:

      1) Verify that the inoperable control rod, if withdrawn, is separated from all other inoperable withdrawn control rods by at least two control cells in all directions.

      2) Disarm the associated directional control valves[a] either:

         a) Electrically, or

         b) Hydraulically by closing the drive water and exhaust water isolation valves.

   b. With the provisions of ACTION 1.a above not met, be in at least HOT SHUTDOWN within the next 12 hours.

4.3 - SURVEILLANCE REQUIREMENTS

C. Control Rod OPERABILITY

1. When above the low power setpoint of the RWM, all withdrawn control rods not required to have their directional control valves disarmed electrically or hydraulically shall be demonstrated OPERABLE by moving each control rod at least one notch:

   a. At least once per 7 days, and

   b. Within ~~At least once per~~ 24 hours when any control rod is immovable as a result of excessive friction or mechanical interference, or known to be unscrammable.

2. All control rods shall be demonstrated OPERABLE by performance of Surveillance Requirements 4.3.D, 4.3.F, 4.3.G, 4.3.H and 4.3.I.

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a May be rearmed intermittently, under administrative control, to permit testing associated with restoring the control rod to OPERABLE status.

3.3 - LIMITING CONDITIONS FOR OPERATION

C. Control Rod OPERABILITY

All control rods shall be OPERABLE.

APPLICABILITY:

OPERATIONAL MODE(s) 1 and 2.

ACTION:

1. With one control rod inoperable due to being immovable as a result of excessive friction or mechanical interference, or known to be unscrammable:

   a. Within one hour:

      1) Verify that the inoperable control rod, if withdrawn, is separated from all other inoperable withdrawn control rods by at least two control cells in all directions.

      2) Disarm the associated directional control valves[a] either:

         a) Electrically, or

         b) Hydraulically by closing the drive water and exhaust water isolation valves.

   b. With the provisions of ACTION 1.a above not met, be in at least HOT SHUTDOWN within the next 12 hours.

4.3 - SURVEILLANCE REQUIREMENTS

C. Control Rod OPERABILITY

1. When above the low power setpoint of the RWM, all withdrawn control rods not required to have their directional control valves disarmed electrically or hydraulically shall be demonstrated OPERABLE by moving each control rod at least one notch:

   a. At least once per 7 days, and

   b. Within ~~At least once per~~ 24 hours when any control rod is immovable as a result of excessive friction or mechanical interference, or known to be unscrammable.

2. All control rods shall be demonstrated OPERABLE by performance of Surveillance Requirements 4.3.D, 4.3.F, 4.3.G, 4.3.H and 4.3.I.

a May be rearmed intermittently, under administrative control, to permit testing associated with restoring the control rod to OPERABLE status.

3.3 - LIMITING CONDITIONS FOR OPERATION

D. Maximum Scram Insertion Times

The maximum scram insertion time of each control rod from the fully withdrawn position to 90% insertion, based on de-energization of the scram pilot valve solenoids as time zero, shall not exceed 7 seconds.

APPLICABILITY:

OPERATIONAL MODE(s) 1 and 2.

ACTION:

With the maximum scram insertion time of one or more control rods exceeding 7 seconds:

1. Declare the control rod(s) exceeding the above maximum scram insertion time inoperable, and
2. When operation is continued with three or more control rods with maximum scram insertion times in excess of 7 seconds, perform Surveillance Requirement 4.3.D.3 at least once per 60 days of POWER OPERATION. No caps

With the provisions of the ACTION(s) above not met, be in at least HOT SHUTDOWN within 12 hours.

4.3 - SURVEILLANCE REQUIREMENTS

D. Maximum Scram Insertion Times

The maximum scram insertion time of the control rods shall be demonstrated through measurement with reactor coolant pressure greater than 800 psig and, during single control rod scram time tests, with the control rod drive pumps isolated from the accumulators:

1. For all control rods prior to THERMAL POWER exceeding 40% of RATED THERMAL POWER:
   a. following CORE ALTERATION(s), or
   b. after a reactor shutdown that is greater than 120 days,
2. For specifically affected individual control rods[a] following maintenance on or modification to the control rod or control rod drive system which could affect the scram insertion time of those specific control rods, and
3. For at least 10% of the control rods, on a rotating basis, at least once per 120 days of POWER OPERATION. No caps

a The provisions of Specification 4.0.D are not applicable provided this surveillance is conducted prior to exceeding 40% of RATED THERMAL POWER.

3.3 - LIMITING CONDITIONS FOR OPERATION

D. Maximum Scram Insertion Times

The maximum scram insertion time of each control rod from the fully withdrawn position to 90% insertion, based on de-energization of the scram pilot valve solenoids as time zero, shall not exceed 7 seconds.

APPLICABILITY:

OPERATIONAL MODE(s) 1 and 2.

ACTION:

With the maximum scram insertion time of one or more control rods exceeding 7 seconds:

1. Declare the control rod(s) exceeding the above maximum scram insertion time inoperable, and

2. When operation is continued with three or more control rods with maximum scram insertion times in excess of 7 seconds, perform Surveillance Requirement 4.3.D.3 at least once per 60 days of POWER OPERATION. No CAPS

With the provisions of the ACTION above not met, be in at least HOT SHUTDOWN within 12 hours.

4.3 - SURVEILLANCE REQUIREMENTS

D. Maximum Scram Insertion Times

The maximum scram insertion time of the control rods shall be demonstrated through measurement with reactor coolant pressure greater than 800 psig and, during single control rod scram time tests, with the control rod drive pumps isolated from the accumulators:

1. For all control rods prior to THERMAL POWER exceeding 40% of RATED THERMAL POWER:

   a. following CORE ALTERATION(s), or

   b. after a reactor shutdown that is greater than 120 days,

2. For specifically affected individual control rods[a] following maintenance on or modification to the control rod or control rod drive system which could affect the scram insertion time of those specific control rods, and

3. For at least 10% of the control rods, on a rotating basis, at least once per 120 days of POWER OPERATION. No CAPS

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a The provisions of Specification 4.0.D are not applicable provided this surveillance is conducted prior to exceeding 40% of RATED THERMAL POWER.

3.3 - LIMITING CONDITIONS FOR OPERATION

H. Control Rod Drive Coupling

All control rods shall be coupled to their drive mechanisms.

APPLICABILITY:

OPERATIONAL MODE(s) 1, 2, and 5[a].

ACTION:

1. In OPERATIONAL MODE 1 or 2 with one control rod not coupled to its associated drive mechanism, within 2 hours:

   a. If permitted by the RWM, insert the control rod drive mechanism to accomplish recoupling and verify recoupling by withdrawing the control rod, and:

      1) Observing any indicated response of the nuclear instrumentation, and

      2) Demonstrating that the control rod will not go to the overtravel position.

   b. If not permitted by the RWM or, if recoupling is not accomplished in accordance with ACTION 1.a above, then declare the control rod inoperable, fully insert the control rod and disarm the associated directional control valves[b] either:

      1) Electrically, or

4.3 - SURVEILLANCE REQUIREMENTS

H. Control Rod Drive Coupling

Each affected control rod shall be demonstrated to be coupled to its drive mechanism by observing any indicated response of the nuclear instrumentation while withdrawing the control rod to the fully withdrawn position and then verifying that the control rod drive does not go to the overtravel position:

'Deleted'

1. Prior to reactor criticality after completing CORE ALTERATION(s) that could have affected the control rod drive coupling integrity,

2. Anytime the control rod is withdrawn to the "Full out" position ~~in subsequent operation~~, and

3. Following maintenance on or modification to the control rod or control rod drive system which could have affected the control rod drive coupling integrity.

---

a In OPERATIONAL MODE 5, this Specification is applicable for withdrawn control rods and is not applicable to control rods removed per Specification 3.10.I or 3.10.J.

b May be rearmed intermittently, under administrative control, to permit testing associated with restoring the control rod to OPERABLE status.

3.3 - LIMITING CONDITIONS FOR OPERATION

H. Control Rod Drive Coupling

All control rods shall be coupled to their drive mechanisms.

APPLICABILITY:

OPERATIONAL MODE(s) 1, 2, and 5[a].

ACTION:

1. In OPERATIONAL MODE 1 or 2 with one control rod not coupled to its associated drive mechanism, within 2 hours:
   a. If permitted by the RWM, insert the control rod drive mechanism to accomplish recoupling and verify recoupling by withdrawing the control rod, and:
      1) Observing any indicated response of the nuclear instrumentation, and
      2) Demonstrating that the control rod will not go to the overtravel position.
   b. If not permitted by the RWM or, if recoupling is not accomplished in accordance with ACTION 1.a above, then declare the control rod inoperable, fully insert the control rod and disarm the associated directional control valves[b] either:
      1) Electrically, or

4.3 - SURVEILLANCE REQUIREMENTS

H. Control Rod Drive Coupling

Each affected control rod shall be demonstrated to be coupled to its drive mechanism by observing any indicated response of the nuclear instrumentation while withdrawing the control rod to the fully withdrawn position and then verifying that the control rod drive does not go to the overtravel position:

'Deleted'

1. Prior to reactor criticality after completing CORE ALTERATION(s) that could have affected the control rod drive coupling integrity,
2. Anytime the control rod is withdrawn to the "Full out" position ~~in subsequent operation~~, and
3. Following maintenance on or modification to the control rod or control rod drive system which could have affected the control rod drive coupling integrity.

a In OPERATIONAL MODE 5, this Specification is applicable for withdrawn control rods and is not applicable to control rods removed per Specification 3.10.i or 3.10.J.

b May be rearmed intermittently, under administrative control, to permit testing associated with restoring the control rod to OPERABLE status.

3.3 - LIMITING CONDITIONS FOR OPERATION

I. Control Rod Position Indication System

All control rod position indicators shall be OPERABLE.

APPLICABILITY:

OPERATIONAL MODE(s) 1, 2, and 5[a].

ACTION:

1. In OPERATIONAL MODE 1 or 2 with one or more control rod position indicators inoperable, within one hour either:
   a. Determine the position of the control rod by an alternate method, or
   b. Move the control rod to a position with an OPERABLE position indicator, or
   c. Declare the control rod inoperable, fully insert the inoperable withdrawn control rod(s), and disarm the associated directional control valves[b] either:
      1) Electrically, or
      2) Hydraulically by closing the drive water and exhaust water isolation valves.

4.3 - SURVEILLANCE REQUIREMENTS

I. Control Rod Position Indication System

The control rod position indication system shall be determined OPERABLE by verifying:

1. At least once per 24 hours that the position of each control rod is indicated.
2. That the indicated control rod position changes during the movement of the control rod drive when performing Surveillance Requirement 4.3.C.1.
3. That the control rod position indicator corresponds to the control rod position indicated by the "Full out" position indicator when performing Surveillance Requirement 4.3.H.2.

'Deleted'

a In OPERATIONAL MODE 5, this Specification is applicable for withdrawn control rods and is not applicable to control rods removed per Specification 3.10.I or 3.10.J.

b May be rearmed intermittently, under administrative control, to permit testing associated with restoring the control rod(s) to OPERABLE status.

3.3 - LIMITING CONDITIONS FOR OPERATION

I. Control Rod Position Indication System

All control rod position indicators shall be OPERABLE.

APPLICABILITY:

OPERATIONAL MODE(s) 1, 2, and 5[a].

ACTION:

1. In OPERATIONAL MODE 1 or 2 with one or more control rod position indicators inoperable, within one hour either:

   a. Determine the position of the control rod by an alternate method, or

   b. Move the control rod to a position with an OPERABLE position indicator, or

   c. Declare the control rod inoperable, fully insert the inoperable withdrawn control rod(s), and disarm the associated directional control valves[b] either:

      1) Electrically, or

      2) Hydraulically by closing the drive water and exhaust water isolation valves.

4.3 - SURVEILLANCE REQUIREMENTS

I. Control Rod Position Indication System

The control rod position indication system shall be determined OPERABLE by verifying:

1. At least once per 24 hours that the position of each control rod is indicated.

2. That the indicated control rod position changes during the movement of the control rod drive when performing Surveillance Requirement 4.3.C.1.

3. That the control rod position indicator corresponds to the control rod position indicated by the "Full out" position indicator when performing Surveillance Requirement 4.3.H.2.

'Deleted'

---

a In OPERATIONAL MODE 5, this Specification is applicable for withdrawn control rods and is not applicable to control rods removed per Specification 3.10.I or 3.10.J.

b May be rearmed intermittently, under administrative control, to permit testing associated with restoring the control rod(s) to OPERABLE status.

3.3 - LIMITING CONDITIONS FOR OPERATION

L. Rod Worth Minimizer (RWM)

The rod worth minimizer (RWM) shall be OPERABLE.

APPLICABILITY:

OPERATIONAL MODE(s) 1 and 2[a], when THERMAL POWER is less than or equal to 20% of RATED THERMAL POWER.

ACTION:

With the RWM inoperable, verify control rod movement and compliance with the prescribed control rod pattern by a second licensed operator or technically qualified individual who is present at the reactor control console. Otherwise, control rod movement may be made only by actuating the manual scram or placing the reactor mode switch in the Shutdown position.

4.3 - SURVEILLANCE REQUIREMENTS

L. Rod Worth Minimizer (RWM)

The RWM shall be demonstrated OPERABLE:

1. By verifying that the control rod patterns and sequence input to the RWM computer are correctly loaded following any loading of the program into the computer.

2. In OPERATIONAL MODE 2 within 8 hours prior to withdrawal of control rods for the purpose of making the reactor critical:

   a. by verifying proper indication of the selection error of at least one out-of-sequence control rod.

   b. by verifying the rod block function by demonstrating inability to withdraw an out-of-sequence control rod.

3. In OPERATIONAL MODE 1 prior to reducing THERMAL POWER below 20% of RATED THERMAL POWER:

   a. by verifying proper indication of the selection error of at least one out-of-sequence control rod.

   b. by verifying the rod block function by demonstrating inability to withdraw an out-of-sequence control rod.

---

a Entry into OPERATIONAL MODE 2 and withdrawal of selected control rods is permitted for the purpose of determining the OPERABILITY of the RWM prior to withdrawal of control rods for the purpose of bringing the reactor to criticality.

3.3 - LIMITING CONDITIONS FOR OPERATION

L. Rod Worth Minimizer (RWM)

The rod worth minimizer (RWM) shall be OPERABLE.

APPLICABILITY:

OPERATIONAL MODE(s) 1 and 2[a], when thermal power is less than or equal to 10% of RATED THERMAL POWER.

ACTION:

With the RWM inoperable, verify control rod movement and compliance with the prescribed control rod pattern by a second licensed operator or technically qualified individual who is present at the reactor control console. Otherwise, control rod movement may be made only by actuating the manual scram or placing the reactor mode switch in the Shutdown position.

4.3 - SURVEILLANCE REQUIREMENTS

L. Rod Worth Minimizer (RWM)

The RWM shall be demonstrated OPERABLE:

1. By verifying that the control rod patterns and sequence input to the RWM computer are correctly loaded following any loading of the program into the computer.

2. In OPERATIONAL MODE 2 within 8 hours prior to withdrawal of control rods for the purpose of making the reactor critical:

   a. by verifying proper indication of the selection error of at least one out-of-sequence control rod.

   b. by verifying the rod block function by demonstrating inability to withdraw an out-of-sequence control rod.

3. In OPERATIONAL MODE 1 prior to reducing thermal power below 10% of RATED THERMAL POWER:

   a. by verifying proper indication of the selection error of at least one out-of-sequence control rod.

   b. by verifying the rod block function by demonstrating inability to withdraw an out-of-sequence control rod.

a Entry into OPERATIONAL MODE 2 and withdrawal of selected control rods is permitted for the purpose of determining the OPERABILITY of the RWM prior to withdrawal of control rods for the purpose of bringing the reactor to criticality.

BASES

3/4.3.A SHUTDOWN MARGIN

A sufficient SHUTDOWN MARGIN ensures that 1) the reactor can be made subcritical from all operating conditions, 2) the reactivity transients associated with postulated accident conditions are controllable within acceptable limits, and 3) the reactor will be maintained sufficiently subcritical to preclude inadvertent criticality in the shutdown condition.

The SHUTDOWN MARGIN limitation is a restriction to be applied principally to a new refueling pattern. Satisfaction of the limitation must be determined at the time of loading and must be such that it will apply to the entire subsequent fuel cycle. This determination is provided by core design calculations and administrative control of fuel loading patterns. These procedures include restrictions to allow only those intermediate fuel assembly configurations that have been shown to provide the required SHUTDOWN MARGIN.

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Since core reactivity values will vary through core life as a function of fuel depletion and poison burnup, the demonstration of SHUTDOWN MARGIN will be performed during xenon free conditions and adjusted to 68°F to accommodate the current moderator temperature. The generalized form is that the reactivity of the core loading will be limited so the core can be made subcritical by at least R + 0.35% Δk/k or R + 0.25% Δk/k, as appropriate, with the strongest control rod fully withdrawn and all others fully inserted. Two different values are supplied in the Limiting Condition for Operation to provide for the different methods of determination of the highest control rod worth, either analytically or by test. This is due to the reduced uncertainty in the SHUTDOWN MARGIN test when the highest worth control rod is determined by demonstration. When SHUTDOWN MARGIN is determined by calculations not associated with a test, additional margin must be added to the specified SHUTDOWN MARGIN limit to account for uncertainties in the calculation.

The value of R in units of % Δk/k is the difference between the calculated beginning-of-life core reactivity, at the beginning of the operating cycle, and the calculated value of maximum core reactivity at any time later in the operating cycle, where it would be greater than at the beginning. The value of R shall include the potential SHUTDOWN MARGIN loss assuming full B_4C settling in all inverted poison tubes present in the core. R must be a positive quantity or zero and a new value of R must be determined for each new fuel cycle.

(or 0.38 % ΔK)

28

The value of % Δk/k in the above expression is provided as a finite, demonstrable, subcriticality margin. This margin is verified using an in-sequence control rod withdrawal at the beginning-of-life fuel cycle conditions. This assures subcriticality with not only the strongest fully withdrawn but at least an R + 0.25% Δk margin beyond this condition. This reactivity characteristic has been a basic assumption in the analysis of plant performance and can be best demonstrated at the time of fuel loading, but the margin must also be determined anytime a control rod is incapable of insertion following a scram signal. Any control rod that is immovable as a result of excessive friction or mechanical interference, or is known to be unscrammable, per Specification 3.3.C, is considered to be incapable of insertion following a scram signal. It is important to note that a control rod can be electrically immovable, but scrammable, and no increase in SHUTDOWN MARGIN is required for these control rods.

'INSERT'

INSERT
[SDM Bases]

During MODE 5, adequate SDM is required to ensure that the reactor does not reach criticality during control rod withdrawals. An evaluation of each in-vessel fuel movement during fuel loading (including shuffling fuel within the core) is required to ensure adequate SDM is maintained during refueling. This evaluation ensures that the intermediate loading patterns are bounded by the safety analyses for the final core loading pattern. For example, bounding analyses that demonstrate adequate SDM for the most reactive configurations during the refueling may be performed to demonstrate acceptability of the entire fuel movement sequence. These bounding analyses include additional margins to the associated uncertainties. Spiral offload/reload sequences inherently satisfy the SR, provided the fuel assemblies are reloaded in the same configuration analyzed for the new cycle. Removing fuel from the core will always result in an increase in SDM.

BASES

3/4.3.A SHUTDOWN MARGIN

A sufficient SHUTDOWN MARGIN ensures that 1) the reactor can be made subcritical from all operating conditions, 2) the reactivity transients associated with postulated accident conditions are controllable within acceptable limits, and 3) the reactor will be maintained sufficiently subcritical to preclude inadvertent criticality in the shutdown condition.

The SHUTDOWN MARGIN limitation is a restriction to be applied principally to a new refueling pattern. Satisfaction of the limitation must be determined at the time of loading and must be such that it will apply to the entire subsequent fuel cycle. This determination is provided by core design calculations and administrative control of fuel loading patterns. These procedures include restrictions to allow only those intermediate fuel assembly configurations that have been shown to provide the required SHUTDOWN MARGIN.

Since core reactivity values will vary through core life as a function of fuel depletion and poison burnup, the demonstration of SHUTDOWN MARGIN will be performed during xenon free conditions and adjusted to 68°F to accommodate the current moderator temperature. The generalized form is that the reactivity of the core loading will be limited so the core can be made subcritical by at least R + 0.35% Δk/k or R + 0.25% Δk/k, as appropriate, with the strongest control rod fully withdrawn and all others fully inserted. Two different values are supplied in the Limiting Condition for Operation to provide for the different methods of determination of the highest control rod worth, either analytically or by test. This is due to the reduced uncertainty in the SHUTDOWN MARGIN test when the highest worth control rod is determined by demonstration. When SHUTDOWN MARGIN is determined by calculations not associated with a test, additional margin must be added to the specified SHUTDOWN MARGIN limit to account for uncertainties in the calculation.

28

38

The value of R in units of % Δk/k is the difference between the calculated beginning-of-life core reactivity, at the beginning of the operating cycle, and the calculated value of maximum core reactivity at any time later in the operating cycle, where it would be greater than at the beginning. The value of R shall include the potential SHUTDOWN MARGIN loss assuming full B_4C settling in all inverted poison tubes present in the core. R must be a positive quantity or zero and a new value of R must be determined for each new fuel cycle.

(or 0.38% Δk)

The value of % Δk/k in the above expression is provided as a finite, demonstrable, subcriticality margin. This margin is verified using an in-sequence control rod withdrawal at the beginning-of-life fuel cycle conditions. This assures subcriticality with not only the strongest fully withdrawn but at least an R + 0.25% Δk margin beyond this condition. This reactivity characteristic has been a basic assumption in the analysis of plant performance and can be best demonstrated at the time of fuel loading, but the margin must also be determined anytime a control rod is incapable of insertion following a scram signal. Any control rod that is immovable as a result of excessive friction or mechanical interference, or is known to be unscrammable, per Specification 3.3.C, is considered to be incapable of insertion following a scram signal. It is important to note that a control rod can be electrically immovable, but scrammable, and no increase in SHUTDOWN MARGIN is required for these control rods.

28

INSERT

INSERT
[SDM Bases]

During MODE 5, adequate SDM is required to ensure that the reactor does not reach criticality during control rod withdrawals. An evaluation of each in-vessel fuel movement during fuel loading (including shuffling fuel within the core) is required to ensure adequate SDM is maintained during refueling. This evaluation ensures that the intermediate loading patterns are bounded by the safety analyses for the final core loading pattern. For example, bounding analyses that demonstrate adequate SDM for the most reactive configurations during the refueling may be performed to demonstrate acceptability of the entire fuel movement sequence. These bounding analyses include additional margins to the associated uncertainties. Spiral offload/reload sequences inherently satisfy the SR, provided the fuel assemblies are reloaded in the same configuration analyzed for the new cycle. Removing fuel from the core will always result in an increase in SDM.

BASES

used to scram the control rods from the nitrogen which provides the required energy. The scram accumulators are necessary to scram the control rods within the required insertion times.

Control rods with inoperable accumulators are declared inoperable and Specification 3.3.C then applies. This prevents a pattern of inoperable accumulators that would result in less reactivity insertion on a scram than has been analyzed even though control rods with inoperable accumulators may still be inserted with normal drive water pressure. OPERABILITY of the accumulator ensures that there is a means available to insert the control rods even under the most unfavorable depressurization of the reactor.

3/4.3.H Control Rod Drive Coupling

Control rod dropout accidents can lead to significant core damage. If coupling integrity is maintained, the possibility of a rod drop accident is eliminated. Neutron instrumentation response to rod movement may provide verification that a rod is following its drive. Absence of such response to drive movement may indicate an uncoupled condition, or may be due to the lack of proximity of the drive to the instrumentation. However, the overtravel position feature provides a positive check, as only uncoupled drives may reach this position.

3/4.3.I Control Rod Position Indication System (RPIS)

In order to ensure that the control rod patterns can be followed and therefore that other parameters are within their limits, the control rod position indication system must be OPERABLE. Normal control rod position is displayed by two-digit indication to the operator from position 00 to 48. Each even number is a latching position, whereas each odd number provides information while the rod is in-motion and inputs for rod drift annunciation. The ACTION statement provides for the condition where no positive information is displayed for a large portion or all of the rod's travel. Usually, only one digit of one or two of a rod's positions is unavailable with a faulty RPIS, and the control rod may be located in a known position. However, there are several alternate methods for determining control rod position including the full core display, the four rod display, the rod worth minimizer, and the process computer. ~~Additionally, there are independent "full-in" and "full-out" indicators at the 00 and 48 positions.~~ Another method to determine position would be to move the control rod, by single notch movement, to a position with an OPERABLE position indicator. The original position would then be established and the control rod could be returned to its original position by single notch movement. As long as no control rod drift alarms are received, the position of the control rod would then be known.

3/4.3.J Control Rod Drive Housing Support

The control rod housing support restricts the outward movement of a control rod to less than 3 inches in the extremely remote event of a housing failure. The amount of reactivity which could be added by this small amount of rod withdrawal, which is less than a normal single withdrawal

BASES

used to scram the control rods from the nitrogen which provides the required energy. The scram accumulators are necessary to scram the control rods within the required insertion times.

Control rods with inoperable accumulators are declared inoperable and Specification 3.3.C then applies. This prevents a pattern of inoperable accumulators that would result in less reactivity insertion on a scram than has been analyzed even though control rods with inoperable accumulators may still be inserted with normal drive water pressure. OPERABILITY of the accumulator ensures that there is a means available to insert the control rods even under the most unfavorable depressurization of the reactor.

3/4.3.H Control Rod Drive Coupling

Control rod dropout accidents can lead to significant core damage. If coupling integrity is maintained, the possibility of a rod drop accident is eliminated. Neutron instrumentation response to rod movement may provide verification that a rod is following its drive. Absence of such response to drive movement may indicate an uncoupled condition, or may be due to the lack of proximity of the drive to the instrumentation. However, the overtravel position feature provides a positive check, as only uncoupled drives may reach this position.

3/4.3.I Control Rod Position Indication System (RPIS)

In order to ensure that the control rod patterns can be followed and therefore that other parameters are within their limits, the control rod position indication system must be OPERABLE. Normal control rod position is displayed by two-digit indication to the operator from position 00 to 48. Each even number is a latching position, whereas each odd number provides information while the rod is in-motion and inputs for rod drift annunciation. The ACTION statement provides for the condition where no positive information is displayed for a large portion or all of the rod's travel. Usually, only one digit of one or two of a rod's positions is unavailable with a faulty RPIS, and the control rod may be located in a known position. However, there are several alternate methods for determining control rod position including the full core display, the four rod display, the rod worth minimizer, and the process computer. ~~Additionally, there are independent "full-in" and "full-out" indicators at the 00 and 48 positions.~~ Another method to determine position would be to move the control rod, by single notch movement, to a position with an OPERABLE position indicator. The original position would then be established and the control rod could be returned to its original position by single notch movement. As long as no control rod drift alarms are received, the position of the control rod would then be known.

3/4.3.J Control Rod Drive Housing Support

The control rod housing support restricts the outward movement of a control rod to less than 3 inches in the extremely remote event of a housing failure. The amount of reactivity which could be added by this small amount of rod withdrawal, which is less than a normal single withdrawal

BASES

four bundle local peaking factor. The NRC approved methodology listed in Specification ~~6.6.A.4~~ 6.9.A.6 provides a detailed description of the methodology used in performing the rod drop analyses.

The rod worth minimizer provides automatic supervision to assure that out-of-sequence control rods will not be withdrawn or inserted, i.e., it limits operator deviations from planned withdrawal sequences (reference UFSAR Section 7.7.2). It serves as a backup to procedural control of control rod worth. In the event that the rod worth minimizer is out-of-service when required, a second licensed operator or other technically qualified individual who is present at the reactor console can manually fulfill the control rod pattern conformance function of the rod worth minimizer. In this case, the normal procedural controls are backed up by independent procedural controls to assure conformance.

3/4.3.M Rod Block Monitor

The rod block monitor (RBM) is designed to automatically prevent fuel damage in the event of erroneous rod withdrawal from locations of high power density during high power operation. Two channels are provided, and one of these may be bypassed from the console for maintenance and/or testing. Tripping of one of the channels will block erroneous rod withdrawal soon enough to prevent fuel damage. This system backs up the operator, who withdraws control rods according to a written sequence. The specified restrictions with one channel out-of-service conservatively assure that fuel damage will not occur due to rod withdrawal errors when this condition exists.

3/4.3.N Economic Generation Control System

Operation of the facility with the economic generation control system (EGC) (automatic flow control) is limited to the range of 65% to 100% of rated core flow. In this flow range and above 20% of RATED THERMAL POWER, the reactor could safely tolerate a rate of change of load of 8 MWe/sec (reference UFSAR Section 7.7.3.2). Limits within the EGC and the flow control system prevent rates of change greater than approximately 4 MWe/sec. When EGC is in operation, this fact will be indicated on the main control room console.

BASES

four bundle local peaking factor. The NRC approved methodology listed in Specification ~~6.6.A.4~~ 6.9.A.6 provides a detailed description of the methodology used in performing the rod drop analyses.

The rod worth minimizer provides automatic supervision to assure that out-of-sequence control rods will not be withdrawn or inserted, i.e., it limits operator deviations from planned withdrawal sequences (reference UFSAR Section 7.7.2). It serves as a backup to procedural control of control rod worth. In the event that the rod worth minimizer is out-of-service when required, a second licensed operator or other technically qualified individual who is present at the reactor console can manually fulfill the control rod pattern conformance function of the rod worth minimizer. In this case, the normal procedural controls are backed up by independent procedural controls to assure conformance.

3/4.3.M Rod Block Monitor

The rod block monitor (RBM) is designed to automatically prevent fuel damage in the event of erroneous rod withdrawal from locations of high power density during high power operation. Two channels are provided, and one of these may be bypassed from the console for maintenance and/or testing. Tripping of one of the channels will block erroneous rod withdrawal soon enough to prevent fuel damage. This system backs up the operator, who withdraws control rods according to a written sequence. The specified restrictions with one channel out-of-service conservatively assure that fuel damage will not occur due to rod withdrawal errors when this condition exists.

3/4.3.N Economic Generation Control System

Operation of the facility with the economic generation control system (EGC) (automatic flow control) is limited to the range of 65% to 100% of rated core flow. In this flow range and above 20% of RATED THERMAL POWER, the reactor could safely tolerate a rate of change of load of 8 MWe/sec (reference UFSAR Section 7.7.3.2). Limits within the EGC and the flow control system prevent rates of change greater than approximately 4 MWe/sec. When EGC is in operation, this fact will be indicated on the main control room console.

3.5 - LIMITING CONDITIONS FOR OPERATION

A. Emergency Core Cooling System - Operating

The emergency core cooling systems (ECCS) shall be OPERABLE with:

1. The core spray (CS) system consisting of two subsystems with each subsystem comprised of:

   a. One OPERABLE CS pump, and

   b. An OPERABLE flow path capable of taking suction from the suppression chamber and transferring the water through the spray sparger to the reactor vessel.

2. The low pressure coolant injection (LPCI) subsystem comprised of:

   a. Four OPERABLE LPCI pumps, and

   b. An OPERABLE flow path capable of taking suction from the suppression chamber and transferring the water to the reactor vessel.

3. The high pressure cooling injection (HPCI) system consisting of:

   a. One OPERABLE HPCI pump, and

   b. An OPERABLE flow path capable of taking suction from the suppression chamber and transferring the water to the reactor vessel.

4. The automatic depressurization system (ADS) with at least 5 OPERABLE ADS valves.

4.5 - SURVEILLANCE REQUIREMENTS

A. Emergency Core Cooling System - Operating

The ECCS shall be demonstrated OPERABLE by:

1. At least once per 31 days:

   a. For the CS system, the LPCI subsystem and the HPCI system:

      1) Verifying by venting at the high point vents that the system piping from the pump discharge valve to the system isolation valve is filled with water.

      2) 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[a] position.

   b. For the HPCI system, verifying that the HPCI pump flow controller is in the correct position.

2. Verifying that, when tested pursuant to Specification 4.0.E:

   a. The CS pump in each subsystem develop a flow of at least 4500 gpm against a test line pressure corresponding to a reactor vessel pressure of ≥90 psig.

---

a Except that an automatic valve capable of automatic return to its ECCS position when an ECCS signal is present may be in position for another mode of operation.

3.5 - LIMITING CONDITIONS FOR OPERATION

A. Emergency Core Cooling System - Operating

The emergency core cooling systems (ECCS) shall be OPERABLE with:

1. The core spray (CS) system consisting of two subsystems with each subsystem comprised of:

   a. One OPERABLE CS pump, and

   b. An OPERABLE flow path capable of taking suction from the suppression chamber and transferring the water through the spray sparger to the reactor vessel.

2. The low pressure coolant injection (LPCI) subsystem comprised of[e]:

   a. Four OPERABLE LPCI pumps, and

   b. An OPERABLE flow path capable of taking suction from the suppression chamber and transferring the water to the reactor vessel.

3. The high pressure cooling injection (HPCI) system consisting of:

   a. One OPERABLE HPCI pump, and

   b. An OPERABLE flow path capable of taking suction from the suppression chamber and transferring the water to the reactor vessel.

4.5 - SURVEILLANCE REQUIREMENTS

A. Emergency Core Cooling System - Operating

The ECCS shall be demonstrated OPERABLE by:

1. At least once per 31 days:

   a. For the CS system, the LPCI subsystem and the HPCI system:

      1) Verifying by venting at the high point vents that the system piping from the pump discharge valve to the system isolation valve is filled with water.

      2) 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[a] position.

   b. For the HPCI system, verifying that the HPCI pump flow controller is in the correct position.

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e The LPCI subsystem may be considered OPERABLE during alignment and operation for decay heat removal when below the actual RHR cut in permissive pressure in MODE 3, if capable of being manually realigned (remote or local) to the LPCI mode and not otherwise inoperable.

a Except that an automatic valve capable of automatic return to its ECCS position when an ECCS signal is present may be in position for another mode of operation.

3.5 - LIMITING CONDITIONS FOR OPERATION

APPLICABILITY:

OPERATIONAL MODE(s) 1, 2(b) and 3(b).

ACTION:

1. For the core spray system:

   a. With one CS subsystem inoperable, provided that the LPCI subsystem is OPERABLE, restore the inoperable CS subsystem to OPERABLE status within 7 days, or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

   b. With both CS subsystems inoperable, be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

2. For the LPCI subsystem:

   a. With one LPCI pump inoperable (d), provided that both CS subsystems are OPERABLE, restore the inoperable LPCI pump to OPERABLE status within 30 days, or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

4.5 - SURVEILLANCE REQUIREMENTS

b. Three LPCI pumps together develop a flow of at least 14,500 gpm against a test line pressure corresponding to a reactor vessel pressure of ≥20 psig.

c. The HPCI pump develops a flow of at least 5000 gpm against a ~~test line pressure~~ system head corresponding to ~~a~~ reactor vessel pressure, ~~of ≥1150 psig~~ when steam is being supplied to the turbine between 920 and 1005 psig(c).

3. At least once per 18 months:

   a. For the CS system, the LPCI subsystem, and the HPCI system, performing a system functional test which includes simulated automatic actuation of the system throughout its emergency operating sequence and verifying that each automatic valve in the flow path actuates to its correct position. Actual injection of coolant into the reactor vessel may be excluded from this test. 'INSERT'

   b. For the HPCI system, verifying that:

      1) The system develops a flow of ≥5000 gpm against a ~~test line pressure~~ system head corresponding to ~~a~~ reactor vessel pressure ~~of ≥300 psig~~ when steam is being supplied to the turbine between ~~200~~ 150 and 350 psig(c).

b The HPCI system and ADS are not required to be OPERABLE when reactor steam dome pressure is ≤150 psig.

c The provisions of Specification 4.0.D are not applicable provided the surveillance is performed within 12 hours after reactor steam pressure is adequate to perform the test.

d 'INSERT'

INSERT
[ECCS SR]

For the CS system, the LPCI subsystem, and the HPCI system, verify each system/subsystem actuates on an actual or simulated automatic initiation signal. Actual injection of coolant into the reactor vessel may be excluded from this test.

INSERT

The provisions of Specification 3.9.A, Actions 4.a or 6.b are applicable to the LPCI subsystem such that with an inoperable diesel generator, for the remaining OPERABLE diesel generator, both LPCI pumps (and their associated flow path) associated with that OPERABLE diesel generator, shall be OPERABLE.

3.5 - LIMITING CONDITIONS FOR OPERATION

4. The automatic depressurization system (ADS) with at least 5 OPERABLE ADS valves.

APPLICABILITY:

OPERATIONAL MODE(s) 1, $2^{(b)}$ and $3^{(b)}$.

ACTION:

1. For the core spray system:

   a. With one CS subsystem inoperable, provided that the LPCI subsystem is OPERABLE, restore the inoperable CS subsystem to OPERABLE status within 7 days, or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

   b. With both CS subsystems inoperable, be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

2. For the LPCI subsystem:

   (f) 'INSERT'

   a. With one LPCI pump inoperable, provided that both CS subsystems are OPERABLE, restore the inoperable LPCI pump to OPERABLE status within 30 days, or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

4.5 - SURVEILLANCE REQUIREMENTS

2. Verifying that, when tested pursuant to Specification 4.0.E:

   a. The CS pump in each subsystem develop a flow of at least 4500 gpm against a test line pressure corresponding to a reactor vessel pressure of ≥90 psig.

   b. Two LPCI pumps together develop a flow of at least 9,000 gpm against a test line pressure corresponding to a reactor vessel pressure of ≥20 psig.

   c. The HPCI pump develops a flow of at least 5000 gpm against a ~~test line pressure~~ system head corresponding to ~~a~~ reactor vessel pressure ~~of~~ ~~≥1150 psig~~ when steam is being supplied to the turbine between 920 and 1005 psig[c].

3. At least once per 18 months:

   a. For the CS system, the LPCI subsystem, and the HPCI system, performing a system functional test which includes simulated automatic actuation of the system throughout its emergency operating sequence and verifying that each automatic valve in the flow path actuates to its correct position. Actual injection of coolant into the reactor vessel may be excluded from this test.

---

b The HPCI system and ADS are not required to be OPERABLE when reactor steam dome pressure is ≤150 psig.

c The provisions of Specification 4.0.D are not applicable provided the surveillance is performed within 12 hours after reactor steam pressure is adequate to perform the test.

(f) 'INSERT'

<u>INSERT</u>
[ECCS SR]

For the CS system, the LPCI subsystem, and the HPCI system, verify each system/subsystem actuates on an actual or simulated automatic initiation signal. Actual injection of coolant into the reactor vessel may be excluded from this test.

INSERT

The provisions of Specification 3.9.A, Actions 4.a or 6.b are applicable to the LPCI subsystem such that with an inoperable diesel generator, for the remaining OPERABLE diesel generator, both LPCI pumps (and their associated flow path) associated with that OPERABLE diesel generator, shall be OPERABLE.

3.5 - LIMITING CONDITIONS FOR OPERATION

b. With the LPCI subsystem otherwise inoperable, ~~provided that both CS~~ subsystems are OPERABLE, restore the LPCI subsystem to OPERABLE status within 7 days or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours. (d)

c. With the LPCI subsystem and one or both CS subsystems inoperable, be in at least HOT SHUTDOWN within 12 hours and in COLD SHUTDOWN within the next 24 hours.

3. With the HPCI system inoperable, provided both CS subsystems, the LPCI subsystem, the ADS and the Isolation Condenser (IC) system are OPERABLE, restore the HPCI system to OPERABLE status within 14 days or be in at least HOT SHUTDOWN within the next 12 hours and reduce reactor steam dome pressure to ≤150 psig within the following 24 hours.

4. For the ADS:

a. With one of the above required ADS valves inoperable, provided the HPCI system, both CS subsystems and ~~the~~ three LPCI ~~subsystem~~ pumps are OPERABLE, restore the inoperable ADS valve to OPERABLE status within 14 days or be in at least HOT SHUTDOWN within the next 12 hours and reduce reactor steam dome pressure to ≤150 psig within the following 24 hours.

4.5 - SURVEILLANCE REQUIREMENTS

2) The pump suction is automatically transferred from the condensate storage tank to the suppression chamber on a condensate storage tank water level - low signal and on a suppression chamber water level - high signal.

c. Performing a CHANNEL CALIBRATION of the CS and LPCI system discharge line "keep filled" alarm instrumentation.

d. ~~Performing a CHANNEL CALIBRATION of the CS header ΔP instrumentation and verifying the setpoint to be ≤0.5 psid.~~ 'Deleted'

4. At least once per 18 months for the ADS:

a. 'INSERT' ~~Performing a system functional test which includes simulated automatic actuation of the system throughout its emergency operating sequence, but excluding actual valve actuation.~~

b. Manually opening each ADS valve when the reactor steam dome pressure is ≥150 psig[c] and observing that either:

1) The turbine control valve or turbine bypass valve position responds accordingly, or

2) There is a corresponding change in the measured steam flow.

---

c The provisions of Specification 4.0.D are not applicable provided the surveillance is performed within 12 hours after reactor steam pressure is adequate to perform the test.

d 'INSERT'

INSERT
[ADS SR]

Verify the ADS actuates on an actual or simulated automatic initiation signal. Actual valve actuation may be excluded from this test.

INSERT

The provisions of Specification 3.9.A, Actions 4.a or 6.b are applicable to the LPCI subsystem such that with an inoperable diesel generator, for the remaining OPERABLE diesel generator, both LPCI pumps (and their associated flow path) associated with that OPERABLE diesel generator, shall be OPERABLE.

3.5 - LIMITING CONDITIONS FOR OPERATION

b. With the LPCI subsystem otherwise inoperable~~, provided that both CS~~ (f) subsystems are OPERABLE, restore the LPCI subsystem to OPERABLE status within 7 days or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours[d].

c. With the LPCI subsystem and one or both CS subsystems inoperable, be in at least HOT SHUTDOWN within 12 hours and in COLD SHUTDOWN within the next 24 hours[d].

3. With the HPCI system inoperable, provided both CS subsystems, the LPCI subsystem, the ADS and the Reactor Core Isolation Cooling (RCIC) system are OPERABLE, restore the HPCI system to OPERABLE status within 14 days or be in at least HOT SHUTDOWN within the next 12 hours and reduce reactor steam dome pressure to $\leq$150 psig within the following 24 hours.

4.5 - SURVEILLANCE REQUIREMENTS

b. For the HPCI system, verifying that:

1) The system develops a flow of $\geq$5000 gpm against a ~~test line pressure~~ system head corresponding to ~~a~~ reactor vessel pressure ~~of~~ ~~$\geq$300 psig~~ when steam is being supplied to the turbine between ~~250~~ 150 and ~~325~~ 180 psig[c].

2) The pump suction is automatically transferred from the condensate storage tank to the suppression chamber on a condensate storage tank water level - low signal and on a suppression chamber water level - high signal.

c. Performing a CHANNEL CALIBRATION of the ECCS discharge line "keep filled" alarm instrumentation.

d. 'Deleted' ~~Performing a CHANNEL CALIBRATION of the CS header ΔP instrumentation and verifying the setpoint to be $\leq$4.4 psid.~~

---

d Whenever the two required RHR SDC mode subsystems are inoperable, if unable to attain COLD SHUTDOWN as required by this ACTION, maintain reactor coolant temperature as low as practical by use of alternate heat removal methods.

c The provisions of Specification 4.0.D are not applicable provided the surveillance is performed within 12 hours after reactor steam pressure is adequate to perform the test.

(f) 'INSERT'

INSERT

The provisions of Specification 3.9.A, Actions 4.a or 6.b are applicable to the LPCI subsystem such that with an inoperable diesel generator, for the remaining OPERABLE diesel generator, <u>both</u> LPCI pumps (and their associated flow path) associated with that OPERABLE diesel generator, shall be OPERABLE.

3.5 - LIMITING CONDITIONS FOR OPERATION

4.5 - SURVEILLANCE REQUIREMENTS

b. With two or more of the above required ADS valves inoperable, be in at least HOT SHUTDOWN within 12 hours and reduce reactor steam dome pressure to ≤150 psig within the following 24 hours.

5. With an ECCS discharge line "keep filled" pressure alarm instrumentation CHANNEL inoperable, perform Surveillance Requirement 4.5.A.1.a.1) for CS and LPCI at least once per 24 hours.

6. With a CS subsystem header ΔP instrumentation CHANNEL inoperable, restore the inoperable CHANNEL to OPERABLE status within 72 hours or determine the CS header ΔP locally at least once per 12 hours; otherwise, declare the associated CS subsystem inoperable.

"Deleted"

7. In the event an ECCS system 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.B 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.

3.5 - LIMITING CONDITIONS FOR OPERATION

4. For the ADS:

   a. With one of the above required ADS valves inoperable, provided the HPCI system, both CS subsystems and ~~the~~ three LPCI ~~subsystem~~ pumps are OPERABLE, restore the inoperable ADS valve to OPERABLE status within 14 days or be in at least HOT SHUTDOWN within the next 12 hours and reduce reactor steam dome pressure to ≤150 psig within the following 24 hours.

   b. With two or more of the above required ADS valves inoperable, be in at least HOT SHUTDOWN within 12 hours and reduce reactor steam dome pressure to ≤150 psig within the following 24 hours.

5. With an ECCS discharge line "keep filled" pressure alarm instrumentation CHANNEL inoperable, perform Surveillance Requirement 4.5.A.1.a.1) for CS and LPCI at least once per 24 hours.

4.5 - SURVEILLANCE REQUIREMENTS

4. At least once per 18 months for the ADS:

   a. 'INSERT' Performing a system functional test which includes simulated automatic actuation of the system throughout its emergency operating sequence, but excluding actual valve actuation.

   b. Manually opening each ADS valve when the reactor steam dome pressure is ≥150 psig[c] and observing that either:

      1) The turbine control valve or turbine bypass valve position responds accordingly, or

      2) There is a corresponding change in the measured steam flow.

c The provisions of Specification 4.0.D are not applicable provided the surveillance is performed within 12 hours after reactor steam pressure is adequate to perform the test.

INSERT

[ADS SR]

Verify the ADS actuates on an actual or simulated automatic initiation signal. Actual valve actuation may be excluded from this test.

3.5 - LIMITING CONDITIONS FOR OPERATION

4.5 - SURVEILLANCE REQUIREMENTS

6. With a CS subsystem header ΔP instrumentation CHANNEL inoperable, restore the inoperable CHANNEL to OPERABLE status within 72 hours or determine the CS header ΔP locally at least once per 12 hours; otherwise, declare the associated CS subsystem inoperable.

Deleted

7. In the event an ECCS system is actuated and injects water into the Reactor Coolant System, a Special Report shall be prepared and submitted to the Commission pursuant to Specification ~~6.6.B.4~~ 6.9.B 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.

3.5 - LIMITING CONDITIONS FOR OPERATION

D. Isolation Condenser

The isolation condenser (IC) system shall be OPERABLE.

APPLICABILITY:

OPERATIONAL MODE(s) 1, 2 and 3 with reactor steam dome pressure > 150 psig.

ACTION:

With the IC system inoperable, operation may continue provided the HPCI system is OPERABLE; restore the IC system to OPERABLE status within 14 days or be in at least HOT SHUTDOWN within the next 12 hours and reduce reactor steam dome pressure to ≤150 psig within the following 24 hours.

4.5 - SURVEILLANCE REQUIREMENTS

D. Isolation Condenser

The IC system shall be demonstrated OPERABLE:

1. At least once per 24 hours by~~:~~
   ~~a.~~ ~~V~~erifying the shell side water volume ~~to be ≥ 11,300 gallons.~~ and
   ~~b. Verifying~~ the shell side water temperature to be ~~≤150°F.~~ within limits
2. 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.
3. At least once per 18 months by ~~performing a system functional test which includes simulated automatic actuation and verifying that each automatic valve in the flow path actuates to its correct position.~~ verifying the IC system actuates on an actual or simulated automatic initiation signal.
4. At least once per 5 years by verifying the system heat removal capability ~~to be ≥252.5 x10⁶ BTU/hour~~.

3.5 - LIMITING CONDITIONS FOR OPERATION

D. Reactor Core Isolation Cooling System

The reactor core isolation cooling (RCIC) system shall be OPERABLE with an OPERABLE flow path capable of automatically taking suction from the suppression chamber and transferring the water to the reactor pressure vessel.

APPLICABILITY:

OPERATIONAL MODE(s) 1, 2 and 3 with reactor steam dome pressure >150 psig.

ACTION:

With the RCIC system inoperable, operation may continue provided the HPCI system is OPERABLE; restore the RCIC system to OPERABLE status within 14 days or be in at least HOT SHUTDOWN within the next 12 hours and reduce reactor steam dome pressure to ≤150 psig within the following 24 hours.

4.5 - SURVEILLANCE REQUIREMENTS

D. Reactor Core Isolation Cooling System

The RCIC system shall be demonstrated OPERABLE:

1. At least once per 31 days by:

   a. Verifying ~~by venting at the high point vents~~ that the system piping from the pump discharge valve to the system isolation valve is filled with water. δ

   b. 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.

   c. Verifying that the pump flow controller is in the correct position.

2. At least once per 92 days, when tested pursuant to 4.0.E, by verifying that the RCIC pump develops a flow of ≥400 gpm ~~in the test flow path with~~ against a system head corresponding to reactor vessel ~~operating~~ pressure when steam is being supplied to the turbine between 920 and 1005 psig[a].

3. At least once per 18 months by:

   a. Performing a system functional test which includes simulated automatic actuation and restart and verifying that each automatic valve in the flow path actuates to its correct position. Actual injection of coolant into the reactor vessel may be excluded. e

   Verifying the RCIC system actuates on an actual or simulated automatic initiation signal.

a The provisions of Specification 4.0.D are not applicable provided the surveillance is performed within 12 hours after reactor steam pressure is adequate to perform the test.

3.5 - LIMITING CONDITIONS FOR OPERATION

4.5 - SURVEILLANCE REQUIREMENTS

against a system head corresponding to reactor vessel pressure,

b. Verifying that the system will develop a flow of ≥400 gpm ~~in the test flow path~~ when steam is supplied to the turbine at a pressure between ~~250~~ 150 and ~~325~~ 180 psig[a].

c. Verifying that the suction for the RCIC system is automatically transferred from the condensate storage tank to the suppression pool on a condensate storage tank water level - low signal and on a suppression pool water level - high signal.

a The provisions of Specification 4.0.D are not applicable provided the surveillance is performed within 12 hours after reactor steam pressure is adequate to perform the test.

BASES

With the HPCI system inoperable, adequate core cooling is assured by the OPERABILITY of the redundant and diversified Automatic Depressurization System and both the CS system and LPCI subsystem. In addition, the Isolation Condenser (IC) system, a system for which no credit is taken in the safety analysis, will automatically initiate on a sustained reactor high pressure condition. The HPCI out-of-service period of 14 days is based on the demonstrated OPERABILITY of redundant and diversified low pressure core cooling systems and the IC system.

The surveillance requirements provide adequate assurance that the HPCI system will be OPERABLE when required. Although all active components are testable and full flow can be demonstrated by recirculation through a test loop during reactor operation, a complete system functional test requires a reactor shutdown. The pump discharge piping is maintained full to prevent water hammer damage and to provide cooling at the earliest moment.

Upon failure of the HPCI system to function properly after a small break loss-of-coolant, the Automatic Depressurization System (ADS) automatically causes all OPERABLE main steamline relief valves to open, depressurizing the reactor so that flow from the low pressure core cooling systems can enter the core in time to limit fuel cladding temperature to less than 2200°F. ADS is conservatively required to be OPERABLE whenever reactor vessel pressure exceeds 150 psig. This pressure is substantially below that for which the low pressure core cooling systems can provide adequate core cooling for events requiring ADS.

ADS automatically controls the five main steamline relief valves although safety analyses support a minimum of 4 OPERABLE valves. It is therefore appropriate to permit one valve to be out-of-service for up to 14 days without materially reducing system reliability, ~~provided the appropriate MAPLHGR reduction factor is applied to assure compliance with 10CFR 50.46. The MAPLHGR reduction factors are contained in the CORE OPERATING LIMITS REPORT.~~ e

To preserve single failure criteria, a minimum of two independent OPERABLE low-pressure ECCS subsystems/loops are required in OPERATIONAL MODE(s) 4 and 5 to ensure adequate vessel inventory makeup in the event of an inadvertent vessel draindown. Only a single LPCI pump is required per loop because of the large injection capacity. All of the ECCS may be inoperable provided the reactor head is removed, the reactor cavity is flooded, the spent fuel gates are removed, and the water level is maintained within the limits required by the Refueling Operations specifications.

'INSERT'

3/4.5.C Suppression Chamber

The suppression chamber is required to be OPERABLE as part of the ECCS to ensure that a sufficient supply of water is available to the HPCI and CS systems and the LPCI subsystem in the event of a LOCA. This limit on suppression chamber minimum water volume ensures that sufficient water is available to permit recirculation cooling flow to the core. The OPERABILITY of the suppression chamber in OPERATIONAL MODE(s) 1, 2 or 3 is also required by Specification 3.7.G. e

K

BASES

With the HPCI system inoperable, adequate core cooling is assured by the OPERABILITY of the redundant and diversified Automatic Depressurization System and both the CS system and LPCI subsystem. In addition, the Reactor Core Isolation Cooling (RCIC) system, a system for which no credit is taken in the safety analysis, will automatically initiate on a reactor low water level condition. The HPCI out-of-service period of 14 days is based on the demonstrated OPERABILITY of redundant and diversified low pressure core cooling systems and the RCIC system.

The surveillance requirements provide adequate assurance that the HPCI system will be OPERABLE when required. Although all active components are testable and full flow can be demonstrated by recirculation through a test loop during reactor operation, a complete system functional test requires a reactor shutdown. The pump discharge piping is maintained full to prevent water hammer damage and to provide cooling at the earliest moment.

Upon failure of the HPCI system to function properly after a small break loss-of-coolant, the Automatic Depressurization System (ADS) automatically causes all OPERABLE main steamline relief valves to open, depressurizing the reactor so that flow from the low pressure core cooling systems can enter the core in time to limit fuel cladding temperature to less than 2200°F. ADS is conservatively required to be OPERABLE whenever reactor vessel pressure exceeds 150 psig. This pressure is substantially below that for which the low pressure core cooling systems can provide adequate core cooling for events requiring ADS.

ADS automatically controls the five main steamline relief valves although the safety analyses support a minimum of 4 OPERABLE valves. It is therefore appropriate to permit one valve to be out-of-service for up to 14 days without materially reducing system reliability.

'INSERT'

To preserve single failure criteria, a minimum of two independent OPERABLE low-pressure ECCS subsystems/loops are required in OPERATIONAL MODE(s) 4 and 5 to ensure adequate vessel inventory makeup in the event of an inadvertent vessel draindown. Only a single LPCI pump is required per loop because of the large injection capacity. All of the ECCS may be inoperable provided the reactor head is removed, the reactor cavity is flooded, the spent fuel gates are removed, and the water level is maintained within the limits required by the Refueling Operations specifications.

3/4.5.C Suppression Chamber

The suppression chamber is required to be OPERABLE as part of the ECCS to ensure that a sufficient supply of water is available to the HPCI and CS systems and the LPCI subsystem in the event of a LOCA. This limit on suppression chamber minimum water volume ensures that sufficient water is available to permit recirculation cooling flow to the core. The OPERABILITY of the suppression chamber in OPERATIONAL MODE(s) 1, 2 or 3 is also required by Specification 3.7.~~G~~ K

Repair work might require making the suppression chamber inoperable. This specification will permit those repairs to be made and concurrently provide assurance that the irradiated fuel has an

INSERT
[ADS Bases]

A manual actuation of each ADS valve is performed to verify that the valve and solenoid are functioning properly and that no blockage exists in the ADS discharge lines. This is demonstrated by the response of the turbine control or bypass valve or by a change in the measured steam flow or by any other method suitable to verify steam flow. Adequate reactor steam dome pressure must be available to perform this test to avoid damaging the valve. Sufficient time is therefore allowed after the required pressure is achieved to perform this test once only. The pressure specified for this test is that pressure recommended by the valve manufacturer. Reactor startup is allowed prior to performing this test because valve OPERABILITY and the setpoints for overpressure protection are verified, per ASME requirements, prior to valve installation. Thus, a footnote is included in this SR to indicate that 4.0.D does not apply.

BASES

Repair work might require making the suppression chamber inoperable. This specification will permit those repairs to be made and concurrently provide assurance that the irradiated fuel has an adequate cooling water supply when the suppression chamber must be made inoperable, including draining, in OPERATIONAL MODE(s) 4 or 5.

In OPERATIONAL MODE(s) 4 and 5 the suppression chamber minimum required water volume is reduced because the reactor coolant is maintained at or below 212°F. Since pressure suppression is not required below 212°F, the minimum water volume is based on net positive suction head (NPSH), recirculation volume and vortex prevention plus a safety margin for conservatism. With the suppression chamber water level less than the required limit, all ECCS subsystems are inoperable unless they are aligned to an OPERABLE condensate storage tank. When the suppression chamber level is less than 8 feet, the CS system or the LPCI subsystem is considered OPERABLE only if it can take suction from the condensate storage tank, and the condensate storage tank water level is sufficient to provide the required NPSH for the CS or LPCI pumps. Therefore, a verification that either the suppression chamber water level is greater than or equal to 8 feet or that CS or LPCI is aligned to take suction from the condensate storage tank and the condensate storage tank contains greater than or equal to 140,000 gallons of water, ensures CS or LPCI can supply at least 50,000 gallons of make-up water to the reactor pressure vessel. The CS suction is uncovered at the 90,000 gallon level.

3/4.5.D Isolation Condenser

The isolation condenser is provided for core decay heat removal following reactor isolation from the main condenser and reactor scram. The isolation condenser has a heat removal capacity (252.5 x 10^6 BTU/hour) sufficient to handle the decay heat production at 300 seconds following a scram. Following a reactor scram and an isolation from the main condenser, water will be lost from the reactor vessel through the relief valves during the first 300 seconds. This represents a minor loss relative to the vessel inventory.

The system may be manually initiated at any time. The system is automatically initiated on high reactor pressure in excess of ~~1060~~ 1070 psig sustained for ~~15~~ 17 seconds. The time delay is provided to prevent unnecessary actuation of the system during anticipated turbine trips. Automatic initiation is provided to minimize the coolant loss following isolation from the main condenser. To be considered OPERABLE, the shell side of the isolation condenser must contain at least ~~11,300~~ 20,000 gallons of water. Make-up water to the shell side of the isolation condenser is provided by the condensate transfer pumps from the condensate storage tank. The condensate transfer pumps are OPERABLE from on-site power. The preferred source of make-up water for the Isolation Condenser is the clean demineralized water system. The fire protection system is also available as make-up water.

3.6 - LIMITING CONDITIONS FOR OPERATION

B. Jet Pumps

All jet pumps shall be OPERABLE and flow indication shall be OPERABLE on at least ~~18~~ 19 jet pumps~~[a]~~.

APPLICABILITY:

OPERATIONAL MODE(s) 1 and 2.

ACTION:

1. With one or more jet pumps inoperable for other than inoperable flow indication, be in at least HOT SHUTDOWN within 12 hours.

2. ~~INTENTIONALLY LEFT BLANK.~~ with flow indication inoperable for two or more jet pumps, flow indication shall be restored such that at least 19 jet pumps have OPERABLE flow indication within 12 hours or be in at least HOT SHUTDOWN within the next 12 hours.

3. ~~With flow indication inoperable for both jet pumps on the same jet pump riser, flow indication shall be restored to OPERABLE status for at least one of these jet pumps within 12 hours or be in at least HOT SHUTDOWN within the next 12 hours.~~

4. ~~With flow indication inoperable on both calibrated (double-tap) jet pumps on the same recirculation loop, flow indication shall be restored to OPERABLE status for at least one of these jet pumps within 12 hours or be in at least HOT SHUTDOWN within the next 12 hours.~~

4.6 - SURVEILLANCE REQUIREMENTS

B. Jet Pumps

All jet pumps shall be demonstrated OPERABLE as follows:

1. During two loop operation, at least once per 24 hours while greater than 25% of RATED THERMAL POWER by determining recirculation loop flow, total core flow and individual jet pump flow for each jet pump and verifying that no two of the following conditions occur when both recirculation pumps are operating in accordance with Specification 3.6.C:

   a. The indicated recirculation pump flow differs by >10% from the established speed-flow characteristics.

   b. The indicated total core flow differs by >10% from the established total core flow value derived from established core plate ΔP/core flow relationships.

   c. The indicated flow of any individual jet pump differs from the established patterns by >10%.

   d. The provisions of Specification 4.0.D are not applicable provided that the surveillance is performed within 24 hours after exceeding 25% of RATED THERMAL POWER.

~~a Inoperable flow indication shall not be allowed on both jet pumps sharing a jet pump riser, nor on both calibrated jet pumps on the same recirculation loop.~~

3.6 - LIMITING CONDITIONS FOR OPERATION

D. Idle Recirculation Loop Startup

An idle recirculation loop shall not be started unless the temperature differential between the reactor pressure vessel ~~steam space coolant~~ and the bottom head ~~drain line~~ coolant is ~~≤145°F~~[a], and:

temperature

1. When both loops have been idle, unless the temperature differential between the reactor coolant within the idle loop to be started up and the coolant in the reactor pressure vessel is ~~≤50°F~~, or

2. When only one loop has been idle, unless the temperature differential between the reactor coolant within the idle and operating recirculation loops is ~~≤50°F and the speed of the operating pump is ≤43% of rated pump speed~~.

within limits

APPLICABILITY:

OPERATIONAL MODE(s) 1, 2, 3 and 4.

ACTION:

With temperature differences and/or flow rates exceeding the above limits, ~~suspend startup of any idle recirculation loop~~.

"INSERT"

4.6 - SURVEILLANCE REQUIREMENTS

D. Idle Recirculation Loop Startup

The temperature differentials and flow rate shall be determined to be within the limits within 15 minutes prior to startup of an idle recirculation loop.

a Below 25 psig reactor pressure, this temperature differential is not applicable.

INSERT

1) Suspend startup of any recirculation loop,

2) Restore the parameter(s) to within limits within 30 minutes,

3) Determine if the reactor coolant system is acceptable for continued operation within 72 hours.

Otherwise be in HOT SHUTDOWN in 12 hours and COLD SHUTDOWN within the following 24 hours.

3.6 - LIMITING CONDITIONS FOR OPERATION

E. Safety Valves

The safety valve function of the 9 reactor coolant system safety valves shall be OPERABLE in accordance with the specified code safety valve function lift settings[a] established as:

1 safety valve[b] @1135 psig ±1%
2 safety valves @1240 psig ±1%
2 safety valves @1250 psig ±1%
4 safety valves @1260 psig ±1%

Each installed safety valve shall be closed with OPERABLE position indication.

APPLICABILITY:

OPERATIONAL MODE(s) 1, 2 and 3.

ACTION:

1. With the safety valve function of one or more of the above required safety valves inoperable, be in at least HOT SHUTDOWN within 12 hours and in COLD SHUTDOWN within the next 24 hours.

2. With all position indication inoperable on one or more safety valve(s), restore the inoperable position indication to OPERABLE status within 30 days or be in HOT SHUTDOWN within 12 hours and in COLD SHUTDOWN within the following 24 hours.

'Deleted'

4.6 - SURVEILLANCE REQUIREMENTS

E. Safety Valves

'Deleted'

1. The position indicators for each safety valve shall be demonstrated OPERABLE by performance of a:

   a. CHANNEL CHECK at least once per 31 days, and a

   b. CHANNEL CALIBRATION at least once per 18 months.

2. At least once per 18 months, 1/2 of the safety valves shall be removed, set pressure tested and reinstalled or replaced with spares that have been previously set pressure tested and stored in accordance with manufacturer's recommendations. At least once per 40 months, the safety valves shall be rotated such that all 9 safety valves are removed, set pressure tested and reinstalled or replaced with spares that have been previously set pressure tested and stored in accordance with manufacturer's recommendations.

---

a The lift setting pressure shall correspond to ambient conditions of the valves at nominal operating temperatures and pressures.

b Target Rock combination safety/relief valve.

3.6 - LIMITING CONDITIONS FOR OPERATION

E. Safety Valves

The safety valve function of the 9 reactor coolant system safety valves shall be OPERABLE in accordance with the specified code safety valve function lift settings[a] established as:

1 safety valve[b] @1135 psig ±1%
2 safety valves @1240 psig ±1%
2 safety valves @1250 psig ±1%
4 safety valves @1260 psig ±1%

Each installed safety valve shall be closed with OPERABLE position indication.

'Deleted'

APPLICABILITY:

OPERATIONAL MODE(s) 1, 2 and 3.

ACTION:

1. With the safety valve function of one or more of the above required safety valves inoperable, be in at least HOT SHUTDOWN within 12 hours and in COLD SHUTDOWN within the next 24 hours.

2. With all position indication inoperable on one or more safety valve(s), restore the inoperable position indication to OPERABLE status within 30 days or be in HOT SHUTDOWN within 12 hours and in COLD SHUTDOWN within the following 24 hours.

'Deleted'

4.6 - SURVEILLANCE REQUIREMENTS

E. Safety Valves

1. The position indicators for each safety valve shall be demonstrated OPERABLE by performance of a:

   a. CHANNEL CHECK at least once per 31 days, and a

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2. At least once per 18 months, 1/2 of the safety valves shall be removed, set pressure tested and reinstalled or replaced with spares that have been previously set pressure tested and stored in accordance with manufacturer's recommendations. At least once per 40 months, the safety valves shall be rotated such that all 9 safety valves are removed, set pressure tested and reinstalled or replaced with spares that have been previously set pressure tested and stored in accordance with manufacturer's recommendations.

---

a The lift setting pressure shall correspond to ambient conditions of the valves at nominal operating temperatures and pressures.

b Target Rock combination safety/relief valve.

3.6 - LIMITING CONDITIONS FOR OPERATION

F. Relief Valves

5 reactor coolant system relief valves and the reactuation time delay of two relief valves shall be OPERABLE with the following settings:

Relief Function Setpoint (psig)
Open
≤ 1112 psig
≤ 1112 psig
≤ 1135 psig
≤ 1135 psig
≤ 1135 psig[a]

Each installed relief valve shall be closed with OPERABLE position indication.

APPLICABILITY:

OPERATIONAL MODE(s) 1, 2 and 3.

ACTION:

1. With one or more relief valves open, provided that suppression pool average water temperature is <110°F, take action to close the open relief valve(s); if suppression pool average water temperature is ≥110°F place the reactor mode switch in the Shutdown position.

4.6 - SURVEILLANCE REQUIREMENTS

F. Relief Valves

1. The relief valve function and the reactuation time delay function instrumentation shall be demonstrated OPERABLE by performance of a:
   a. ~~INTENTIONALLY LEFT BLANK~~ CHANNEL FUNCTIONAL TEST of the relief valve function at least once per 18 months, and a
   b. CHANNEL CALIBRATION and LOGIC SYSTEM FUNCTIONAL TEST ~~and simulated automatic operation~~ of the entire system at least once per 18 months.

2. A position indicator for each relief valve shall be demonstrated OPERABLE by performance of a:
   a. CHANNEL CHECK at least once per 31 days, and a
   b. CHANNEL CALIBRATION at least once per 18 months.

'Deleted'

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a Target Rock combination safety/relief valve.

3.6 - LIMITING CONDITIONS FOR OPERATION

F. Relief Valves

5 reactor coolant system relief valves and the reactuation time delay of two relief valves shall be OPERABLE with the following settings:

Relief Function
Setpoint (psig)

Open
≤1115 psig
≤1115 psig
≤1135 psig
≤1135 psig
≤1135 psig[a]

Each installed relief valve shall be closed with OPERABLE position indication.

APPLICABILITY:

OPERATIONAL MODE(s) 1, 2 and 3.

ACTION:

1. With one or more relief valves open, provided that suppression pool average water temperature is <110°F, take action to close the open relief valve(s); if suppression pool average water temperature is ≥110°F place the reactor mode switch in the Shutdown position.

4.6 - SURVEILLANCE REQUIREMENTS

F. Relief Valves

1. The relief valve function and the reactuation time delay function instrumentation shall be demonstrated OPERABLE by performance of a:

   a. ~~INTENTIONALLY LEFT BLANK~~ CHANNEL FUNCTIONAL TEST of the relief valve function at least once per 18 months, and a

   b. CHANNEL CALIBRATION~~,~~ and LOGIC SYSTEM FUNCTIONAL TEST ~~and simulated automatic operation~~ of the entire system at least once per 18 months.

2. A position indicator for each relief valve shall be demonstrated OPERABLE by performance of a:

   a. CHANNEL CHECK at least once per 31 days, and a

   b. CHANNEL CALIBRATION at least once per 18 months.

'Deleted'

a Target Rock combination safety/relief valve.

3.6 - LIMITING CONDITIONS FOR OPERATION

4.6 - SURVEILLANCE REQUIREMENTS

2. With the relief valve function and/or the reactuation time delay of one of the above required reactor coolant system relief valves inoperable, restore the inoperable relief valve function and the reactuation time delay function to OPERABLE status within 14 days or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

3. With the relief valve function and/or the reactuation time delay of more than one of the above required reactor coolant system relief valves inoperable, be in at least HOT SHUTDOWN within 12 hours and in COLD SHUTDOWN within the next 24 hours.

4. With all position indication inoperable on one or more relief valve(s), restore the inoperable position indication to OPERABLE status within 30 days or be in HOT SHUTDOWN within 12 hours and in COLD SHUTDOWN within the following 24 hours.

'Deleted'

3.6 - LIMITING CONDITIONS FOR OPERATION

2. With the relief valve function and/or the reactuation time delay of one of the above required reactor coolant system relief valves inoperable, restore the inoperable relief valve function and the reactuation time delay function to OPERABLE status within 14 days or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

3. With the relief valve function and/or the reactuation time delay of more than one of the above required reactor coolant system relief valves inoperable, be in at least HOT SHUTDOWN within 12 hours and in COLD SHUTDOWN within the next 24 hours.

4. With all position indication inoperable on one or more relief valve(s), restore the inoperable position indication to OPERABLE status within 30 days or be in HOT SHUTDOWN within 12 hours and in COLD SHUTDOWN within the following 24 hours.

'Deleted'

4.6 - SURVEILLANCE REQUIREMENTS

TABLE 3.6.I-1

REACTOR COOLANT SYSTEM CHEMISTRY LIMITS

OPERATIONAL MODE(s)	Chlorides	Conductivity (μmhos/cm @25°C)	pH
1	≤0.2 ppm	≤1.0	5.6≤ pH ≤8.6
2 and 3	≤0.1 ppm	≤2.0	5.6≤ pH ≤8.6

(a) Except during chemical decontamination of Reactor Recirculation or Reactor Water Clean-Up system piping.

3.6 - LIMITING CONDITIONS FOR OPERATION

O. Residual Heat Removal - HOT SHUTDOWN

Two shutdown cooling mode subsystems of the residual heat removal (RHR) system shall be OPERABLE[a] and, unless at least one recirculation pump is in operation, at least one shutdown cooling mode subsystem shall be capable of circulating reactor coolant[b] with each subsystem consisting of at least:

1. One OPERABLE RHR pump, and
2. One OPERABLE RHR heat exchanger.

APPLICABILITY:

OPERATIONAL MODE 3, with reactor vessel pressure less than the RHR cut-in permissive setpoint.

ACTION:

1. With less than the above required RHR shutdown cooling mode subsystems OPERABLE, immediately initiate corrective action to return the required subsystems to OPERABLE status as soon as possible. Within 1 hour and at least once per 24 hours thereafter, demonstrate the operability of at least one alternate method capable of decay heat removal for each inoperable RHR shutdown cooling mode subsystem. Be in at least COLD SHUTDOWN within 24 hours.[c]

4.6 - SURVEILLANCE REQUIREMENTS

O. Residual Heat Removal - HOT SHUTDOWN

At least one shutdown cooling mode subsystem of the residual heat removal system, recirculation pump or alternate method shall be verified to be capable of circulating reactor coolant at least once per 12 hours.

---

a Each shutdown cooling subsystem is considered OPERABLE if it can be manually aligned (remote or local) in the shutdown cooling mode for removal of decay heat. The provisions of Specification 3.0.D are not applicable.

b The RHR shutdown cooling ~~pump~~ subsystem may be removed from operation during hydrostatic testing.

c Whenever the two required RHR SDC mode subsystems are inoperable, if unable to attain COLD SHUTDOWN as required by this ACTION, maintain reactor coolant temperature as low as practical by use of alternate heat removal methods.

3.6 - LIMITING CONDITIONS FOR OPERATION

P. Residual Heat Removal - COLD SHUTDOWN

Two shutdown cooling mode subsystems of the residual heat removal (RHR) system shall be OPERABLE[a] and, unless at least one recirculation pump is in operation, at least one shutdown cooling mode subsystem shall be capable of circulating reactor coolant[b] with each subsystem consisting of at least:

1. One OPERABLE RHR pump, and
2. One OPERABLE RHR heat exchanger.

APPLICABILITY:

OPERATIONAL MODE 4.

ACTION:

1. With less than the above required RHR shutdown cooling mode subsystems OPERABLE, within 1 hour and at least once per 24 hours thereafter, demonstrate the operability of at least one alternate method capable of decay heat removal for each inoperable RHR shutdown cooling mode subsystem.

4.6 - SURVEILLANCE REQUIREMENTS

P. Residual Heat Removal - COLD SHUTDOWN

At least one shutdown cooling mode subsystem of the residual heat removal system, recirculation pump or alternate method shall be verified to be capable of circulating reactor coolant at least once per 12 hours.

---

a Each shutdown cooling subsystem is considered OPERABLE if it can be manually aligned (remote or local) in the shutdown cooling mode for removal of decay heat.

b The RHR shutdown cooling ~~loop~~ subsystem may be removed from operation during hydrostatic testing.

BASES

assurance that the recirculation flow is not bypassing the core through inactive or broken jet pumps. The change in the flow rate of the failed jet pump produces a change in the indicated flow rate of that pump relative to the other pumps in that loop. Comparison of the data with a normal ~~relationship or~~ pattern provides the indication necessary to detect a failed jet pump. 'INSERT'

e

The accuracy of the core flow measurement system is assumed in the derivation of the Safety Limit MINIMUM CRITICAL POWER RATIO. An analysis assuming a loss of flow indication for three jet pumps resulted in uncertainties within the values assumed for the core flow measurement system in the Safety Limit MINIMUM CRITICAL POWER RATIO calculation for both two loop operation and single loop operation. Therefore, plant operation with loss of flow indication in up to two jet pumps is acceptable as long as each jet pump is on a separate riser and no more than one calibrated double tap jet pump per loop is affected.

Recirculation pump speed mismatch limits are in compliance with the ECCS LOCA analysis design criteria. For some limited low probability events with the recirculation loop operating with large speed differences, it is possible for the LPCI loop selection logic to select the wrong loop for injection. Above 80% of RATED THERMAL POWER, the LPCI selection logic is expected to function at a speed differential of 15%. Below 80% of RATED THERMAL POWER, the loop select logic would be expected to function at a speed differential of 20%. Therefore, this specification provides a margin of 5% in pump speed differential before a problem could arise.

the limit specified in the Dresden Administrative Technical Requirements

In order to prevent undue stress on the vessel nozzles and bottom head region, the recirculation loop temperatures shall be within ~~50°F of each other~~ prior to startup of an idle loop. The loop temperature must also be within ~~50°F of the reactor pressure vessel steam space coolant temperature~~ to prevent thermal shock to the recirculation pump and recirculation nozzles. Since the coolant in the bottom of the vessel is at a lower temperature than the coolant in the upper regions of the core, undue stress on the vessel would result if the temperature difference was greater than ~~145°F~~. Additionally, asymmetric speed operation of the recirculation pumps during idle loop startup induces levels of jet pump riser vibration that are higher than normal. The specific limitation of ~~43% of~~ rated pump speed for the operating recirculation pump prior to the start of the idle recirculation pump ensures that the recirculation pump speed mismatch requirements are maintained.

e

the

limit specified in the Dresden Administrative Technical Requirements

'INSERT'

<u>3/4.6.E</u> <u>Safety Valves</u>

<u>3/4.6.F</u> <u>Relief Valves</u>

The American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code requires the reactor pressure vessel be protected from overpressure during upset conditions by self-actuated safety valves. As part of the nuclear pressure relief system, the size and number of safety valves are selected such that peak pressure in the nuclear system will not exceed the ASME Code limits for the reactor coolant pressure boundary. The overpressure protection system must accommodate the most severe pressurization transient. Evaluations have determined that the most severe transient is the closure of all the main steam line isolation valves followed by a reactor

INSERT

[Jet Pump Flow - Bases]

Allowable deviations from the established patterns have been developed based on operation. Since refueling activities (fuel assembly replacement or shuffle, as well as any modification to fuel support orifice size or core plate bypass flow) can affect the relationship between core flow, jet pump flow, and recirculation loop flow, these relationships may need to be reestablished each cycle. Similarly, initial entry into extended single loop operation may also require establishment of these relationships. During the initial weeks of operation under such conditions, while base-lining new "established patterns," engineering judgement of the daily surveillance results is used to detect significant abnormalities which could indicate jet pump failure.

INSERT

[Idle Loop Startup - Bases]

In addition to suspending startup of an idle recirculation loop not meeting the temperature limits, the temperature parameters must be restored within 30 minutes. The 30 minute completion time reflects the urgency of restoring the parameters to within the analyzed range. Most violations will not be severe, and the activity can be accomplished in this time in a controlled manner.

Besides restoring operation within limits, an evaluation is required to determine if operation can continue. The evaluation must verify the reactor coolant system integrity remains acceptable and must be completed if continued operation is desired. Several methods may be used, including comparison with pre-analyzed transients in the stress analyses, new analyses, or inspection of the components.

The 72 hour completion time is reasonable to accomplish the evaluation of a mild violation. More severe violations may require special, event specific stress analyses or inspections. A favorable evaluation must be completed if continued operation is desired.

BASES

assurance that the recirculation flow is not bypassing the core through inactive or broken jet pumps. The change in the flow rate of the failed jet pump produces a change in the indicated flow rate of that pump relative to the other pumps in that loop. Comparison of the data with a normal ~~relationship of~~ pattern provides the indication necessary to detect a failed jet pump. 'INSERT'

The accuracy of the core flow measurement system is assumed in the derivation of the Safety Limit MINIMUM CRITICAL POWER RATIO. An analysis assuming a loss of flow indication for three jet pumps resulted in uncertainties within the values assumed for the core flow measurement system in the Safety Limit MINIMUM CRITICAL POWER RATIO calculation for both two loop operation and single loop operation. Therefore, plant operation with loss of flow indication in up to two jet pumps is acceptable as long as each jet pump is on a separate riser and no more than one calibrated double tap jet pump per loop is affected.

Recirculation pump speed mismatch limits are in compliance with the ECCS LOCA analysis design criteria. For some limited low probability events with the recirculation loop operating with large speed differences, it is possible for the LPCI loop selection logic to select the wrong loop for injection. Above 80% of RATED THERMAL POWER, the LPCI selection logic is expected to function at a speed differential of 15%. Below 80% of RATED THERMAL POWER, the loop select logic would be expected to function at a speed differential of 20%. Therefore, this specification provides a margin of 5% in pump speed differential before a problem could arise.

In order to prevent undue stress on the vessel nozzles and bottom head region, the recirculation loop temperatures shall be within 50°F of each other prior to startup of an idle loop. The loop temperature must also be within 50°F of the reactor pressure vessel steam space coolant temperature to prevent thermal shock to the recirculation pump and recirculation nozzles. Since the coolant in the bottom of the vessel is at a lower temperature than the coolant in the upper regions of the core, undue stress on the vessel would result if the temperature difference was greater than 145°F. Additionally, asymmetric speed operation of the recirculation pumps during idle loop startup induces levels of jet pump riser vibration that are higher than normal. The specific limitation of 45% of rated pump speed for the operating recirculation pump prior to the start of the idle recirculation pump ensures that the recirculation pump speed mismatch requirements are maintained.

3/4.6.E Safety Valves

3/4.6.F Relief Valves

The American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code requires the reactor pressure vessel be protected from overpressure during upset conditions by self-actuated safety valves. As part of the nuclear pressure relief system, the size and number of safety valves are selected such that peak pressure in the nuclear system will not exceed the ASME Code limits for the reactor coolant pressure boundary. The overpressure protection system must accommodate the most severe pressurization transient. Evaluations have determined that the most severe transient is the closure of all the main steam line isolation valves followed by a reactor

INSERT

[Jet Pump Flow - Bases]

Allowable deviations from the established patterns have been developed based on operation. Since refueling activities (fuel assembly replacement or shuffle, as well as any modification to fuel support orifice size or core plate bypass flow) can affect the relationship between core flow, jet pump flow, and recirculation loop flow, these relationships may need to be reestablished each cycle. Similarly, initial entry into extended single loop operation may also require establishment of these relationships. During the initial weeks of operation under such conditions, while base-lining new "established patterns," engineering judgement of the daily surveillance results is used to detect significant abnormalities which could indicate jet pump failure.

BASES

3/4.6.H Operational Leakage

The allowable leakage rates from the reactor coolant system have been based on the predicted and experimentally observed behavior of cracks in pipes. The normally expected background leakage due to equipment design and the detection capability of the instrumentation for determining system leakage was also considered. The evidence obtained from experiments suggests that for leakage somewhat greater than that specified for UNIDENTIFIED LEAKAGE the probability is small that the imperfection or crack associated with such leakage would grow rapidly. However, in all cases, if the leakage rates exceed the values specified or the leakage is located and known to be PRESSURE BOUNDARY LEAKAGE, the reactor will be shutdown to allow further investigation and corrective action.

An UNIDENTIFIED LEAKAGE increase of more than 2 gpm within a 24 hour period is an indication of a potential flaw in the reactor coolant pressure boundary and must be quickly evaluated. Although the increase does not necessarily violate the absolute UNIDENTIFIED LEAKAGE limit, IGSCC susceptible components must be determined not to be the source of the leakage within the required completion time.

3/4.6.I Chemistry

The water chemistry limits of the reactor coolant system are established to prevent damage to the reactor materials in contact with the coolant. Chloride limits are specified to prevent stress corrosion cracking of the stainless steel. The effect of chloride is not as great when the oxygen concentration in the coolant is low, thus the 0.2 ppm limit on chlorides is permitted during POWER OPERATION.

Conductivity measurements are required on a continuous basis since changes in this parameter are an indication of abnormal conditions. When the conductivity is within limits, the pH, chlorides and other impurities affecting conductivity must also be within their acceptable limits. With the conductivity meter inoperable, additional samples must be analyzed to ensure that the chlorides are not exceeding the limits.

Action 1 permits temporary operation with chemistry limits outside of the limits required in OPERATIONAL MODE 1 without requiring Commission notification. The surveillance requirements provide adequate assurance that concentrations in excess of the limits will be detected in sufficient time to take corrective action.

3/4.6.J Specific Activity

The limitations on the specific activity of the primary coolant ensure that the 2 hour thyroid and whole body doses resulting from a main steam line failure outside the containment during steady state operation will not exceed small fractions of the dose guidelines of 10 CFR 100. The values

BASES

for pressure in excess of 20% of the preservice hydrostatic test pressure (1563 psig). At approximately 650 psig the effects of pressurization are more limiting than the boltup stresses at the closure flange region, hence a family of non-linear curves intersect the 130°F vertical line. Beltline as well as non-beltline curves have been provided to allow separate monitoring of the two regions. Beltline curves as a function of vessel exposure for 12, 14 and 16 effective full power years (EFPY) are presented to allow the use of the appropriate curve up to 16 EFPY of operation.

A typical sequence involved in pressure testing is a heatup to the required temperature and then pressurization to the required pressure for the inspection. During the heatup, at 100°F/hour or less, Curve B is the governing curve. Since the vessel is not pressurized during the heatup, Curves A and B are the same. When temperatures are stabilized to within 20°F/hour rates, at temperatures above those required by curve A, pressurization begins, at which point Curve A is the governing curve. During the inspection period with the vessel at the required pressure, temperature changes are limited to 20°F/hour.

Curve B - Non-Nuclear Heatup/Cooldown

Curve B of Figure 3.6.K-1 applies during heatups with non-nuclear heat (e.g., recirculation pump heat) and during cooldowns when the reactor is not critical (e.g., following a scram). The curve provides the minimum reactor vessel metal temperatures based on the most limiting vessel stress. As indicated by the vertical 100°F line, the boltup stresses at the closure flange region are most limiting for reactor pressures below approximately 110 psig. For reactor pressures greater than approximately 110 psig, pressurization and thermal stresses become more limiting than the boltup stresses, which is reflected by the nonlinear portion of curve B. The non-linear portion of the curve is dependent on non-beltline and beltline regions, with the beltline region temperature limits having been adjusted to account for vessel irradiation (up to a vessel exposure of 16 EFPY). The non-beltline region is limiting between approximately 110 psig and 830 psig. Above approximately 803 psig, the beltline region becomes limiting.

Curve C - Core Critical Operation

Curve C, the core critical operation curve shown in Figure 3.6.K-1, is generated in accordance with 10CFR Part 50 Appendix G which requires core critical pressure-temperature limits to be 40°F above any curve A or B limits. Since curve B is more limiting, (curve C is curve B plus 40°F.

The actual shift in RT_{NDT} of the vessel material will be established periodically during operation by removing and evaluating, in accordance with ASTM E185-73 and 10CFR Part 50, Appendix H, irradiated reactor vessel material specimens installed near the inside wall of the reactor vessel in the core area. The irradiated specimens can be used with confidence in predicting reactor vessel material transition temperature shift. The operating limit curves of Figure 3.6.~~J~~K-1 shall be adjusted, as required, on the basis of the specimen data and recommendations of Regulatory Guide 1.99, Revision 2.

BASES

for pressure in excess of 20% of the preservice hydrostatic test pressure (1563 psig). At approximately 650 psig the effects of pressurization are more limiting than the boltup stresses at the closure flange region, hence a family of non-linear curves intersect the 130°F vertical line. Beltline as well as non-beltline curves have been provided to allow separate monitoring of the two regions. Beltline curves as a function of vessel exposure for 12, 14 and 16 effective full power years (EFPY) are presented to allow the use of the appropriate curve up to 16 EFPY of operation.

A typical sequence involved in pressure testing is a heatup to the required temperature and then pressurization to the required pressure for the inspection. During the heatup, at 100°F/hour or less, Curve B is the governing curve. Since the vessel is not pressurized during the heatup, Curves A and B are the same. When temperatures are stabilized to within 20°F/hour rates, at temperatures above those required by curve A, pressurization begins, at which point Curve A is the governing curve. During the inspection period with the vessel at the required pressure, temperature changes are limited to 20°F/hour.

Curve B - Non-Nuclear Heatup/Cooldown

Curve B of Figure 3.6.K-1 applies during heatups with non-nuclear heat (e.g., recirculation pump heat) and during cooldowns when the reactor is not critical (e.g., following a scram). The curve provides the minimum reactor vessel metal temperatures based on the most limiting vessel stress. As indicated by the vertical 100°F line, the boltup stresses at the closure flange region are most limiting for reactor pressures below approximately 110 psig. For reactor pressures greater than approximately 110 psig, pressurization and thermal stresses become more limiting than the boltup stresses, which is reflected by the nonlinear portion of curve B. The non-linear portion of the curve is dependent on non-beltline and beltline regions, with the beltline region temperature limits having been adjusted to account for vessel irradiation (up to a vessel exposure of 16 EFPY). The non-beltline region is limiting between approximately 110 psig and 830 psig. Above approximately 803 psig, the beltline region becomes limiting.

Curve C - Core Critical Operation

Curve C, the core critical operation curve shown in Figure 3.6.K-1, is generated in accordance with 10CFR Part 50 Appendix G which requires core critical pressure-temperature limits to be 40°F above any curve A or B limits. Since curve B is more limiting, (curve C is curve B plus 40°F.

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K

BASES

3/4.6.L Reactor Steam Dome Pressure

The reactor steam dome pressure is an assumed initial condition of Design Basis Accidents and transients and is also an assumed value in the determination of compliance with reactor pressure vessel overpressure protection criteria. The reactor steam dome pressure of ≤1005 psig is an initial condition of the vessel overpressure protection analysis. This analysis assumes an initial maximum reactor steam dome pressure and evaluates the response of the pressure relief system, primarily the safety valves, during the limiting pressurization transient. The determination of compliance with the overpressure criteria is dependent on the initial reactor steam dome pressure; therefore, the limit on this pressure ensures that the assumptions of the overpressure protection analysis are conserved.

3/4.6.M Main Steam Line Isolation Valves

Double isolation valves are provided on each of the main steam lines to minimize the potential leakage paths from the containment in case of a line break. Only one valve in each line is required to maintain the integrity of the containment, however, single failure considerations require that two valves be OPERABLE. The surveillance requirements are based on the operating history of this type of valve. The maximum closure time has been selected to contain fission products and to ensure the core is not uncovered following line breaks. The minimum closure time is consistent with the assumptions in the safety analyses to prevent pressure surges.

3/4.6.N Structural Integrity

The inspection programs for ASME Code Class 1, 2 and 3 components ensure that the structural integrity of these components will be maintained at an acceptable level throughout the life of the plant.

The inservice inspection program for ASME Code Class 1, 2 and 3 components will be performed in accordance with Section XI of the ASME Boiler and Pressure Vessel Code and applicable addenda as required by 10 CFR Part 50.55a(g) except where specific written relief has been granted by the NRC pursuant to 10 CFR Part 50.55a(g)(6)(i).

3/4.6.O Shutdown Cooling - HOT SHUTDOWN

3/4.6.P Shutdown Cooling - COLD SHUTDOWN

Irradiated fuel in the reactor pressure vessel generates decay heat during normal and abnormal shutdown conditions, potentially resulting in an increase in the temperature of the reactor coolant. This decay heat is required to be removed such that the reactor coolant temperature can be reduced in preparation for performing refueling, maintenance operations or for maintaining the

BASES

3/4.6.L Reactor Steam Dome Pressure

The reactor steam dome pressure is an assumed initial condition of Design Basis Accidents and transients and is also an assumed value in the determination of compliance with reactor pressure vessel overpressure protection criteria. The reactor steam dome pressure of ≤1005 psig is an initial condition of the vessel overpressure protection analysis. This analysis assumes an initial maximum reactor steam dome pressure and evaluates the response of the pressure relief system, primarily the safety valves, during the limiting pressurization transient. The determination of compliance with the overpressure criteria is dependent on the initial reactor steam dome pressure; therefore, the limit on this pressure ensures that the assumptions of the overpressure protection analysis are conserved.

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Double isolation valves are provided on each of the main steam lines to minimize the potential leakage paths from the containment in case of a line break. Only one valve in each line is required to maintain the integrity of the containment, however, single failure considerations require that two valves be OPERABLE. The surveillance requirements are based on the operating history of this type of valve. The maximum closure time has been selected to contain fission products and to ensure the core is not uncovered following line breaks. The minimum closure time is consistent with the assumptions in the safety analyses to prevent pressure surges.

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3/4.6.O Residual Heat Removal - HOT SHUTDOWN

3/4.6.P Residual Heat Removal - COLD SHUTDOWN

Irradiated fuel in the reactor pressure vessel generates decay heat during normal and abnormal shutdown conditions, potentially resulting in an increase in the temperature of the reactor coolant. This decay heat is required to be removed such that the reactor coolant temperature can be reduced in preparation for performing refueling, maintenance operations or for maintaining the

3.7 - LIMITING CONDITIONS FOR OPERATION

A. PRIMARY CONTAINMENT INTEGRITY

PRIMARY CONTAINMENT INTEGRITY shall be maintained.

APPLICABILITY:

OPERATIONAL MODE(s) 1, 2[a] and 3.

ACTION:

Without PRIMARY CONTAINMENT INTEGRITY, restore PRIMARY CONTAINMENT INTEGRITY within 1 hour or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

4.7 - SURVEILLANCE REQUIREMENTS

A. PRIMARY CONTAINMENT INTEGRITY

PRIMARY CONTAINMENT INTEGRITY shall be demonstrated:

1. After each closing of each penetration subject to Type B testing, except the primary containment air locks, if opened following Type A or B test, by leak rate testing the seals with gas at $\geq P_a$ (48 psig), and verifying that when the measured leakage rate for these seals is added to the leakage rates determined pursuant to Surveillance Requirement 4.7.B.4 for all other Type B and C penetrations, the combined leakage rate is $\leq 0.60\ L_a$.

2. At least once per 31 days by verifying that all primary containment penetrations[b] not capable of being closed by OPERABLE containment automatic isolation valves and required to be closed during accident conditions ~~are closed~~ e by valves, blind flanges, or deactivated automatic valves secured in position, except for valves that are open under administrative control as permitted by Specification 3.7.D.

3. By verifying each primary containment air lock is in compliance with the requirements of Specification 3.7.C.

4. By verifying the suppression chamber is in compliance with the requirements of Specification 3.7.K.

---

a See Special Test Exception 3.12.A.

b Except valves, blind flanges, and deactivated automatic valves which are located inside the containment ~~and are locked, sealed or otherwise secured in the closed position~~ e. These penetrations shall be verified closed during each COLD SHUTDOWN except such verification need not be performed when the primary containment has not been de-inerted since the last verification or more often than once per 92 days.

Valves and blind flanges in high radiation areas may be verified by use of administrative controls.

3.7 - LIMITING CONDITIONS FOR OPERATION

A. PRIMARY CONTAINMENT INTEGRITY

PRIMARY CONTAINMENT INTEGRITY shall be maintained.

APPLICABILITY:

OPERATIONAL MODE(s) 1, 2[a] and 3.

ACTION:

Without PRIMARY CONTAINMENT INTEGRITY, restore PRIMARY CONTAINMENT INTEGRITY within 1 hour or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

4.7 - SURVEILLANCE REQUIREMENTS

A. PRIMARY CONTAINMENT INTEGRITY

PRIMARY CONTAINMENT INTEGRITY shall be demonstrated:

1. After each closing of each penetration subject to Type B testing, except the primary containment air locks, if opened following Type A or B test, by leak rate testing the seals with gas at $\geq P_a$ (48 psig), and verifying that when the measured leakage rate for these seals is added to the leakage rates determined pursuant to Surveillance Requirement 4.7.B.4 for all other Type B and C penetrations, the combined leakage rate is $\leq 0.60\ L_a$.
2. At least once per 31 days by verifying that all primary containment penetrations[b] not capable of being closed by OPERABLE containment automatic isolation valves and required to be closed during accident conditions are closed by valves, blind flanges, or deactivated automatic valves secured in position, except for valves that are open under administrative control as permitted by Specification 3.7.D.
3. By verifying each primary containment air lock is in compliance with the requirements of Specification 3.7.C.
4. By verifying the suppression chamber is in compliance with the requirements of Specification 3.7.K.

---

a See Special Test Exception 3.12.A.

b Except valves, blind flanges, and deactivated automatic valves which are located inside the containment ~~and are locked, sealed or otherwise secured in the closed position~~. These penetrations shall be verified closed during each COLD SHUTDOWN except such verification need not be performed when the primary containment has not been de-inerted since the last verification or more often than once per 92 days.

Valves and blind flanges in high radiation areas may be verified by use of administrative controls.

3.7 - LIMITING CONDITIONS FOR OPERATION

B. Primary Containment Leakage

Primary containment leakage rates shall be limited to:

1. An overall integrated leakage rate of $\leq L_a$ which is defined as 1.6 percent by weight of the containment air per 24 hours at P_a (48 psig).
2. A combined leakage rate of $\leq 0.60\ L_a$ for all primary containment penetrations, except[a] for main steam line isolation valves, subject to Type B and C tests when pressurized to P_a (48 psig).
3. ≤ 11.5 scfh for any one main steam line isolation valve when tested at P_t (25 psig)[a].

APPLICABILITY:

When PRIMARY CONTAINMENT INTEGRITY is required per Specification 3.7.A.

ACTION:

With the measured combined leakage rate for all primary containment penetrations subject to Type B and C tests $> 0.60\ L_a$, restore the combined leakage rate to $\leq 0.60\ L_a$, within 1 hour. Otherwise, be in HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

4.7 - SURVEILLANCE REQUIREMENTS

B. Primary Containment Leakage

The primary containment leakage rates shall be demonstrated at the following test schedule and shall be determined in conformance with the criteria, methods and provisions specified in Appendix J of 10CFR Part 50: as modified by approved exemptions

1. Three Type A overall integrated containment leakage rate tests shall be conducted at approximately equal intervals during shutdown at $\geq P_a$ (48 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.
2. If the results of any periodic Type A test are $> 0.75\ L_a$, the test schedule for subsequent Type A tests shall be reviewed and approved by the Commission. If the results of two consecutive Type A tests are $> 0.75\ L_a$, a Type A test shall be performed ~~at least every 18 months~~ at intervals in accordance with 10 CFR Part 50, Appendix J, as modified by approved exemptions, until the results of two consecutive Type A tests are $\leq 0.75\ L_a$, at which time the above test schedule may be resumed.
3. The accuracy of each Type A test shall be verified by a supplemental test which:
   a. Confirms the accuracy of the test by verifying that the difference between the supplemental data and the Type A test data is within $0.25\ L_a$.

a Exemption from Appendix J to 10CFR Part 50.

3.7 - LIMITING CONDITIONS FOR OPERATION

B. Primary Containment Leakage

Primary containment leakage rates shall be limited to:

1. An overall integrated leakage rate of $\leq L_a$ which is defined as 1.0 percent by weight of the containment air per 24 hours at P_a (48 psig).
2. A combined leakage rate of $\leq 0.60\ L_a$ for all primary containment penetrations, except[a] for main steam line isolation valves, subject to Type B and C tests when pressurized to P_a (48 psig).
3. ≤ 11.5 scfh for any one main steam line isolation valve when tested at P_t (25 psig)[a].

APPLICABILITY:

When PRIMARY CONTAINMENT INTEGRITY is required per Specification 3.7.A.

ACTION:

With the measured combined leakage rate for all primary containment penetrations subject to Type B and C tests $> 0.60\ L_a$, restore the combined leakage rate to $\leq 0.60\ L_a$, within 1 hour. Otherwise, be in HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

4.7 - SURVEILLANCE REQUIREMENTS

B. Primary Containment Leakage

The primary containment leakage rates shall be demonstrated at the following test schedule and shall be determined in conformance with the criteria, methods and provisions specified in Appendix J of 10CFR Part 50 , as modified by approved exemptions.

1. Three Type A overall integrated containment leakage rate tests shall be conducted at approximately equal intervals during shutdown at $\geq P_a$ (48 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.
2. If the results of any periodic Type A test are $> 0.75\ L_a$, the test schedule for subsequent Type A tests shall be reviewed and approved by the Commission. If the results of two consecutive Type A tests are $> 0.75\ L_a$, a Type A test shall be performed ~~at least every 18 months~~ at intervals in accordance with 10 CFR Part 50, Appendix J, as modified by approved exemptions, until the results of two consecutive Type A tests are $\leq 0.75\ L_a$, at which time the above test schedule may be resumed.
3. The accuracy of each Type A test shall be verified by a supplemental test which:

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

a Exemption from Appendix J to 10CFR Part 50.

3.7 - LIMITING CONDITIONS FOR OPERATION

4.7 - SURVEILLANCE REQUIREMENTS

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

c. Requires the quantity of gas to be bled from the containment during the supplemental test to be equivalent to at least 25% of the total measured leakage at $\geq P_a$ (48 psig).

4. Type B and C tests shall be conducted with gas at $\geq P_a$ (48 psig) ~~at intervals no greater than 24 months~~ at intervals in accordance with 10 CFR 50, Appendix J, as modified by approved exemptions, except for tests involving:

a. Air locks which shall be leak tested in accordance with Surveillance Requirement 4.7.C,

b. Main steam line isolation valves[a] which shall be leak tested at $\geq P_t$ (25 psig)[a] ~~at least once per 18 months~~, and

c. Bolted double-gasketed seals which shall be leak tested at $\geq P_a$ (48 psig) following each closure of the seal ~~and at least once every 18 months~~ and at intervals in accordance with 10 CFR 50, Appendix J, as modified by approved exemptions.

~~The provisions of Specification 4.0.B are not applicable to the 24 month surveillance intervals.~~

a Exemption from Appendix J to 10CFR Part 50.

3.7 - LIMITING CONDITIONS FOR OPERATION

4.7 - SURVEILLANCE REQUIREMENTS

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

c. Requires the quantity of gas to be bled from the containment during the supplemental test to be equivalent to at least 25% of the total measured leakage at $\geq P_a$ (48 psig).

4. Type B and C tests shall be conducted with gas at $\geq P_a$ (48 psig) ~~at intervals no greater than 24 months~~ at intervals in accordance with 10 CFR 50, Appendix J, as modified by approved exemptions, except for tests involving:

a. Air locks which shall be leak tested in accordance with Surveillance Requirement 4.7.C,

b. Main steam line isolation valves[a] which shall be leak tested at $\geq P_t$ (25 psig)[a] ~~at least once per 18 months~~, and

c. Bolted double-gasketed seals which shall be leak tested at $\geq P_a$ (48 psig) following each closure of the seal ~~and at least once every 18 months~~ and at intervals in accordance with 10 CFR 50, Appendix J, as modified by approved exemptions.

~~The provisions of Specification 4.0.B are not applicable to the 24 month surveillance intervals.~~

a Exemption from Appendix J to 10CFR Part 50.

3.7 - LIMITING CONDITIONS FOR OPERATION

c. Otherwise, be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

2. With the primary containment air lock interlock mechanism inoperable, restore the air lock interlock mechanism to OPERABLE status within 24 hours, or lock at least one air lock door closed and verify that the door is locked closed at least once per 31 days. Personnel entry and exit through the airlock is permitted provided one OPERABLE air lock door remains locked closed at all times and an individual is dedicated to assure that both air lock doors are not open simultaneously.
ed

3. With the primary containment air lock inoperable, except as a result of an inoperable air lock door or interlock mechanism, maintain at least one air lock door closed; restore the inoperable air lock to OPERABLE status within 24 hours or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

4.7 - SURVEILLANCE REQUIREMENTS

3.7 - LIMITING CONDITIONS FOR OPERATION

c. Otherwise, be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

2. With the primary containment air lock interlock mechanism inoperable, restore the air lock interlock mechanism to OPERABLE status within 24 hours, or lock at least one air lock door closed and verify that the door is locked closed at least once per 31 days. Personnel entry and exit through the airlock is permitted provided one OPERABLE air lock door remains locked closed at all times and an individual is dedicated to assure that both air lock doors are not open ed simultaneously.

3. With the primary containment air lock inoperable, except as a result of an inoperable air lock door or interlock mechanism, maintain at least one air lock door closed; restore the inoperable air lock to OPERABLE status within 24 hours or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

4.7 - SURVEILLANCE REQUIREMENTS

3.7 - LIMITING CONDITIONS FOR OPERATION

H. Drywell - Suppression Chamber Differential Pressure

Differential pressure between the drywell and the suppression chamber shall be $\geq$1.0 psid[a].

APPLICABILITY:

OPERATIONAL MODE 1, during the time period:

1. Beginning within 24 hours after THERMAL POWER is >15% of RATED THERMAL POWER following startup, and
2. Ending within 24 hours prior to reducing THERMAL POWER to <15% of RATED THERMAL POWER preliminary to a scheduled reactor shutdown.

≤

ACTION:

1. With the drywell - suppression chamber differential pressure less than the above limit, restore the required differential pressure within 24 hours or reduce THERMAL POWER to <15% RATED THERMAL POWER within the next 8 hours.
2. With the drywell - suppression chamber differential pressure instrumentation CHANNEL inoperable, restore the inoperable CHANNEL to OPERABLE status within 30 days or reduce THERMAL POWER to <15% RATED THERMAL POWER within the next 8 hours.

4.7 - SURVEILLANCE REQUIREMENTS

H. Drywell - Suppression Chamber Differential Pressure

1. The drywell - suppression chamber differential pressure shall be demonstrated to be within limits by verifying the differential pressure at least once per 12 hours.
2. At least one drywell - suppression chamber differential pressure instrumentation CHANNEL, and at least one drywell pressure and one suppression chamber pressure instrumentation CHANNEL shall be demonstrated OPERABLE by performance of a:
   a. CHANNEL CHECK at least once per 24 hours,
   b. CHANNEL CALIBRATION at least once per 18 months

every 31 days.

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a Except for up to 4 hours for required surveillance which reduces the differential pressure.

3.7 - LIMITING CONDITIONS FOR OPERATION

H. Drywell - Suppression Chamber Differential Pressure

Differential pressure between the drywell and the suppression chamber shall be $\geq$1.0 psid[a].

APPLICABILITY:

OPERATIONAL MODE 1, during the time period:

1. Beginning within 24 hours after THERMAL POWER is >15% of RATED THERMAL POWER following startup, and
2. Ending within 24 hours prior to reducing THERMAL POWER to <15% of RATED THERMAL POWER preliminary to a scheduled reactor shutdown.

≤

ACTION:

1. With the drywell - suppression chamber differential pressure less than the above limit, restore the required differential pressure within 24 hours or reduce THERMAL POWER to <15% RATED THERMAL POWER within the next 8 hours.
2. With the drywell - suppression chamber differential pressure instrumentation CHANNEL inoperable, restore the inoperable CHANNEL to OPERABLE status within 30 days or reduce THERMAL POWER to <15% RATED THERMAL POWER within the next 8 hours.

4.7 - SURVEILLANCE REQUIREMENTS

H. Drywell - Suppression Chamber Differential Pressure

1. The drywell - suppression chamber differential pressure shall be demonstrated to be within limits by verifying the differential pressure at least once per 12 hours.
2. At least one drywell - suppression chamber differential pressure instrumentation CHANNEL, and at least one drywell pressure and one suppression chamber pressure instrumentation CHANNEL shall be demonstrated OPERABLE by performance of a:
   a. CHANNEL CHECK at least once per 24 hours,
   b. CHANNEL CALIBRATION at least once per 18 months.

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a Except for up to 4 hours for required surveillance which reduces the differential pressure.

3.7 - LIMITING CONDITIONS FOR OPERATION

I. ~~Primary Containment Nitrogen System~~

The primary ~~containment nitrogen~~ system shall be OPERABLE with:

1. An OPERABLE inerting flow path, and
2. An OPERABLE make-up flow path.

APPLICABILITY:

OPERATIONAL MODE(s) 1 and 2.

ACTION:

With the primary containment nitrogen system inoperable, restore the system to OPERABLE status within 30 days or be in at least HOT SHUTDOWN within the next 12 hours.

4.7 - SURVEILLANCE REQUIREMENTS

I. ~~Primary Containment Nitrogen System~~

The primary ~~containment nitrogen~~ system shall be demonstrated to be OPERABLE at least once per 31 days by verifying that:

1. The liquid nitrogen storage tank level is ≥70 inches, and
2. Each valve, manual, power operated or automatic, in the flow path not locked, sealed, or otherwise secured in position, is in its correct position.

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3.7 - LIMITING CONDITIONS FOR OPERATION

J. Primary Containment Oxygen Concentration

The suppression chamber and drywell atmosphere oxygen concentration shall be <4% by volume.

APPLICABILITY:

OPERATIONAL MODE 1, during the time period:

1. Beginning within 24 hours after THERMAL POWER is >15% of RATED THERMAL POWER following startup, and
2. Ending within 24 hours prior to reducing THERMAL POWER to <15% of RATED THERMAL POWER preliminary to a scheduled reactor shutdown.

ACTION:

With the drywell and/or suppression chamber oxygen concentration exceeding the limit, restore the oxygen concentration to within the limit within 24 hours or reduce THERMAL POWER to <15% RATED THERMAL POWER within the next 8 hours.

4.7 - SURVEILLANCE REQUIREMENTS

J. Primary Containment Oxygen Concentration

The suppression chamber and drywell oxygen concentration shall be verified to be within the limit within 24 hours after THERMAL POWER is >15% of RATED THERMAL POWER and at least once per 7 days thereafter.

3.7 - LIMITING CONDITIONS FOR OPERATION

I. ~~Primary Containment Nitrogen System~~

The primary containment nitrogen system shall be OPERABLE with:

1. An OPERABLE inerting flow path, and

2. An OPERABLE make-up flow path.

APPLICABILITY:

OPERATIONAL MODE(s) 1 and 2.

ACTION:

With the primary containment nitrogen system inoperable, restore the system to OPERABLE status within 30 days or be in at least HOT SHUTDOWN within the next 12 hours.

4.7 - SURVEILLANCE REQUIREMENTS

I. ~~Primary Containment Nitrogen System~~

The primary containment nitrogen system shall be demonstrated to be OPERABLE at least once per 31 days by verifying that:

1. The liquid nitrogen storage tank level is ≥70 inches, and

2. Each valve, manual, power operated or automatic, in the flow path not locked, sealed, or otherwise secured in position, is in its correct position.

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3.7 - LIMITING CONDITIONS FOR OPERATION

J. Primary Containment Oxygen Concentration

The suppression chamber and drywell atmosphere oxygen concentration shall be <4% by volume.

APPLICABILITY:

OPERATIONAL MODE 1, during the time period:

1. Beginning within 24 hours after THERMAL POWER is >15% of RATED THERMAL POWER following startup, and
2. Ending within 24 hours prior to reducing THERMAL POWER to <15% of RATED THERMAL POWER preliminary to a scheduled reactor shutdown.

ACTION:

With the drywell and/or suppression chamber oxygen concentration exceeding the limit, restore the oxygen concentration to within the limit within 24 hours or reduce THERMAL POWER to <15% RATED THERMAL POWER within the next 8 hours.

4.7 - SURVEILLANCE REQUIREMENTS

J. Primary Containment Oxygen Concentration

The suppression chamber and drywell oxygen concentration shall be verified to be within the limit within 24 hours after THERMAL POWER is >15% of RATED THERMAL POWER and at least once per 7 days thereafter.

3.7 - LIMITING CONDITIONS FOR OPERATION

within 1 hour or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

2. In OPERATIONAL MODE(s) 1 or 2 with the suppression pool average water temperature >95°F, except as permitted above, restore the average temperature to ≤95°F within 24 hours or reduce THERMAL POWER to ≤1% RATED THERMAL POWER within the next 12 hours.

3. With the suppression pool average water temperature >105°F during testing which adds heat to the suppression pool, except as permitted above, stop all testing which adds heat to the suppression pool and restore the average temperature to ≤95°F within 24 hours or reduce THERMAL POWER to ≤1% RATED THERMAL POWER within the next 12 hours.

4. With the suppression pool average water temperature >110°F, immediately place the reactor mode switch in the Shutdown position and operate at least one low pressure coolant injection loop in the suppression pool cooling mode.

5. With the suppression pool average water temperature >120°F, depressurize the reactor pressure vessel to <150 psig (reactor steam dome pressure) within 12 hours.

4.7 - SURVEILLANCE REQUIREMENTS

'Deleted'

3. By an external visual examination of the suppression chamber after main steam relief valve operation with the suppression pool average water temperature ≥160°F and reactor coolant system pressure >150 psig.

4. At least once per 18 months by a visual inspection of the accessible interior and exterior of the suppression chamber.

5. At least once per 18 months by conducting a drywell to suppression chamber bypass leak test at an initial differential pressure of 1.0 psid and verifying that the measured leakage is within the specified limit. If any drywell to suppression chamber bypass leak test fails to meet the specified limit, the test schedule for subsequent tests shall be reviewed and approved by the Commission. If two consecutive tests fail to meet the specified limit, a test shall be performed at least every 9 months until two consecutive tests meet the specified limit, at which time the 18 month test schedule may be resumed.

3.7 - LIMITING CONDITIONS FOR OPERATION

within 1 hour or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

2. In OPERATIONAL MODE(s) 1 or 2 with the suppression pool average water temperature >95°F, except as permitted above, restore the average temperature to ≤95°F within 24 hours or reduce THERMAL POWER to ≤1% RATED THERMAL POWER within the next 12 hours.

3. With the suppression pool average water temperature >105°F during testing which adds heat to the suppression pool, except as permitted above, stop all testing which adds heat to the suppression pool and restore the average temperature to ≤95°F within 24 hours or reduce THERMAL POWER to ≤1% RATED THERMAL POWER within the next 12 hours.

4. With the suppression pool average water temperature >110°F, immediately place the reactor mode switch in the Shutdown position and operate at least one residual heat removal loop in the suppression pool cooling mode.

5. With the suppression pool average water temperature >120°F, depressurize the reactor pressure vessel to <150 psig (reactor steam dome pressure) within 12 hours.

4.7 - SURVEILLANCE REQUIREMENTS

'Deleted'

3. By an external visual examination of the suppression chamber after main steam relief valve operation with the suppression pool average water temperature ≥160°F and reactor coolant system pressure >150 psig.

4. At least once per 18 months by a visual inspection of the accessible interior and exterior of the suppression chamber.

5. At least once per 18 months by conducting a drywell to suppression chamber bypass leak test at an initial differential pressure of 1.0 psid and verifying that the measured leakage is within the specified limit. If any drywell to suppression chamber bypass leak test fails to meet the specified limit, the test schedule for subsequent tests shall be reviewed and approved by the Commission. If two consecutive tests fail to meet the specified limit, a test shall be performed at least every 9 months until two consecutive tests meet the specified limit, at which time the 18 month test schedule may be resumed.

3.7 - LIMITING CONDITIONS FOR OPERATION

N. SECONDARY CONTAINMENT INTEGRITY

SECONDARY CONTAINMENT INTEGRITY shall be maintained.

APPLICABILITY:

OPERATIONAL MODE(s) 1, 2, 3 and *.

ACTION:

1. Without SECONDARY CONTAINMENT INTEGRITY in OPERATIONAL MODES(s) 1, 2 or 3, restore SECONDARY CONTAINMENT INTEGRITY within 4 hours or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

2. Without SECONDARY CONTAINMENT INTEGRITY in OPERATIONAL MODE *, suspend handling of irradiated fuel in the secondary containment, CORE ALTERATION(s), and operations with a potential for draining the reactor vessel. The provisions of Specification 3.0.C are not applicable.

4.7 - SURVEILLANCE REQUIREMENTS

N. SECONDARY CONTAINMENT INTEGRITY

SECONDARY CONTAINMENT INTEGRITY shall be demonstrated by:

1. Verifying at least once per 24 hours that the pressure within the secondary containment is ≥0.25 inches of vacuum water gauge.

2. Verifying at least once per 31 days that:

   a. At least one door in each secondary containment air lock is closed.

   b. All secondary (a) containment penetrations not capable of being closed by OPERABLE secondary containment automatic isolation dampers and required to be closed during accident conditions are closed by valves, blind flanges, or deactivated automatic dampers secured in position.

3. At least once per 18 months by operating one standby gas treatment subsystem at a flow rate ≤4000 cfm for one hour and maintaining ≥0.25 inches of vacuum water gauge in the secondary containment.

'INSERT'

* When handling irradiated fuel in the secondary containment, during CORE ALTERATION(s), and operations with a potential for draining the reactor vessel.

INSERT

a Valves and blind flanges in high-radiation areas may be verified by use of administrative controls. Normally locked or sealed-closed penetrations may be opened intermittently under administrative control.

3.7 - LIMITING CONDITIONS FOR OPERATION

N. SECONDARY CONTAINMENT INTEGRITY

SECONDARY CONTAINMENT INTEGRITY shall be maintained.

APPLICABILITY:

OPERATIONAL MODE(s) 1, 2, 3 and *.

ACTION:

1. Without SECONDARY CONTAINMENT INTEGRITY in OPERATIONAL MODES(s) 1, 2 or 3, restore SECONDARY CONTAINMENT INTEGRITY within 4 hours or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

2. Without SECONDARY CONTAINMENT INTEGRITY in OPERATIONAL MODE *, suspend handling of irradiated fuel in the secondary containment, CORE ALTERATION(s), and operations with a potential for draining the reactor vessel. The provisions of Specification 3.0.C are not applicable.

4.7 - SURVEILLANCE REQUIREMENTS

N. SECONDARY CONTAINMENT INTEGRITY

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   b. All secondary (a) containment penetrations not capable of being closed by OPERABLE secondary containment automatic isolation dampers and required to be closed during accident conditions are closed by valves, blind flanges, or deactivated automatic dampers secured in position.

3. At least once per 18 months by operating one standby gas treatment subsystem at a flow rate ≤4000 cfm for one hour and maintaining ≥0.25 inches of vacuum water gauge in the secondary containment.

'INSERT'

* When handling irradiated fuel in the secondary containment, during CORE ALTERATION(s), and operations with a potential for draining the reactor vessel.

INSERT

a Valves and blind flanges in high-radiation areas may be verified by use of administrative controls. Normally locked or sealed-closed penetrations may be opened intermittently under administrative control.

3.7 - LIMITING CONDITIONS FOR OPERATION

P. Standby Gas Treatment System

Two independent standby gas treatment subsystems shall be OPERABLE~~, each with an OPERABLE diesel generator power source~~.

APPLICABILITY:

OPERATIONAL MODE(s) 1, 2, 3 and *.

ACTION:

1. With one standby gas treatment subsystem inoperable, restore the inoperable subsystem to OPERABLE status within 7 days, or:

   a. In OPERATIONAL MODE(s) 1,2 or 3, be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

   b. In OPERATIONAL MODE *, suspend handling of irradiated fuel in the secondary containment, CORE ALTERATION(s), and operations with a potential for draining the reactor vessel. The provisions of Specification 3.0.C are not applicable.

2. With one standby gas treatment subsystem inoperable due to an inoperable diesel generator power source and the other standby gas treatment subsystem otherwise inoperable, restore either the inoperable diesel generator power source or the

'Deleted'

4.7 - SURVEILLANCE REQUIREMENTS

P. Standby Gas Treatment System

Each standby gas treatment subsystem shall be demonstrated OPERABLE:

1. At least once per 31 days by initiating, from the control room, flow through the HEPA filters and charcoal adsorbers and verifying that the subsystem operates for at least 10 hours with the heaters operating.

2. 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 subsystem by:

   a. Verifying that the subsystem satisfies the in-place penetration and bypass leakage testing acceptance criteria of <1% 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 4000 cfm ±10%.

   b. 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 ASTM-D-3803-89, for a methyl iodide penetration of <10%, when tested at 30°C and 70% relative humidity; and

* When handling irradiated fuel in the secondary containment, during CORE ALTERATION(s), and operations with a potential for draining the reactor vessel.

.7 - LIMITING CONDITIONS FOR OPERATION

otherwise inoperable subsystem to OPERABLE status within 72 hours, or:

a. In OPERATIONAL MODE(s) 1,2 or 3, be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

b. In OPERATIONAL MODE *, suspend handling of irradiated fuel in the secondary containment, CORE ALTERATION(s), and operations with a potential for draining the reactor vessel. The provisions of Specification 3.0.C are not applicable.

3. With both standby gas treatment subsystems inoperable due to inoperable diesel generator power sources, restore at least one of the inoperable diesel generator power sources to OPERABLE status within 72 hours, or:

a. In OPERATIONAL MODE(s) 1,2 or 3, be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

b. In OPERATIONAL MODE *, suspend handling of irradiated fuel in the secondary containment, CORE ALTERATION(s), and operations with a potential for draining the reactor vessel. The provisions of Specification 3.0.C are not applicable.

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4.7 - SURVEILLANCE REQUIREMENTS

c. Verifying a subsystem flow rate of 4000 cfm ±10% during system operation when tested in accordance with ANSI N510-1980.

3. After every 1440 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 ASTM-D-3803-89, for a methyl iodide penetration of <10%, when tested at 30°C and 70% relative humidity.

4. At least once per 18 months by:

a. Verifying that the pressure drop across the combined HEPA filters and charcoal adsorber banks is <6 inches water gauge while operating the filter train at a flow rate of 4000 cfm ±10%.

b. Verifying that the filter train starts and isolation dampers open on each of the following test signals:

1) Manual initiation from the control room, and

2) Simulated automatic initiation signal.

c. Verifying that the heaters dissipate 30 ±3 kw when tested in accordance with ANSI N510-1980. 9 This reading shall include the appropriate correction for variations from 480 volts at the bus.

* When handling irradiated fuel in the secondary containment, during CORE ALTERATION(s), and operations with a potential for draining the reactor vessel.

3.7 - LIMITING CONDITIONS FOR OPERATION

P. Standby Gas Treatment System

Two independent standby gas treatment subsystems shall be OPERABLE~~, each with an OPERABLE diesel generator power source~~.

APPLICABILITY:

OPERATIONAL MODE(s) 1, 2, 3 and *.

ACTION:

1. With one standby gas treatment subsystem inoperable, restore the inoperable subsystem to OPERABLE status within 7 days, or:

   a. In OPERATIONAL MODE(s) 1,2 or 3, be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

   b. In OPERATIONAL MODE *, suspend handling of irradiated fuel in the secondary containment, CORE ALTERATION(s), and operations with a potential for draining the reactor vessel. The provisions of Specification 3.0.C are not applicable.

2. With one standby gas treatment subsystem inoperable due to an inoperable diesel generator power source and the other standby gas treatment subsystem otherwise inoperable, restore either the inoperable diesel generator power source or the

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2

4.7 - SURVEILLANCE REQUIREMENTS

P. Standby Gas Treatment System

2

Each standby gas treatment subsystem shall be demonstrated OPERABLE:

1. At least once per 31 days by initiating, from the control room, flow through the HEPA filters and charcoal adsorbers and verifying that the subsystem operates for at least 10 hours with the heaters operating.

2. 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 subsystem by:

   a. Verifying that the subsystem satisfies the in-place penetration and bypass leakage testing acceptance criteria of <1% 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 4000 cfm ±10%.

   b. 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 ASTM-D-3803-89, for a methyl iodide penetration of <10%, when tested at 30°C and 70% relative humidity; and

* When handling irradiated fuel in the secondary containment, during CORE ALTERATION(s), and operations with a potential for draining the reactor vessel.

3.7 - LIMITING CONDITIONS FOR OPERATION

otherwise inoperable subsystem to OPERABLE status within 72 hours, or:

a. In OPERATIONAL MODE(s) 1,2 or 3, be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

b. In OPERATIONAL MODE *, suspend handling of irradiated fuel in the secondary containment, CORE ALTERATION(s), and operations with a potential for draining the reactor vessel. The provisions of Specification 3.0.C are not applicable.

3. With both standby gas treatment subsystems inoperable due to inoperable diesel generator power sources, restore at least one of the inoperable diesel generator power sources to OPERABLE status within 72 hours, or:

a. In OPERATIONAL MODE(s) 1,2 or 3, be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

b. In OPERATIONAL MODE *, suspend handling of irradiated fuel in the secondary containment, CORE ALTERATION(s), and operations with a potential for draining the reactor vessel. The provisions of Specification 3.0.C are not applicable.

'Deleted'

4.7 - SURVEILLANCE REQUIREMENTS

c. Verifying a subsystem flow rate of 4000 cfm ±10% during system operation when tested in accordance with ANSI N510-1980.

3. After every 1440 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 ASTM-D-3803-89, for a methyl iodide penetration of <10%, when tested at 30°C and 70% relative humidity.

4. At least once per 18 months by:

a. Verifying that the pressure drop across the combined HEPA filters and charcoal adsorber banks is <6 inches water gauge while operating the filter train at a flow rate of 4000 cfm ±10%.

b. Verifying that the filter train starts and isolation dampers open on each of the following test signals:

1) Manual initiation from the control room, and

2) Simulated automatic initiation signal.

c. Verifying that the heaters dissipate 30 ±3 kw when tested in accordance with ANSI N510-19~~80~~ 89. This reading shall include the appropriate correction for variations from 480 volts at the bus.

* When handling irradiated fuel in the secondary containment, during CORE ALTERATION(s), and operations with a potential for draining the reactor vessel.

BASES

which temporarily increases the suppression chamber pressure and reduces the differential pressure. Only one direct suppression chamber to drywell differential pressure instrumentation CHANNEL is provided. However, any pair of the redundant drywell and suppression chamber pressure instrumentation CHANNEL(s) are sufficient to determine the differential pressure.

3/4.7.I ~~Primary Containment Nitrogen System~~ DELETED

The nitrogen system functions to maintain oxygen concentrations within the primary containment at or below the explosive levels. To ensure that a combustible gas mixture does not occur, oxygen concentration is kept below 4.0 volume percent. The system operates in conjunction with emergency operating procedures that are used to reduce primary containment pressure periodically during system operation. This combination results in a feed-and-bleed approach to maintaining hydrogen and/or oxygen concentrations below combustible levels. Sufficient liquid nitrogen is maintained to provide approximately a seven day supply to allow for establishing an additional nitrogen supply following a LOCA.

3/4.7.J Primary Containment Oxygen Concentration

All nuclear reactors must be designed to withstand events that generate hydrogen either due to the zirconium metal-water reaction in the core or due to radiolysis. The primary method to control hydrogen is to inert the primary containment. With the primary containment inerted, that is, oxygen concentration less than 4.0 volume percent, a combustible mixture cannot be present in the primary containment for any hydrogen concentration. The Design Basis Accident (DBA) loss-of-coolant accident (LOCA) analysis assumes that the primary containment is inerted when the DBA occurs. Thus, the hydrogen assumed to be released to the primary containment as a result of a metal-water reaction in the reactor core will not produce combustible gas mixtures in the primary containment.

The primary containment oxygen concentration must be within the specified limit when primary containment is inerted, except as allowed by the relaxations during startup and shutdown. The primary containment must be inert in OPERATIONAL MODE 1, since this is the condition with the highest probability of an event that could produce hydrogen. Inerting the primary containment is an operational problem because it prevents containment access without an appropriate breathing apparatus. Therefore, the primary containment is inerted as late as possible in the plant startup and de-inerted as soon as possible in the plant shutdown. As long as reactor power is below 15% of RATED THERMAL POWER, the potential for an event that generates significant hydrogen is low and the primary containment does not need to be inert. Furthermore, the probability of an event that generates hydrogen occurring within the first 24 hours of a reactor startup or within the last 24 hours before a shutdown is low enough that these windows, when the primary containment is not inerted, are also justified. The 24 hour time frame is a reasonable amount of time to allow plant personnel to perform inerting or de-inerting.

BASES

which temporarily increases the suppression chamber pressure and reduces the differential pressure. Only one direct suppression chamber to drywell differential pressure instrumentation CHANNEL is provided. However, any pair of the redundant drywell and suppression chamber pressure instrumentation CHANNEL(s) are sufficient to determine the differential pressure.

3/4.7.I ~~Primary Containment Nitrogen System~~ DELETED

~~The nitrogen system functions to maintain oxygen concentrations within the primary containment at or below the explosive levels. To ensure that a combustible gas mixture does not occur, oxygen concentration is kept below 4.0 volume percent. The system operates in conjunction with emergency operating procedures that are used to reduce primary containment pressure periodically during system operation. This combination results in a feed-and-bleed approach to maintaining hydrogen and/or oxygen concentrations below combustible levels. Sufficient liquid nitrogen is maintained to provide approximately a seven day supply to allow for establishing an additional nitrogen supply following a LOCA.~~

3/4.7.J Primary Containment Oxygen Concentration

All nuclear reactors must be designed to withstand events that generate hydrogen either due to the zirconium metal-water reaction in the core or due to radiolysis. The primary method to control hydrogen is to inert the primary containment. With the primary containment inerted, that is, oxygen concentration less than 4.0 volume percent, a combustible mixture cannot be present in the primary containment for any hydrogen concentration. The Design Basis Accident (DBA) loss-of-coolant accident (LOCA) analysis assumes that the primary containment is inerted when the DBA occurs. Thus, the hydrogen assumed to be released to the primary containment as a result of a metal-water reaction in the reactor core will not produce combustible gas mixtures in the primary containment.

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BASES

3/4.7.N SECONDARY CONTAINMENT INTEGRITY

The function of the secondary containment is to isolate and contain fission products that escape from primary containment following a Design Basis Accident (DBA), to confine the postulated release of radioactive material within the requirements of 10CFR Part 100, and to isolate and contain fission products that are released during certain operations that take place inside primary containment, when primary containment is not required to be OPERABLE, or that take place outside of primary containment. The reactor building and associated structures provide secondary containment during normal operation when the drywell is sealed and in service. At other times the drywell may be open and, when required, secondary containment integrity is specified. There are two principal accidents for which credit is taken for secondary containment OPERABILITY. These are a LOCA and fuel-handling accident inside secondary containment. The secondary containment performs no active function in response to each of these limiting events; however, its leak tightness is required to limit offsite radiation doses to below those required by 10CFR Part 100. Maintaining secondary containment OPERABLE ensures that the release of radioactive materials from the primary containment is restricted to those leakage paths and associated leakage rates assumed in the accident analysis and that fission products entrapped within the secondary containment structure will be treated prior to discharge to the environment. Establishing and maintaining a vacuum in the reactor building with the standby gas treatment system during testing, along with the surveillance of the doors, hatches, dampers and valves, is adequate to ensure that there are no violations of the integrity of the secondary containment. This surveillance is normally conducted during periods of calm winds (<5 mph), but may be conducted under higher wind conditions with appropriate application of correction factors.

INSERT

3/4.7.O Secondary Containment Automatic Isolation Dampers

The function of the secondary containment ventilation system automatic isolation dampers, in combination with other accident-mitigation systems, is to limit fission-product release during and following postulated Design Basis Accidents (DBA) such that offsite radiation exposures are maintained within the requirements of 10CFR Part 100. Secondary containment isolation ensures that fission products that escape from primary containment following a DBA, or which are released during certain operations when primary containment is not required, or take place outside primary containment, are maintained within applicable limits. The OPERABILITY requirements for the secondary containment ventilation system isolation dampers help ensure that adequate secondary containment leak tightness is maintained during and after an accident by minimizing potential paths to the environment.

3/4.7.P Standby Gas Treatment System

The standby gas treatment system (SBGT) is required to ensure that radioactive materials that leak from the primary containment into the secondary containment following a Design Basis Accident (DBA) are filtered and adsorbed prior to exhausting to the environment. This system reduces the potential releases of radioactive material, principally iodine, to within values specified in 10CFR Part 100.

INSERT

Valves and blind flanges located in high radiation areas may be verified by use of administrative controls. Allowing verification by administrative controls is considered acceptable, since access to these areas is typically restricted during MODES 1, 2, and 3 for ALARA reasons. Therefore, the probability of misalignment of these SCIVs, once they have been verified to be in the proper position, is low. Normally locked or sealed closed penetrations may be opened intermittently under administrative controls. These administrative controls consist of stationing a dedicated operator, who is in continuous communication with the control room, at the controls of the penetration. In this way, the penetration can be rapidly isolated when a valid secondary containment isolation signal is indicated.

BASES

3/4.7.N SECONDARY CONTAINMENT INTEGRITY

The function of the secondary containment is to isolate and contain fission products that escape from primary containment following a Design Basis Accident (DBA), to confine the postulated release of radioactive material within the requirements of 10CFR Part 100, and to isolate and contain fission products that are released during certain operations that take place inside primary containment, when primary containment is not required to be OPERABLE, or that take place outside of primary containment. The reactor building and associated structures provide secondary containment during normal operation when the drywell is sealed and in service. At other times the drywell may be open and, when required, secondary containment integrity is specified. There are two principal accidents for which credit is taken for secondary containment OPERABILITY. These are a LOCA and fuel-handling accident inside secondary containment. The secondary containment performs no active function in response to each of these limiting events; however, its leak tightness is required to limit offsite radiation doses to below those required by 10CFR Part 100. Maintaining secondary containment OPERABLE ensures that the release of radioactive materials from the primary containment is restricted to those leakage paths and associated leakage rates assumed in the accident analysis and that fission products entrapped within the secondary containment structure will be treated prior to discharge to the environment. Establishing and maintaining a vacuum in the reactor building with the standby gas treatment system during testing, along with the surveillance of the doors, hatches, dampers and valves, is adequate to ensure that there are no violations of the integrity of the secondary containment. This surveillance is normally conducted during periods of calm winds (<5 mph), but may be conducted under higher wind conditions with appropriate application of correction factors.

INSERT

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INSERT

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BASES

The standby gas treatment system is designed to filter and exhaust the reactor building atmosphere to the main chimney during secondary containment isolation conditions, with a minimum release of radioactive materials from the reactor building to the environment. One standby gas treatment fan is designed to automatically start upon secondary containment isolation and to maintain the reactor building pressure to approximately a negative ¼ inch water gauge pressure; all leakage should be in-leakage. Should the fan fail to start, the redundant alternate fan and filter subsystem is designed to start automatically.

The OPERABILITY of the standby gas treatment system reduces the potential release of radioactive material, principally iodine, following a design basis accident. The reduction in containment iodine inventory reduces the resulting site boundary radiation doses associated with containment leakage. The operation of this system and resultant iodine removal capacity are consistent with the assumptions used in the LOCA analyses. Periodic operation of the system with the heaters is sufficient to reduce the buildup of moisture on the adsorbers and HEPA filters.

Since the standby gas treatment subsystems are shared by both units, one subsystem is powered by the unit diesel generator power source of each unit. ~~This unique arrangement requires that special allowed out-of-service times be provided for the combinations of subsystem and diesel generator power source inoperability that may occur.~~ For example, if conducting the alternate offsite power source cross-tie surveillance were to require the inoperability of both unit diesel generator power sources, neither of the standby gas treatment subsystems would have an OPERABLE diesel generator power source and the appropriate ACTION would have to be entered.

'INSERT'

INSERT
[SBGT Bases]

The emergency power supply OPERABILITY requirements for the standby gas treatment system are addressed within Specification 3.9.A, Actions.

BASES

The standby gas treatment system is designed to filter and exhaust the reactor building atmosphere to the main chimney during secondary containment isolation conditions, with a minimum release of radioactive materials from the reactor building to the environment. One standby gas treatment fan is designed to automatically start upon secondary containment isolation and to maintain the reactor building pressure to approximately a negative ¼ inch water gauge pressure; all leakage should be in-leakage. Should the fan fail to start, the redundant alternate fan and filter subsystem is designed to start automatically.

The OPERABILITY of the standby gas treatment system reduces the potential release of radioactive material, principally iodine, following a design basis accident. The reduction in containment iodine inventory reduces the resulting site boundary radiation doses associated with containment leakage. The operation of this system and resultant iodine removal capacity are consistent with the assumptions used in the LOCA analyses. Periodic operation of the system with the heaters is sufficient to reduce the buildup of moisture on the adsorbers and HEPA filters.

Since the standby gas treatment subsystems are shared by both units, one subsystem is powered by the unit diesel generator power source of each unit. ~~This unique arrangement requires that special allowed out-of-service times be provided for the combinations of subsystem and diesel generator power source inoperability that may occur.~~ For example, if conducting the alternate offsite power source cross-tie surveillance were to require the inoperability of both unit diesel generator power sources, neither of the standby gas treatment subsystems would have an OPERABLE diesel generator power source and the appropriate ACTION would have to be entered.

'INSERT'

INSERT
[SBGT Bases]

The emergency power supply OPERABILITY requirements for the standby gas treatment system are addressed within Specification 3.9.A, Actions.

3.8 - LIMITING CONDITIONS FOR OPERATION

A. Containment Cooling Service Water System

At least the following independent containment cooling service water (CCSW) subsystems, with each subsystem comprised of:

1. Two OPERABLE CCSW pumps, and

2. An OPERABLE flow path capable of taking suction from the ultimate heat sink and transferring the water:

   a. Through one LPCI heat exchanger[a], and separately,

   b. To the associated safety related equipment,

shall be OPERABLE:

1. In OPERATIONAL MODE(s) 1, 2 and 3, two subsystems.

2. In OPERATIONAL MODE(s) 4, 5 and *, the subsystem(s) associated with subsystems/loops and components required OPERABLE by Specification 3.8.D.

APPLICABILITY:

OPERATIONAL MODE(s) 1, 2, 3, 4, 5 and *.

4.8 - SURVEILLANCE REQUIREMENTS

A. Containment Cooling Service Water System

Each of the required CCSW subsystems shall be demonstrated OPERABLE 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.

---

a The LPCI heat exchanger is not required to support operation of the CREFS.

* When handling irradiated fuel in the secondary containment, during CORE ALTERATION(s), and operations with a potential for draining the reactor vessel.

3.8 - LIMITING CONDITIONS FOR OPERATION

A. Residual Heat Removal Service Water System

At least the following independent residual heat removal service water (RHRSW) subsystems, with each subsystem comprised of:

1. Two OPERABLE RHRSW pumps, and

2. An OPERABLE flow path capable of taking suction from the ultimate heat sink and transferring the water:

    a. Through one RHR heat exchanger, and separately,

    b. To the associated safety related equipment,

shall be OPERABLE:

1. In OPERATIONAL MODE(s) 1, 2 and 3, two subsystems.

2. In OPERATIONAL MODE(s) 4, 5 and * the subsystem(s) associated with subsystems/loops and components required OPERABLE by Specifications 3.6.O, 3.6.P, 3.8.D, 3.10.K and 3.10.L.

APPLICABILITY:

OPERATIONAL MODE(s) 1, 2, 3, 4, 5 and *.

4.8 - SURVEILLANCE REQUIREMENTS

A. Residual Heat Removal Service Water System

Each of the required RHRSW subsystems shall be demonstrated OPERABLE at least once per 31 days by verifying that each valve, manual, or power operated or automatic, in the flow path that is not locked, sealed or otherwise secured in position, is in its correct position.

* When handling irradiated fuel in the secondary containment, during CORE ALTERATION(s), and operations with a potential for draining the reactor vessel.

3.8 - LIMITING CONDITIONS FOR OPERATION

2. In OPERATIONAL MODE 4, 5 or * with the CCSW subsystem which is associated with the safety related equipment required OPERABLE by Specification 3.8.D inoperable, declare the associated safety related equipment inoperable and take the ACTION required by Specification 3.8.D.

4.8 - SURVEILLANCE REQUIREMENTS

* When handling irradiated fuel in the secondary containment, during CORE ALTERATION(s), and operations with a potential for draining the reactor vessel.

Ventilation

3.8 - LIMITING CONDITIONS FOR OPERATION

D. Control Room Emergency ~~Filtration~~ System

ventilation

The control room emergency ~~filtration~~ system shall be OPERABLE, with the system comprised of an OPERABLE control room emergency filtration system and an OPERABLE refrigeration control unit (RCU).

APPLICABILITY:

OPERATIONAL MODE(s) 1, 2, 3, and *.

ACTION:

'INSERT'

1. In OPERATIONAL MODE(s) 1, 2 or 3 with the control room emergency filtration system inoperable, restore the inoperable system to OPERABLE status within 14 days or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.
2. In OPERATIONAL MODE *, with the control room emergency filtration system or the RCU inoperable, immediately suspend CORE ALTERATION(s), handling of irradiated fuel in the secondary containment and operations with a potential for draining the reactor vessel.
3. The provisions of Specification 3.0.C are not applicable in OPERATIONAL MODE *.

4.8 - SURVEILLANCE REQUIREMENTS

D. Control Room Emergency ~~Filtration~~ System

The control room emergency ~~filtration~~ system shall be demonstrated OPERABLE:

1. At least once per ~~12 hours~~ 18 months by verifying that ~~the control room air temperature is ≤95°F.~~ RCU has the capability to remove the required heat load.
2. At least once per 31 days by initiating, from the control room, flow through the HEPA filters and charcoal adsorbers and verifying that the system operates for at least 10 hours with the heaters operating.
3. 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:
   a. Verifying that the system satisfies the in-place penetration and bypass leakage testing acceptance criteria of <0.05% 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 scfm ±10%.
   b. 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

* When handling irradiated fuel in the secondary containment, during CORE ALTERATION(s), and operations with a potential for draining the reactor vessel.

INSERT
[CREFS]

1. In OPERATIONAL MODE(s) 1, 2 or 3:

   a. With the control room emergency filtration system inoperable, restore the inoperable system to OPERABLE status within 7 days or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

   b. With the refrigeration control unit (RCU) inoperable, restore the inoperable RCU to OPERABLE status within 30 days or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

3.8 - LIMITING CONDITIONS FOR OPERATION

D. Control Room Emergency ~~Filtration~~ Ventilation System

The control room emergency ~~filtration~~ ventilation system shall be OPERABLE, with the system comprised of an OPERABLE control room emergency filtration system and an OPERABLE refrigeration control unit (RCU).

APPLICABILITY:

OPERATIONAL MODE(s) 1, 2, 3, and *.

ACTION:

'INSERT'

1. In OPERATIONAL MODE(s) 1, 2 or 3 with the control room emergency filtration system inoperable, restore the inoperable system to OPERABLE status within 14 days or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

2. In OPERATIONAL MODE *, with the control room emergency filtration system or the RCU inoperable, immediately suspend CORE ALTERATION(s), handling of irradiated fuel in the secondary containment and operations with a potential for draining the reactor vessel.

3. The provisions of Specification 3.0.C are not applicable in OPERATIONAL MODE *.

4.8 - SURVEILLANCE REQUIREMENTS

D. Control Room Emergency ~~Filtration~~ Ventilation System

The control room emergency ~~filtration~~ ventilation system shall be demonstrated OPERABLE:

1. At least once per ~~12 hours~~ 18 months by verifying that the ~~control room air temperature is ≤95°F~~ RCU has the capability to remove the required heat load.

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

3. 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:

   a. Verifying that the system satisfies the in-place penetration and bypass leakage testing acceptance criteria of <0.05% 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 scfm ±10%.

   b. 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

* When handling irradiated fuel in the secondary containment, during CORE ALTERATION(s), and operations with a potential for draining the reactor vessel.

INSERT
[CREFS]

1. In OPERATIONAL MODE(s) 1, 2 or 3:

   a. With the control room emergency filtration system inoperable, restore the inoperable system to OPERABLE status within 7 days or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

   b. With the refrigeration control unit (RCU) inoperable, restore the inoperable RCU to OPERABLE status within 30 days or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

3.8 - LIMITING CONDITIONS FOR OPERATION

4.8 - SURVEILLANCE REQUIREMENTS

"Accept" line, testing of that type of snubber may be terminated. If the point plotted falls above the "Accept" line, testing must continue until the point falls on or below the "Accept" line or all the snubbers of that type have been tested.

The representative sample selected for the functional test sample plans shall be randomly selected from the snubbers of each type and reviewed before beginning the testing. The review shall ensure as far as practical that they are representative of the various configurations, operating environments, range of size, and capacity of snubbers of each type.

Snubbers placed in the same location as snubbers which failed the previous functional test shall be retested at the time of the next functional test but shall not be included in the sample plan, and failure of this functional test shall not be the sole cause for increasing the sample size under the sample plan. If during testing, additional sampling is required due to failure of only on type of snubber, the functional testing results shall be reviewed at the time to determine if additional samples should be limited to the type of snubber which has failed the functional testing.

one

6. Functional Test Acceptance Criteria

The snubber functional test shall verify that:

a. Activation (restraining action) is achieved within the specified range in both tension and compression;

3.8 - LIMITING CONDITIONS FOR OPERATION

4.8 - SURVEILLANCE REQUIREMENTS

the snubber is required to be OPERABLE. The parts replacements shall be documented and the documentation shall be retained in accordance with Specification 6.5.B.

3.8 - LIMITING CONDITIONS FOR OPERATION

4.8 - SURVEILLANCE REQUIREMENTS

the snubber is required to be OPERABLE. The parts replacements shall be documented and the documentation shall be retained in accordance with Specification 6.5.B.

3.8 - LIMITING CONDITIONS FOR OPERATION

I. Main Condenser Offgas Activity

The release rate of the sum of the activities of the noble gases measured prior to the offgas holdup line shall be limited to ≤100 μCi/sec/MWt, after 30 minutes decay.

APPLICABILITY:

OPERATIONAL MODE(s) 1, 2[a] and 3[a].

ACTION:

With the release rate of the sum of the activities of the noble gases at the main condenser air ejector effluent (as measured prior to the offgas holdup line) >100 μCi/sec/MWt, after 30 minutes decay, restore the release rate to within its limit within 72 hours or be in at least STARTUP with the main steam isolation valves closed within the next 8 hours.

4.8 - SURVEILLANCE REQUIREMENTS

I. Main Condenser Offgas Activity

1. The release rate of noble gases from the main condenser air ejector shall be continuously monitored in accordance with the ODCM.

2. The release rate of the sum of the activities from noble gases from the main condenser air ejector shall be determined to be within the limits of Specification 3.8.I at the following frequencies[b] by performing an isotopic analysis of a representative sample of gases taken at the recombiner outlet, or the air ejector outlet, if the recombiner is bypassed:

   a. At least once per 31 days, and

   b. Within 4 hours following the determination of an increase, as indicated by the air ejector noble gas monitor, of >50%, after ~~factoring out increases~~ due to changes in THERMAL POWER level, in the nominal steady state fission gas release from the primary coolant.

a When the main condenser air ejector is in operation.

b The provisions of Specification 4.0.D are not applicable.

3.8 - LIMITING CONDITIONS FOR OPERATION

I. Main Condenser Offgas Activity

The release rate of the sum of the activities of the noble gases measured prior to the offgas holdup line shall be limited to ≤100 μCi/sec/MWt, after 30 minutes decay.

APPLICABILITY:

OPERATIONAL MODE(s) 1, 2[a] and 3[a].

ACTION:

With the release rate of the sum of the activities of the noble gases in the main condenser air ejector effluent (as measured prior to the offgas holdup line) >100 μCi/sec/MWt, after 30 minutes decay, restore the release rate to within its limit within 72 hours or be in at least STARTUP with the main steam isolation valves closed within the next 8 hours.

4.8 - SURVEILLANCE REQUIREMENTS

I. Main Condenser Offgas Activity

1. The release rate of noble gases from the main condenser air ejector shall be continuously monitored in accordance with the ODCM.

2. The release rate of the sum of the activities from noble gases from the main condenser air ejector shall be determined to be within the limits of Specification 3.8.I at the following frequencies[b] by performing an isotopic analysis of a representative sample of gases taken at the recombiner outlet, or the air ejector outlet, if the recombiner is bypassed:

   a. At least once per 31 days, and

   b. Within 4 hours following the determination of an increase, as indicated by the air ejector noble gas monitor, of >50%, after factoring out increases due to changes in THERMAL POWER level, in the nominal steady state fission gas release from the primary coolant. ℓ ℓ

a When the main condenser air ejector is in operation.

b The provisions of Specification 4.0.D are not applicable.

3.8 LIMITING CONDITIONS FOR OPERATION

J. Safe Shutdown Makeup Pump

The Safe Shutdown Makeup Pump (SSMP) shall be OPERABLE.

APPLICABILITY:

OPERATIONAL MODE(s) 1, 2 and 3 with reactor steam dome pressure greater than 150 psig.

ACTION:

1. With the SSMP system inoperable, restore the inoperable SSMP system to OPERABLE status within 67 days, or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

4.8 - SURVEILLANCE REQUIREMENTS

J. Safe Shutdown Makeup Pump

The SSMP system shall be demonstrated OPERABLE:

1. At least once per 31 days by:

   a. 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. Verifying that the pump flow controller is in the correct position.

2. At least once per 92 days by verifying that the SSMP develops a flow of greater than or equal to 400 gpm ~~in the test flow path with~~ against a system head corresponding to reactor vessel ~~operating~~ pressure of greater than ~~150~~ 1120 psig.

BASES

3/4.8.A Containment Cooling Service Water System

The containment cooling service water system, with the ultimate heat sink, provides sufficient cooling capacity for continued operation of the containment cooling system and of other safety-related equipment (e.g., CCSW keep-fill, the control room emergency ventilation system refrigeration units), during normal and accident conditions. The redundant cooling capacity of the system, assuming a single failure, is consistent with the assumptions used in the safety analysis to keep the accident conditions within acceptable limits. Since only one of the four pumps is required to provide the necessary cooling capacity, a thirty day repair period is allowed for one pump out of service. OPERABILITY of this system is also dependent upon special measures for protection from flooding in the condenser pit area.

two

3/4.8.B Diesel Generator Cooling Water System

The diesel generator cooling water system, with the ultimate heat sink, provides sufficient cooling capacity for continued operation of the diesel generators during normal and accident conditions. The cooling capacity of the system is consistent with the assumptions used in the safety analysis to keep the accident conditions within acceptable limits. OPERABILITY of this system is also dependent upon special measures for protection from flooding in the condenser pit area.

3/4.8.C Ultimate Heat Sink

The canals provide an ultimate heat sink with sufficient cooling capacity to either provide normal cooldown of the units, or to mitigate the effects of accident conditions within acceptable limits for one unit while conducting a normal cooldown on the other unit.

Ventilation

3/4.8.D Control Room Emergency Filtration System

The control room emergency filtration system maintains habitable conditions for operations personnel during and following all design basis accident conditions. This system, in conjunction with control room design, is based on limiting the radiation exposure to personnel occupying the room to five rem or less whole body, or its equivalent.

The frequency of tests and sample analysis is necessary to show that the HEPA filters and charcoal adsorbers can perform as evaluated. The control room emergency filtration system in-place testing procedures are established utilizing applicable sections of ANSI N510-1980 standard. Operation of the system with the heaters OPERABLE for ten hours a month is sufficient to reduce the buildup of moisture on the adsorbers and HEPA filters. The charcoal adsorber efficiency test procedures allow for the removal of one representative sample cartridge and testing in accordance with the guidelines of ASTM-D-3803-89. The sample is at least two inches in diameter and has a length equivalent to the thickness of the bed. If the iodine removal efficiency test results are

BASES

3/4.8.A Residual Heat Removal Service Water System

The residual heat removal service water system, with the ultimate heat sink, provides sufficient cooling capacity for continued operation of the residual heat removal system and of other safety-related equipment, e.g., RHRSW vault coolers and the control room emergency ventilation system refrigeration units, during normal and accident conditions. The redundant cooling capacity of the system, assuming a single failure, is consistent with the assumptions used in the safety analysis to keep the accident conditions within acceptable limits. Since only one of the four pumps is required to provide the necessary cooling capacity, a thirty day repair period is allowed for one pump out of service. OPERABILITY of this system is also dependent upon special measures for protection from flooding in the condenser pit area.

3/4.8.B Diesel Generator Cooling Water System

The diesel generator cooling water system, with the ultimate heat sink, provides sufficient cooling capacity for continued operation of the diesel generators during normal and accident conditions. The cooling capacity of the system is consistent with the assumptions used in the safety analysis to keep the accident conditions within acceptable limits. OPERABILITY of this system is also dependent upon special measures for protection from flooding in the condenser pit area.

3/4.8.C Ultimate Heat Sink

The Mississippi River provides an ultimate heat sink with sufficient cooling capacity to either provide normal cooldown of the units, or to mitigate the effects of accident conditions within acceptable limits for one unit while conducting a normal cooldown on the other unit.

3/4.8.D Control Room Emergency ~~Filtration~~ Ventilation System

The control room emergency filtration system maintains habitable conditions for operations personnel during and following all design basis accident conditions. This system, in conjunction with control room design, is based on limiting the radiation exposure to personnel occupying the room to five rem or less whole body, or its equivalent.

The frequency of tests and sample analysis is necessary to show that the HEPA filters and charcoal adsorbers can perform as evaluated. The control room emergency filtration system in-place testing procedures are established utilizing applicable sections of ANSI N510-1980 standard. Operation of the system with the heaters OPERABLE for ten hours a month is sufficient to reduce the buildup of moisture on the adsorbers and HEPA filters. The charcoal adsorber efficiency test procedures allow for the removal of one representative sample cartridge and testing in accordance with the guidelines of ASTM-D-3803-89. The sample is at least two inches in diameter and has a length equivalent to the thickness of the bed. If the iodine removal efficiency test results are

BASES

unacceptable, all adsorbent in the system is replaced. HEPA filter particulate removal efficiency is verified to be at least 99% by in-place testing with a DOP testing medium.

INSERT

3/4.8.E Flood Protection

Flood protection measures are provided to protect the systems and equipment necessary for safe shutdown during high water conditions. The equipment necessary to implement the appropriate measures, as detailed in plant procedures, is required to be available, but not necessarily onsite, to implement the procedures in a timely manner. The selected water levels are based on providing timely protection from the design basis flood of the river.

3/4.8.F Snubbers

Mechanical snubbers are provided to ensure that the structural integrity of the reactor coolant system and all other safety-related systems is maintained during and following a seismic event or other event initiating dynamic loads. Snubbers are classified and grouped by design, manufacturer and accessibility. A list of individual snubbers with information of snubber location, classification or group, and system affected is maintained at the plant. The accessibility of each snubber is determined and documented for each snubber. The determination is based upon the existing radiation levels and the expected time to perform a visual inspection in each snubber location as well as other factors associated with accessibility during plant operation (e.g., temperature, atmosphere, location, etc.), and the recommendations of Regulatory Guides 8.8 and 8.10.

The visual inspection frequency is based upon maintaining a constant level of snubber protection to the systems. Therefore, the required inspection interval varies with the number of unacceptable snubbers found during the previous inspection, the total population or category size for each snubber type, and the previous inspection interval. A snubber is considered unacceptable if it fails to satisfy the acceptance criteria of the visual inspection. Snubbers may be categorized, based upon their accessibility during power operation, as accessible or inaccessible. These categories may be examined separately or jointly as determined and documented prior to the inspections. The categorization is used as the basis for determining the next inspection interval for that category.

If a review and evaluation can not justify continued operation with an unacceptable snubber, the snubber is declared inoperable and the applicable action taken. To determine the next surveillance interval, the unacceptable snubber may be reclassified as acceptable if it can be demonstrated that the snubber is OPERABLE in its as-found condition by the performance of a functional test. The next visual inspection interval may be twice, the same, or reduced by as much as two-thirds of the previous inspection interval, depending on the number of unacceptable snubbers found in proportion to the size of the population or category for each type of snubber included in the previous inspection. The inspection interval may be as long as 48 months and the provisions of Specification 4.0.B may be applied.

INSERT
[CREFS]

The control room refrigeration control unit (RCU) provides conditioned air for personnel comfort, safety and equipment reliability. The testing of the control room RCU system verifies that the heat-removal capability of the system is sufficient to remove sufficient heat load from the control room such that the control room air temperature is ≤ 95 °F. The test frequency is appropriate since significant degradation of the control room RCU system is not expected over this time period.

unacceptable, all adsorbent in the system is replaced. HEPA filter particulate removal efficiency is verified to be at least 99% by in-place testing with a DOP testing medium.

INSERT

3/4.8.E Flood Protection

Flood protection measures are provided to protect the systems and equipment necessary for safe shutdown during high water conditions. The equipment necessary to implement the appropriate measures, as detailed in plant procedures, is required to be available, but not necessarily onsite, to implement the procedures in a timely manner. The selected water levels are based on providing timely protection from the design basis flood of the river.

3/4.8.F Snubbers

Mechanical snubbers are provided to ensure that the structural integrity of the reactor coolant system and all other safety-related systems is maintained during and following a seismic event or other event initiating dynamic loads. Snubbers are classified and grouped by design, manufacturer and accessibility. A list of individual snubbers with information of snubber location, classification or group, and system affected is maintained at the plant. The accessibility of each snubber is determined and documented for each snubber. The determination is based upon the existing radiation levels and the expected time to perform a visual inspection in each snubber location as well as other factors associated with accessibility during plant operation (e.g., temperature, atmosphere, location, etc.), and the recommendations of Regulatory Guides 8.8 and 8.10.

The visual inspection frequency is based upon maintaining a constant level of snubber protection to the systems. Therefore, the required inspection interval varies with the number of unacceptable snubbers found during the previous inspection, the total population or category size for each snubber type, and the previous inspection interval. A snubber is considered unacceptable if it fails to satisfy the acceptance criteria of the visual inspection. Snubbers may be categorized, based upon their accessibility during power operation, as accessible or inaccessible. These categories may be examined separately or jointly as determined and documented prior to the inspections. The categorization is used as the basis for determining the next inspection interval for that category.

If a review and evaluation can not justify continued operation with an unacceptable snubber, the snubber is declared inoperable and the applicable action taken. To determine the next surveillance interval, the unacceptable snubber may be reclassified as acceptable if it can be demonstrated that the snubber is OPERABLE in its as-found condition by the performance of a functional test. The next visual inspection interval may be twice, the same, or reduced by as much as two-thirds of the previous inspection interval, depending on the number of unacceptable snubbers found in proportion to the size of the population or category for each type of snubber included in the previous inspection. The inspection interval may be as long as 48 months and the provisions of Specification 4.0.B may be applied.

INSERT
[CREFS]

The control room refrigeration control unit (RCU) provides conditioned air for personnel comfort, safety and equipment reliability. The testing of the control room RCU system verifies that the heat-removal capability of the system is sufficient to remove sufficient heat load from the control room such that the control room air temperature is ≤ 95 °F. The test frequency is appropriate since significant degradation of the control room RCU system is not expected over this time period.

within a shielded mechanism, i.e., sealed sources within radiation monitoring or boron measuring devices, are considered to be stored and need not be tested unless they are removed from the shielded mechanism.

3/4.8.H Explosive Gas Mixture

This specification is provided to ensure that the concentration of potentially explosive gas mixtures contained in the offgas holdup system is maintained below the flammability limits of hydrogen and oxygen. Maintaining the concentration of hydrogen and oxygen below their flammability limits provides assurance that the releases of radioactive materials will be controlled in conformance with the requirements of General Design Criterion 60 of Appendix A to 10CFR Part 50.

3/4.8.I Main Condenser Offgas Activity

Restricting the gross radioactivity rate of noble gases from the main condenser provides reasonable assurance that the total body exposure to an individual at the exclusion area boundary will not exceed a small fraction of the limits of 10CFR Part 100 in the event this effluent is inadvertently discharged directly to the environment without treatment. This specification implements the requirements of General Design Criteria 60 and 64 of Appendix A to 10CFR Part 50. INSERT

3/4.8.J Liquid Holdup Tanks

Restricting the quantity of radioactive material contained in the specified tanks provides assurance that in the event of an uncontrolled release of the tanks' contents, the resulting concentrations would be less than the limits of 10CFR Part 20, Appendix B, Table II, Column 2, at the nearest potable water supply and the nearest surface water supply in an unrestricted area. Recirculation of the tank contents for the purpose of reducing the radioactive content is not considered to be an addition of radioactive material to the tank.

BASES

within a shielded mechanism, i.e., sealed sources within radiation monitoring or boron measuring devices, are considered to be stored and need not be tested unless they are removed from the shielded mechanism.

3/4.8.H Explosive Gas Mixture

This specification is provided to ensure that the concentration of potentially explosive gas mixtures contained in the offgas holdup system is maintained below the flammability limits of hydrogen and oxygen. Maintaining the concentration of hydrogen and oxygen below their flammability limits provides assurance that the releases of radioactive materials will be controlled in conformance with the requirements of General Design Criterion 60 of Appendix A to 10CFR Part 50.

3/4.8.I Main Condenser Offgas Activity

Restricting the gross radioactivity rate of noble gases from the main condenser provides reasonable assurance that the total body exposure to an individual at the exclusion area boundary will not exceed a small fraction of the limits of 10CFR Part 100 in the event this effluent is inadvertently discharged directly to the environment without treatment. This specification implements the requirements of General Design Criteria 60 and 64 of Appendix A to 10CFR Part 50.

'INSERT'

3/4.8.J Safe Shutdown Makeup Pump System (SSMP)

The SSMP system provides a common backup to the Unit 1 and 2 RCIC systems to satisfy the requirements of 10 CFR 50, Appendix R, Section III.G, "Fire Protection of Safe Shutdown Capability." The system bypasses fire zones which could theoretically disable the RCIC system.

In the event that the reactor vessel becomes isolated, and the feedwater supply becomes unavailable, makeup can be provided by manually initiating the SSMP system to supply demineralized makeup water from the CCST or as an alternate source, makeup water from the fire header. The flow rate of the SSMP system is approximately equal to the reactor water boil-off rate 15 minutes after shutdown.

The SSMP system is required to be OPERABLE when either Unit 1 or Unit 2 is in OPERATIONAL MODE(s) 1, 2 or 3 with reactor steam dome pressure greater than 150 psig. With the SSMP system inoperable, a 67-day allowable out-of-service (AOT) is provided to restore the inoperable system to OPERABLE status before the Unit(s) must be shut down. (Reference: Fire Protection Plan Documentation Package (FPPDP), "Fire Protection Reports," Volume 2, Tab 4, Safe Shutdown Analysis.)

The surveillance requirements provide adequate assurance that the SSMP system will be OPERABLE when required. A design flow test can be performed during plant operation using a full flow test return line to the CCST.

INSERT

[Offgas Release Rates Bases]

The release rates are determined at a more expeditious frequency following the determination of an increase of greater than 50%, as indicated by the air ejector noble gas monitor, after factoring out increases due to changes in THERMAL POWER level and off-gas flow in the nominal steady-state fission gas release from the primary coolant.

3.9 - LIMITING CONDITIONS FOR OPERATION

A. A.C. Sources - Operating

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

1. Two physically independent circuits between the offsite transmission network and the onsite Class 1E distribution system, and
2. Two separate and independent diesel generators, each with:
   a. A separate fuel oil day tank containing ≥205 gallons of available fuel,
   b. A separate bulk fuel storage system containing ≥10,000 gallons of available fuel, and
   c. A separate fuel oil transfer pump.

APPLICABILITY:

OPERATIONAL MODE(s) 1, 2, and 3.

ACTION:

1. With one of the above required offsite circuit power sources inoperable:
   a. Demonstrate the OPERABILITY of the remaining offsite circuit by performing Surveillance Requirement 4.9.A.1.a within 1 hour and at least once per 8 hours thereafter.

4.9 - SURVEILLANCE REQUIREMENTS

A. A.C Sources - Operating

1. Each of the required independent circuits between the offsite transmission network and the onsite Class 1E distribution system shall be determined OPERABLE:
   a. At least once per 7 days by verifying correct breaker alignments and indicated power availability, and

      At least once per 18 months by manually transferring the power supply from the normal circuit to the alternate circuit.
   b. [INTENTIONALLY BLANK]
      This requirement is an open item to be addressed in the TSUP clean-up amendment
2. Each of the required diesel generators shall be demonstrated OPERABLE[a] ~~in accordance with the frequency specified in Table 4.9.A-1~~ at least once per 31 days by:
   a. Verifying the fuel levels in both the fuel oil day tank and the bulk fuel storage tank.
   b. Verifying the fuel transfer pump starts and transfers fuel from the bulk fuel storage system to the fuel oil day tank.

'INSERT'

a All planned diesel generator tests shall be conducted in accordance with the manufacturer's recommendations regarding engine prelube, leak detection and warmup procedures, and as applicable regarding loading and shutdown recommendations.

INSERT

All diesel generator starts may be preceded by an engine prelube period. All diesel generator starts that require loading may be preceded by an engine prelube period and followed by a warmup period prior to loading. Diesel generator loadings may include gradual loading as recommended by the manufacturer/vendor.

3.9 - LIMITING CONDITIONS FOR OPERATION

A. A.C. Sources - Operating

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

1. Two physically independent circuits between the offsite transmission network and the onsite Class 1E distribution system, and

2. Two separate and independent diesel generators, each with:

   a. A separate fuel oil day tank containing ≥205 gallons of available fuel,

   b. A separate bulk fuel storage system containing ≥10,000 gallons of available fuel, and

   c. A separate fuel oil transfer pump.

APPLICABILITY:

OPERATIONAL MODE(s) 1, 2, and 3.

ACTION:

1. With one of the above required offsite circuit power sources inoperable:

   a. Demonstrate the OPERABILITY of the remaining offsite circuit by performing Surveillance Requirement 4.9.A.1.a within 1 hour and at least once per 8 hours thereafter.

4.9 - SURVEILLANCE REQUIREMENTS

A. A.C Sources - Operating

1. Each of the required independent circuits between the offsite transmission network and the onsite Class 1E distribution system shall be determined OPERABLE:

   a. At least once per 7 days by verifying correct breaker alignments and indicated power availability, and

   At least once per 18 months by manually transferring the power supply from the normal circuit to the alternate circuit.

   b. ~~[INTENTIONALLY BLANK] This requirement is an open item to be addressed in the TSUP clean-up amendment~~

2. Each of the required diesel generators shall be demonstrated OPERABLE[a] ~~in accordance with the frequency specified in Table 4.9.A-1~~ by:

   a. Verifying the fuel levels in both the fuel oil day tank and the bulk fuel storage tank.

   b. Verifying the fuel transfer pump starts and transfers fuel from the bulk fuel storage system to the fuel oil day tank.

at least once per 31 days

'INSERT'

---

a ~~All planned diesel generator tests shall be conducted in accordance with the manufacturer's recommendations regarding engine prelube, leak detection and warmup procedures, and as applicable regarding loading and shutdown recommendations.~~

INSERT

All diesel generator starts may be preceded by an engine prelube period. All diesel generator starts that require loading may be preceded by an engine prelube period and followed by a warmup period prior to loading. Diesel generator loadings may include gradual loading as recommended by the manufacturer/vendor.

3.9 - LIMITING CONDITIONS FOR OPERATION

b. Demonstrate the OPERABILITY of each diesel generator by performing Surveillance Requirement 4.9.A.2.c for each diesel generator separately within 24 hours (if it has not been successfully tested within the past 24 hours) and within the subsequent 72 hours, and

b ~~c~~. Restore the inoperable offsite circuit to OPERABLE status within 7 days or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

2. With one of the above required diesel generator power sources inoperable:

   a. Demonstrate the OPERABILITY of the offsite circuit power sources by performing Surveillance Requirement 4.9.A.1.a within 1 hour and at least once per 8 hours thereafter.

   b. If the diesel generator is inoperable due to any cause other than an inoperable support system, an independently testable component, or preplanned ~~preventive~~ maintenance or testing, demonstrate the OPERABILITY of the remaining OPERABLE diesel generator by performing Surveillance Requirement 4.9.A.2.c within 24 hours[b] (if it has not been successfully tested within the past 24 hours) and within the subsequent 72 hours, and

4.9 - SURVEILLANCE REQUIREMENTS

c. Verifying[c] the diesel starts and accelerates to synchronous speed with generator voltage and frequency at 4160 ±420 volts and 60 ±1.2 Hz, respectively.

d. Verifying[e] ~~[c]~~ the diesel generator is synchronized, loaded to between 2470 and 2600 kW[d], in accordance with the manufacturer's/vendor's recommendations and operates with this load for ≥60 minutes.

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

f. Verifying the pressure in required starting air receiver tanks to be ≥220 psig.

3. Each of the required diesel generators shall be demonstrated OPERABLE at least once per 31 days and after each operation of the diesel where the period of operation was ≥1 hour by removing any accumulated water from the day tank.

4. Each of the required diesel generators shall be demonstrated OPERABLE at least once per 92 days by checking for and removing accumulated water from the fuel oil bulk storage tanks.

unless the absence of any potential common mode failure for the remaining diesel generator is demonstrated

---

b Contrary to the provisions of Specification 3.0.B, this test is required to be completed regardless of when the inoperable diesel generator is restored to OPERABILITY for failures that are potentially generic to the remaining diesel generator and for which appropriate alternative testing cannot be designed.

c Surveillance Requirement~~s~~ 4.9.A.7 ~~a and b~~ may be substituted for Surveillance Requirement~~s~~ 4.9.A.2.c ~~and d~~. e

d This band is meant as guidance to avoid routine overloading of the engine. Loads in excess of this band for special testing under direct monitoring by the manufacturer or momentary variations due to changing bus loads shall not invalidate the test.

'INSERT'

INSERT

Momentary transients outside of the load range do not invalidate this test. Diesel generator loadings may include gradual loading as recommended by the manufacturer/vendor. This surveillance shall be conducted on only one diesel generator at a time.

3.9 - LIMITING CONDITIONS FOR OPERATION

e

b. Demonstrate the OPERABILITY of each diesel generator by performing Surveillance Requirement 4.9.A.2.c for each diesel generator separately within 24 hours (if it has not been successfully tested within the past 24 hours) and within the subsequent 72 hours, and

b e

~~c.~~ Restore the inoperable offsite circuit to OPERABLE status within 7 days or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

2. With one of the above required diesel generator power sources inoperable:

   a. Demonstrate the OPERABILITY of the offsite circuit power sources by performing Surveillance Requirement 4.9.A.1.a within 1 hour and at least once per 8 hours thereafter.

   b. If the diesel generator is inoperable due to any cause other than an inoperable support system, an independently testable component, or preplanned ~~preventive~~ maintenance or testing, demonstrate the OPERABILITY of the remaining OPERABLE diesel generator by performing Surveillance Requirement 4.9.A.2.c within 24 hours[b] unless the absence of any potential common mode failure for the remaining diesel generator is demonstrated (if it has not been successfully tested within the past 24 hours) and within the subsequent 72 hours, and

4.9 - SURVEILLANCE REQUIREMENTS

   c. Verifying[c] the diesel starts and accelerates to synchronous speed with generator voltage and frequency at 4160 ±420 volts and 60 ±1.2 Hz, respectively.

   d. Verifying~~[c]~~ the diesel generator is synchronized, loaded to between 2375 and 2500 kW[d] in accordance with the manufacturer's/vendor's recommendations, and operates with this load for ≥60 minutes.

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

   f. Verifying the pressure in required starting air receiver tanks to be ≥230 psig.

3. Each of the required diesel generators shall be demonstrated OPERABLE at least once per 31 days and after each operation of the diesel where the period of operation was ≥1 hour by removing any accumulated water from the day tank.

4. Each of the required diesel generators shall be demonstrated OPERABLE at least once per 92 days by checking for and removing accumulated water from the fuel oil bulk storage tanks.

---

b Contrary to the provisions of Specification 3.0.B, this test is required to be completed regardless of when the inoperable diesel generator is restored to OPERABILITY for failures that are potentially generic to the remaining diesel generator and for which appropriate alternative testing cannot be designed.

c Surveillance Requirement~~s~~ 4.9.A.7 ~~a and b~~ may be substituted for Surveillance Requirement~~s~~ 4.9.A.2.c ~~and d~~.

d This band is meant as guidance to avoid routine overloading of the engine. Loads in excess of this band for special testing under direct monitoring by the manufacturer or momentary variations due to changing bus loads shall not invalidate the test.

'INSERT'

INSERT

Momentary transients outside of the load range do not invalidate this test. Diesel generator loadings may include gradual loading as recommended by the manufacturer/vendor. This surveillance shall be conducted on only one diesel generator at a time.

3.9 - LIMITING CONDITIONS FOR OPERATION

c. Restore the diesel generator to OPERABLE status within 7 days or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

3. With one of the above offsite circuit power sources and one of the above required diesel generator power sources inoperable:

   a. Demonstrate the OPERABILITY of the remaining offsite circuit power source by performing Surveillance Requirement 4.9.A.1.a within 1 hour and at least once per 8 hours thereafter.

   b. If the diesel generator is inoperable due to any cause other than preplanned ~~preventive~~ maintenance or testing, demonstrate the OPERABILITY[e] of the remaining OPERABLE diesel generator by performing Surveillance Requirement 4.9.A.2.c[b] within 8 hours ~~(if it has not been~~ successfully tested within the past 24 hours) and within the subsequent 72 hours for each OPERABLE diesel generator.

   c. Restore at least one of the inoperable A.C. power sources to OPERABLE status within 12 hours or be in at least HOT SHUTDOWN

4.9 - SURVEILLANCE REQUIREMENTS

5. Each of the required diesel generators shall be demonstrated OPERABLE by:

   a. Sampling new fuel oil prior to addition to the storage tanks in accordance with applicable ASTM standards, and

   b. Verifying prior to addition to the storage tanks that the sample meets the applicable ASTM standards for API gravity, water and sediment, and the visual test for free water and particulate contamination, and

   c. Verifying within 31 days of obtaining the sample that the kinematic viscosity is within applicable ASTM limits.

6. Each of the required diesel generators shall be demonstrated OPERABLE by:

   a. Sampling and analyzing the bulk fuel storage tanks at least once per 31days in accordance with applicable ASTM standards, and

   b. Verifying that the sample meets the applicable ASTM standards for water and sediment, kinematic viscosity, and ASTM particulate contaminant is <10 mg/liter.

unless the absence of any potential common mode failure for the remaining diesel generator is demonstrated

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e A successful test of OPERABILITY per Surveillance Requirement 4.9.A.2.c under this ACTION statement satisfies the ~~diesel generator test requirements of ACTION(s) 1 or 2 above.~~

b Contrary to the provisions of Specification 3.0.B, this test is required to be completed regardless of when the inoperable diesel generator is restored to OPERABILITY for failures that are potentially generic to the remaining diesel generator and for which appropriate alternative testing cannot be designed.

3.9 - LIMITING CONDITIONS FOR OPERATION

c. Restore the diesel generator to OPERABLE status within 7 days or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

3. With one of the above offsite circuit power sources and one of the above required diesel generator power sources inoperable:

a. Demonstrate the OPERABILITY of the remaining offsite circuit power source by performing Surveillance Requirement 4.9.A.1.a within 1 hour and at least once per 8 hours thereafter.

b. If the diesel generator is inoperable due to any cause other than preplanned ~~preventive~~ maintenance or testing, demonstrate the OPERABILITY[e] of the remaining OPERABLE diesel generator by performing Surveillance Requirement 4.9.A.2.c[b] within 8 hours ~~(if it has not been successfully tested within the past 24 hours)~~ unless the absence of any potential common mode failure for the remaining diesel generator is demonstrated and within the subsequent 72 hours for each OPERABLE diesel generator.

c. Restore at least one of the inoperable A.C. power sources to OPERABLE status within 12 hours or be in at least HOT SHUTDOWN

4.9 - SURVEILLANCE REQUIREMENTS

5. Each of the required diesel generators shall be demonstrated OPERABLE by:

a. Sampling new fuel oil prior to addition to the storage tanks in accordance with applicable ASTM standards, and

b. Verifying prior to addition to the storage tanks that the sample meets the applicable ASTM standards for API gravity, water and sediment, and the visual test for free water and particulate contamination[g], ~~and~~ (h)

c. Verifying within 31 days of obtaining the sample that the kinematic viscosity is within applicable ASTM limits.

6. Each of the required diesel generators shall be demonstrated OPERABLE by:

a. Sampling and analyzing the bulk fuel storage tanks at least once per 31days in accordance with applicable ASTM standards, and

b. Verifying that the sample meets the applicable ASTM standards for water and sediment, kinematic viscosity, and ASTM particulate contaminant[g] is < 10 mg/liter. (h)

---

e A successful test of OPERABILITY per Surveillance Requirement 4.9.A.2.c under this ACTION statement satisfies the diesel generator test requirements of ACTION(s) 1 or 2 above.

b Contrary to the provisions of Specification 3.0.B, this test is required to be completed regardless of when the inoperable diesel generator is restored to OPERABILITY for failures that are potentially generic to the remaining diesel generator and for which appropriate alternative testing cannot be designed.

g (h) The particulate contamination surveillance is not required for No. 1 fuel oil. It is required for No. 2 fuel oil and for blends.

3.9 - LIMITING CONDITIONS FOR OPERATION

within the next 12 hours and in COLD SHUTDOWN within the following 24 hours, and

d. Restore both offsite circuits and both diesel generators to OPERABLE status within 7 days from the time of the initial loss or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

4. With one of the above required diesel generator power sources inoperable, in addition to ACTION 2 or 3, as applicable:

   a. Verify within 2 hours that at least one of the required two systems, subsystems, trains, components and devices in two train systems is OPERABLE including its emergency power supply.

   b. Otherwise, take the applicable ACTIONs for both systems, subsystems, trains, components or devices inoperable, or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

4.9 - SURVEILLANCE REQUIREMENTS

7. Each of the required diesel generators shall be demonstrated OPERABLE[a] at least once per 184 days by:

   ~~a.~~ Verifying[c] the diesel starts and accelerates to synchronous speed in ≤13 seconds. The generator voltage and frequency shall be verified to reach 4160 ±420 volts and 60 ±1.2 Hz, respectively, in ≤13 seconds after the start signal.

   b. Verifying[c] the diesel generator is synchronized, loaded to between 2470 and 2600 kW[d] in ≤200 seconds, and operates with this load for ≥60 minutes.

8. Each of the required diesel generators shall be demonstrated OPERABLE[a] at least once per 18 months by:

   a. Subjecting the diesel to an inspection in accordance with instructions prepared in conjunction with its manufacturer's recommendations for this class of standby service.

'Deleted'

INSERT

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a All planned diesel generator tests shall be conducted in accordance with the manufacturer's recommendations regarding engine prelube, leak detection and warmup procedures, and as applicable regarding loading and shutdown recommendations.

c Surveillance Requirement~~s~~ 4.9.A.7~~.a and b~~ may be substituted for Surveillance Requirement~~s~~ 4.9.A.2.c ~~and d~~.

d This band is meant as guidance to avoid routine overloading of the engine. Loads in excess of this band for special testing under direct monitoring by the manufacturer or momentary variations due to changing bus loads shall not invalidate the test.

INSERT

All diesel generator starts may be preceded by an engine prelube period. All diesel generator starts that require loading may be preceded by an engine prelube period and followed by a warmup period prior to loading. Diesel generator loadings may include gradual loading as recommended by the manufacturer/vendor.

3.9 - LIMITING CONDITIONS FOR OPERATION

within the next 12 hours and in COLD SHUTDOWN within the following 24 hours, and

d. Restore both offsite circuits and both diesel generators to OPERABLE status within 7 days from the time of the initial loss or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

4. With one of the above required diesel generator power sources inoperable, in addition to ACTION 2 or 3, as applicable:

   a. Verify within 2 hours that at least one of the required two systems, subsystems, trains, components and devices in two train systems is OPERABLE including its emergency power supply.

   b. Otherwise, take the applicable ACTIONs for both systems, subsystems, trains, components or devices inoperable, or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

4.9 - SURVEILLANCE REQUIREMENTS

7. Each of the required diesel generators shall be demonstrated OPERABLE[a] at least once per 184 days by:

   a. Verifying[c] the diesel starts and accelerates to synchronous speed in ≤13 seconds. The generator voltage and frequency shall be verified to reach 4160 ±420 volts and 60 ±1.2 Hz, respectively, in ≤13 seconds after the start signal.

   10

   b. Verifying[c] the diesel generator is synchronized, loaded to between 2375 and 2500 kW[d] in ≤200 seconds, and operates with this load for ≥60 minutes.

8. Each of the required diesel generators shall be demonstrated OPERABLE[a] at least once per 18 months by:

   a. Subjecting the diesel to an inspection in accordance with instructions prepared in conjunction with its manufacturer's recommendations for this class of standby service.

'Deleted'

'INSERT'

a All planned diesel generator tests shall be conducted in accordance with the manufacturer's recommendations regarding engine prelube, leak detection and warmup procedures, and as applicable regarding loading and shutdown recommendations.

c Surveillance Requirements 4.9.A.7 ~~a and b~~ may be substituted for Surveillance Requirements 4.9.A.2.c ~~and d~~.

d This band is meant as guidance to avoid routine overloading of the engine. Loads in excess of this band for special testing under direct monitoring by the manufacturer or momentary variations due to changing bus loads shall not invalidate the test.

INSERT

All diesel generator starts may be preceded by an engine prelube period. All diesel generator starts that require loading may be preceded by an engine prelube period and followed by a warmup period prior to loading. Diesel generator loadings may include gradual loading as recommended by the manufacturer/vendor.

3.9 - LIMITING CONDITIONS FOR OPERATION

5. With two of the above required offsite circuit power sources inoperable:

e

a. Demonstrate the OPERABILITY[e] of both of the above required diesel generators separately by performing Surveillance Requirement 4.9.A.2.c within 8 hours (if it has not been successfully tested within the past 24 hours), unless the diesel generators are already operating, and within the subsequent 72 hours.

a. b. Restore at least one of the inoperable offsite circuits to OPERABLE status within 24 hours or be in at least HOT SHUTDOWN within the next 12 hours and e ~~with~~in COLD SHUTDOWN within the following 24 hours, and

b. c. e Restore at least two offsite circuits to OPERABLE status within 7 days from the time of initial loss or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

6. With both of the above required diesel generator power sources inoperable:

a. Demonstrate the OPERABILITY of the offsite circuit power sources by performing Surveillance Requirement 4.9.A.1.a within 1 hour and at least once per 8 hours thereafter.

4.9 - SURVEILLANCE REQUIREMENTS

b. [INTENTIONALLY BLANK] e

This requirement is an open item to be addessed in the TSUP clean-up amendment

Verifying the diesel generator capability to reject its largest single emergency load (≥642 kW) while maintaining speed ≤1001 rpm and voltage at 4160 ± 420 volts.

c. Verifying the diesel generator capability to reject a ~~full emergency~~ e load between 2470 and 2600kW[d], without tripping on overspeed. The generator voltage shall not exceed 5000 volts during or following the load rejection.

d. Simulating a loss of offsite power by itself, and:

1) Verifying de-energization of the emergency buses, and load shedding from the emergency buses.

2) Verifying the diesel starts on the auto-start signal, energizes the emergency buses with permanently connected loads in ≤13 seconds, energizes the auto-connected shutdown loads, and operates with this load for ≥5 minutes. After energization, the steady-state voltage and frequency of the emergency busses shall be maintained at 4160 ±420 volts and 60 ±1.2 Hz, respectively, during this test.

d 'INSERT'

e

e A successful test of OPERABILITY per Surveillance Requirement 4.9.A.2.c under this ACTION statement satisfies the diesel generator test requirements of ACTION(s) 1 or 2 above.

INSERT

Momentary transients outside of the load range do not invalidate this test. Diesel generator loadings may include gradual loading as recommended by the manufacturer/vendor. This surveillance shall be conducted on only one diesel generator at a time.

3.9 - LIMITING CONDITIONS FOR OPERATION

5. With two of the above required offsite circuit power sources inoperable:

    a. Demonstrate the OPERABILITY[e] of both of the above required diesel generators separately by performing Surveillance Requirement 4.9.A.2.c within 8 hours (if it has not been successfully tested within the past 24 hours), unless the diesel generators are already operating, and within the subsequent 72 hours.

    b. Restore at least one of the inoperable offsite circuits to OPERABLE status within 24 hours or be in at least HOT SHUTDOWN within the next 12 hours and within COLD SHUTDOWN within the following 24 hours.

    c. Restore at least two offsite circuits to OPERABLE status within 7 days from the time of initial loss or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

6. With both of the above required diesel generator power sources inoperable:

    a. Demonstrate the OPERABILITY of the offsite circuit power sources by performing Surveillance Requirement 4.9.A.1.a within 1 hour and at least once per 8 hours thereafter.

'INSERT'

4.9 - SURVEILLANCE REQUIREMENTS

b. [INTENTIONALLY BLANK]

This requirement is an open item to be addessed in the TSUP clean-up amendment

Verifying the diesel generator capability to reject its largest single emergency load (≥ 725 kW) while maintaining speed ≤ 1001 rpm and voltage at 4160 ± 420 volts.

c. Verifying the diesel generator capability to reject a ~~full emergency~~ load between 2375 and 3500 kW,(d) without tripping. The generator voltage shall not exceed 5000 volts during or following the load rejection. on overspeed

d. Simulating a loss of offsite power by itself, and:

    1) Verifying de-energization of the emergency buses, and load shedding from the emergency buses.

    2) Verifying the diesel starts on the auto-start signal, energizes the emergency buses with permanently connected loads in ≤13 10 seconds, energizes the auto-connected shutdown loads, and operates with this load for ≥5 minutes. After energization, the steady-state voltage and frequency of the emergency busses shall be maintained at 4160 ±420 volts and 60 ±1.2 Hz, respectively, during this test.

---

e A successful test of OPERABILITY per Surveillance Requirement 4.9.A.2.c under this ACTION statement satisfies the diesel generator test requirements of ACTION(s) 1 or 2 above.

INSERT

Momentary transients outside of the load range do not invalidate this test. Diesel generator loadings may include gradual loading as recommended by the manufacturer/vendor. This surveillance shall be conducted on only one diesel generator at a time.

3.9 - LIMITING CONDITIONS FOR OPERATION

b. Within 2 hours, restore at least one of the above required diesel generators to OPERABLE[e] status and verify that at least one of the required two systems, subsystems, trains, components and devices in two train systems is OPERABLE including its emergency power supply. Otherwise, take the applicable ACTIONs for both systems, subsystems, trains, components or devices inoperable, or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

c. Demonstrate the continued OPERABILITY of the restored diesel generator by performing Surveillance Requirement 4.9.A.2.c within the subsequent 72 hours, and

d. Restore at least two required diesel generators to OPERABLE status within 7 days from the time of initial loss or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

7. With the fuel oil contained in the bulk fuel storage tank(s) not meeting the properties specified in Surveillance Requirements 4.9.A.5 and 4.9.A.6, restore the fuel oil properties to within the specified limits within 7 days. Otherwise, declare the associated diesel generator(s) inoperable.

4.9 - SURVEILLANCE REQUIREMENTS

e. Verifying that on an ECCS actuation test signal, without loss of offsite power, the diesel generator starts on the auto-start signal and operates on standby for ≥5 minutes. The generator voltage and frequency shall be 4160 ±420 volts and 60 ±1.2 Hz, respectively, in ≤~~13~~ 10 seconds after the auto-start signal; the steady state generator voltage and frequency shall be maintained within these limits during this test.

f. Simulating a loss of offsite power in conjunction with an ECCS actuation test signal , and

1) Verifying de-energization of the emergency buses, and load shedding from the emergency buses.

2) Verifying the diesel starts on the auto-start signal, energizes the emergency buses with permanently connected loads in ≤~~13~~ 10 seconds, energizes the auto-connected emergency loads through the load sequencer, and operates with this load for ≥5 minutes. After energization, the steady-state voltage and frequency of the emergency busses shall be maintained at 4160 ±420 volts and 60 ±1.2 Hz, respectively, during this test.

---

e A successful test of OPERABILITY per Surveillance Requirement 4.9.A.2.c under this ACTION statement satisfies the diesel generator test requirements of ACTION(s) 1 or 2 above.

3.9 - LIMITING CONDITIONS FOR OPERATION

8. [INTENTIONALLY BLANK]
This requirement is an open item to be addressed in the TSUP clean-up amendment

4.9 - SURVEILLANCE REQUIREMENTS

g. Verifying that all automatic diesel generator trips, except engine overspeed and generator differential current are automatically bypassed upon an emergency actuation signal.

h. Verifying the diesel generator operates for ≥24 hours. During the first 2 hours of this test, the diesel generator shall be loaded to between 2730 and 2860 kW[d] and during the remaining 22 hours of this test, the diesel generator shall be loaded to between 2470 and 2600 kW[d]. The generator voltage and frequency shall be 4160 ±420 volts and 60 ±1.2 Hz, respectively, in ≤13 seconds after the start signal; the steady state generator voltage and frequency shall be maintained within these limits during this test. Within 5 minutes after completing this 24 hour test, perform Surveillance Requirement ~~4.9.A.8.f.2[e]~~. 4.9.A.2.c (f)

i. Verifying that the auto-connected loads to each diesel generator do not exceed the 2000 hour rating of 2860 kW.

'INSERT'

f Criteria for determining the number of valid failures and number of valid tests shall be in accordance with draft Revision 3 of Regulatory Guide 1.9, January 1991, but determined on a per diesel generator basis. With the exception of the semi-annual fast start, no starting time requirements are required to meet the valid test requirements.

d This band is meant as guidance to avoid routine overloading of the engine. Loads in excess of this band for special testing under direct monitoring by the manufacturer or momentary variations due to changing bus loads shall not invalidate the test.

~~e~~ f If Surveillance Requirement ~~4.9.A.8.f.2~~ 4.9.A.2.c is not satisfactorily completed, it is not necessary to repeat the preceding 24 hour test. Instead, the diesel generator may be operated at approximately full load for ~~1 hour~~ 2 hours or until the operating temperature has stabilized.

INSERT

Momentary transients outside of the load range do not invalidate this test. Diesel generator loadings may include gradual loading as recommended by the manufacturer/vendor. This surveillance shall be conducted on only one diesel generator at a time.

3.9 - LIMITING CONDITIONS FOR OPERATION

8. [INTENTIONALLY BLANK]
This requirement is an open item to be addressed in the TSUP clean-up amendment

4.9 - SURVEILLANCE REQUIREMENTS

g. Verifying that all automatic diesel generator trips, except engine overspeed and generator differential current are automatically bypassed upon an emergency actuation signal.

h. Verifying the diesel generator operates for ≥24 hours. During the first 2 hours of this test, the diesel generator shall be loaded to between 2625 and 2750 kW[d] and during the remaining 22 hours of this test, the diesel generator shall be loaded to between 2375 and 2500 kW[d]. The generator voltage and frequency shall be 4160 ±420 volts and 60 ±1.2 Hz, respectively, in ≤~~13~~ 10 seconds after the start signal; the steady state generator voltage and frequency shall be maintained within these limits during this test. Within 5 minutes after completing this 24 hour test, perform Surveillance Requirement ~~4.9.A.8.1.2~~ 4.9.A.2.c[~~e~~ f].

i. Verifying that the auto-connected loads to each diesel generator do not exceed the 2000 hour rating of 2850 kW.

---

f. Criteria for determining the number of valid failures and number of valid tests shall be in accordance with draft Revision 3 of Regulatory Guide 1.9, January 1991, but determined on a per diesel generator basis. With the exception of the semi-annual fast start, no starting time requirements are required to meet the valid test requirements.

d. This band is meant as guidance to avoid routine overloading of the engine. Loads in excess of this band for special testing under direct monitoring by the manufacturer or momentary variations due to changing bus loads shall not invalidate the test.

~~e~~ f. If Surveillance Requirement ~~4.9.A.8.1.2~~ 4.9.A.2.c is not satisfactorily completed, it is not necessary to repeat the preceding 24 hour test. Instead, the diesel generator may be operated at approximately full load for ~~1 hour~~ 2 hours or until the operating temperature has stabilized.

INSERT

Momentary transients outside of the load range do not invalidate this test. Diesel generator loadings may include gradual loading as recommended by the manufacturer/vendor. This surveillance shall be conducted on only one diesel generator at a time.

3.9 - LIMITING CONDITIONS FOR OPERATION

4.9 - SURVEILLANCE REQUIREMENTS

j. Verifying the diesel generator's capability to:

1) synchronize with the offsite power source while the generator is loaded with its emergency loads upon a simulated restoration of offsite power,

2) transfer its loads to the offsite power source, and

3) be restored to its standby status.

k. Verifying that the automatic load sequence logic ~~timer~~ is OPERABLE with the interval between each load block within ±10% of its design interval.

9. Each of the required diesel generators shall be demonstrated OPERABLE[a] at least once per 10 years or after any modifications which could affect diesel generator interdependence by starting both diesel generators simultaneously, and verifying that both diesel generators accelerate to ≥900 rpm in ≤13 seconds.

10. Each of the required diesel generators shall be demonstrated OPERABLE at least once per 10 years by draining each fuel oil storage tank, removing the accumulated sediment and cleaning the tank.

INSERT

a All planned diesel generator tests shall be conducted in accordance with the manufacturer's recommendations regarding engine prelube, leak detection and warmup procedures, and as applicable regarding loading and shutdown recommendations.

INSERT

All diesel generator starts may be preceded by an engine prelube period. All diesel generator starts that require loading may be preceded by an engine prelube period and followed by a warmup period prior to loading. Diesel generator loadings may include gradual loading as recommended by the manufacturer/vendor.

3.9 - LIMITING CONDITIONS FOR OPERATION

4.9 - SURVEILLANCE REQUIREMENTS

j. Verifying the diesel generator's capability to:

   1) synchronize with the offsite power source while the generator is loaded with its emergency loads upon a simulated restoration of offsite power,

   2) transfer its loads to the offsite power source, and

   3) be restored to its standby status.

logic

k. Verifying that the automatic load sequence timer is OPERABLE with the interval between each load block within ±10% of its design interval.

9. Each of the required diesel generators shall be demonstrated OPERABLE[a] at least once per 10 years or after any modifications which could affect diesel generator interdependence by starting both diesel generators simultaneously, and verifying that both diesel generators accelerate to ≥900 rpm in ≤13 seconds.

10

10. Each of the required diesel generators shall be demonstrated OPERABLE at least once per 10 years by draining each fuel oil storage tank, removing the accumulated sediment and cleaning the tank.

'INSERT'

a All planned diesel generator tests shall be conducted in accordance with the manufacturer's recommendations regarding engine prelube, leak detection and warmup procedures, and as applicable regarding loading and shutdown recommendations.

INSERT

All diesel generator starts may be preceded by an engine prelube period. All diesel generator starts that require loading may be preceded by an engine prelube period and followed by a warmup period prior to loading. Diesel generator loadings may include gradual loading as recommended by the manufacturer/vendor.

TABLE 4.9.A-1

DIESEL GENERATOR TEST SCHEDULE

NUMBER OF FAILURES IN LAST 20 VALID TESTS[a]	TEST FREQUENCY
≤1	At least once per 31 days
≤2[b]	At least once per 7 days

(Not Used)

---

a Criteria for determining the number of valid failures and number of valid tests shall be in accordance with draft Revision 3 of Regulatory Guide 1.9, January 1991, but determined on a per diesel generator basis. With the exception of the semi-annual fast start, no starting time requirements are required to meet the valid test requirements.

b The associated test frequency shall be maintained until 7 consecutive failure free demands have been performed AND the number of failures in the last 20 valid demands has been reduced to one.

TABLE 4.9.A-1

DIESEL GENERATOR TEST SCHEDULE

NUMBER OF FAILURES IN LAST 20 VALID TESTS[a]	TEST FREQUENCY
≤1	At least once per 31 days
≤2[b]	At least once per 7 days

(Not Used)

a Criteria for determining the number of valid failures and number of valid tests shall be in accordance with draft Revision 3 of Regulatory Guide 1.9, January 1991, but determined on a per diesel generator basis. With the exception of the semi-annual fast start, no starting time requirements are required to meet the valid test requirements.

b The associated test frequency shall be maintained until 7 consecutive failure free demands have been performed AND the number of failures in the last 20 valid demands has been reduced to one.

3.9 - LIMITING CONDITIONS FOR OPERATION

C. D.C. Sources - Operating

As a minimum, the following D.C. electrical power sources shall be OPERABLE with the identified parameters within the limits specified in Table 4.9.C-1:

1. Two station 250 volt batteries, each with a full capacity charger.
2. Two station 125 volt batteries, each with a full capacity charger.
3. ~~One~~ Two unit 24/48 volt battery ies, with a full capacity charger. each

APPLICABILITY:

OPERATIONAL MODE(s) 1, 2, and 3.

ACTION:

With one of the above required 24/48 volt or 250 volt station batteries and/or chargers inoperable, restore the inoperable equipment to OPERABLE status within 2 hours[b]

1. [INTENTIONALLY BLANK]

This requirement is an open item to be addressed in the TSUP clean-up amendment

2. [INTENTIONALLY BLANK]

This requirement is an open item to be addressed in the TSUP clean-up amendment

With one of the above required 125 volt station batteries and/or chargers inoperable, within 2 hours[c], either restore the inoperable equipment to OPERABLE status, or place an OPERABLE corresponding alternate 125 volt battery (with an OPERABLE full capacity charger) in service.

4.9 - SURVEILLANCE REQUIREMENTS

C. D.C. Sources - Operating

Each of the required 24/48 volt, 125 volt and 250 volt batteries and chargers shall be demonstrated OPERABLE[a]:

1. At least once per 7 days by verifying that:
   a. The parameters in Table 4.9.C-1 meet Category A limits, and
   b. There is correct breaker alignment to the battery chargers and total battery terminal voltage is ≥26.0, ≥125.9, or ≥260.4 volts, as applicable, on float charge.

2. At least once per 92 days and within 7 days after a battery discharge with a battery terminal voltage below 21.7, 105 or 210 volts, as applicable, or battery overcharge with battery terminal voltage above 30, 150 or 300 volts, as applicable, by verifying that:
   a. The parameters in Table 4.9.C-1 meet the Category B limits,
   b. There is no visible corrosion at either terminals or connectors, or the connection resistance of these items is ≤150 x10^{-6} ohms or ≤20% above baseline connection resistance, whichever is higher, and
   c. ~~[INTENTIONALLY BLANK]~~

   The average electrolyte temperature of all connected cells is above 60°F.

---

a An alternate 125 volt battery shall adhere to these same Surveillance Requirements to be considered OPERABLE, except the Unit 2 total battery terminal voltage on float charge shall be verified weekly as ≥130.2 volts.

'INSERT'

INSERT

[Dresden]

b Each 250 volt battery may be inoperable for a maximum of seven days per operating cycle for maintenance or testing. If it is determined that a 250 volt battery need be replaced as a result of maintenance or testing, a specific battery may be inoperable for an additional seven days per operating cycle.

c With Unit 2 and 3 in OPERATIONAL MODE(s) 1, 2 or 3, each 125 volt battery may be inoperable for up to a maximum of seven days per operating cycle for maintenance or testing provided the alternate 125 volt battery is placed into service and is OPERABLE. If it is determined that a 125 volt battery need be replaced as a result of maintenance or testing, a specific battery may be inoperable for an additional seven days provided the alternate 125 volt battery is placed into service and is OPERABLE. With the other Unit in MODE(s) 4 or 5, operations may continue with one of the two 125 volt battery systems inoperable provided the alternate 125 volt battery is placed into service and is OPERABLE.

3.9 - LIMITING CONDITIONS FOR OPERATION

C. D.C. Sources - Operating

As a minimum, the following D.C. electrical power sources shall be OPERABLE with the identified parameters within the limits specified in Table 4.9.C-1:

1. Two station 250 volt batteries, each with a full capacity charger.
2. Two station 125 volt batteries, each with a full capacity charger.

APPLICABILITY:

OPERATIONAL MODE(s) 1, 2, and 3.

ACTION:

With one of the above required 250 volt station batteries and/or chargers inoperable, restore the inoperable equipment to OPERABLE status within 72 hours.

1. [INTENTIONALLY BLANK]

This requirement is an open item to be addressed in the TSUP clean-up amendment

2. [INTENTIONALLY BLANK]

This requirement is an open item to be addressed in the TSUP clean-up amendment

With one of the above required 125 volt station batteries and/or chargers inoperable, within 72 hours[b] either restore the inoperable equipment to OPERABLE status, or place an OPERABLE corresponding alternate 125 volt battery (with an OPERABLE full capacity charger) in service.

4.9 - SURVEILLANCE REQUIREMENTS

C. D.C. Sources - Operating

Each of the required 125 volt and 250 volt batteries and chargers shall be demonstrated OPERABLE[a]:

1. At least once per 7 days by verifying that:
   a. The parameters in Table 4.9.C-1 meet Category A limits, and
   b. There is correct breaker alignment to the battery chargers and total battery terminal voltage is ≥125.9 or ≥260.4 volts, as applicable, on float charge.
2. At least once per 92 days and within 7 days after a battery discharge with a battery terminal voltage below 105 or 210 volts, as applicable, or battery overcharge with battery terminal voltage above 150 or 300 volts, as applicable, by verifying that:
   a. The parameters in Table 4.9.C-1 meet the Category B limits,
   b. There is no visible corrosion at either terminals or connectors, or the connection resistance of these items is ≤150 x10^{-6} ohms or ≤20% above baseline connection resistance, whichever is higher, and
   c. ~~[INTENTIONALLY BLANK]~~

   The average electrolyte temperature of all connected cells is above 60°F.

a An alternate 125 volt battery shall adhere to these same Surveillance Requirements to be considered OPERABLE.

'INSERT'

INSERT

[Quad Cities]

b With Unit 1 and 2 in OPERATIONAL MODE(s) 1, 2 or 3, each 125 volt battery may be inoperable for up to a maximum of seven days per operating cycle for maintenance or testing provided the alternate 125 volt battery is placed into service and is OPERABLE. If it is determined that a 125 volt battery need be replaced as a result of maintenance or testing, a specific battery may be inoperable for an additional seven days provided the alternate 125 volt battery is placed into service and is OPERABLE. With the other Unit in MODE(s) 4 or 5, operations may continue with one of the two 125 volt battery systems inoperable provided the alternate 125 volt battery is placed into service and is OPERABLE.

3.9 - LIMITING CONDITIONS FOR OPERATION

3. With the provisions of either ACTION 1 or 2 above not met, be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

4. With any Category A parameter(s) outside the limit(s) shown in Table 4.9.C-1, the battery may be considered OPERABLE provided that its associated charger is OPERABLE, and 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.

5. With any Category B parameter(s) outside the limit(s) shown in Table 4.9.C-1, 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 the limit(s) within 7 days.

6. With any Category B parameter not within its allowable value(s), immediately declare the battery inoperable.

4.9 - SURVEILLANCE REQUIREMENTS

3. At least every 18 months by verifying that:

   a. The cells, cell plates and battery racks show no visual indication of physical damage or abnormal deterioration.

   b. The cell-to-cell and terminal connections are clean, tight, free of corrosion and coated with anti-corrosion material.

   c. The resistance of each cell-to-cell and terminal connection is $\leq 150 \times 10^{-6}$ ohms or $\leq 20\%$ above baseline connection resistance, whichever is higher.

   d. The battery chargers will supply a load equal to the manufacturer's rating for at least 4 hours.

4. At least every 18 months, by verifying that the battery capacity is adequate to supply and maintain in OPERABLE status all of the actual or simulated emergency loads for design duty cycle when the battery is subjected to a battery service test.

5. [INTENTIONALLY BLANK]

   This requirement is an open item to be addessed in the TSUP clean-up amendment

'INSERT'

INSERT

[Dresden - 4.9.C.5]

At least once per 60 months, verify that the battery capacity is 80% of the manufacturer's rating when subjected to either a performance discharge test or a modified performance discharge test. The modified performance discharge test satisfies the requirements of both the service test and performance test and therefore, may be performed in lieu of a service test.

3.9 - LIMITING CONDITIONS FOR OPERATION

3. With the provisions of either ACTION 1 or 2 above not met, be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

4. With any Category A parameter(s) outside the limit(s) shown in Table 4.9.C-1, the battery may be considered OPERABLE provided that its associated charger is OPERABLE, and 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.

5. With any Category B parameter(s) outside the limit(s) shown in Table 4.9.C-1, 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 the limit(s) within 7 days.

6. With any Category B parameter not within its allowable value(s), immediately declare the battery inoperable.

4.9 - SURVEILLANCE REQUIREMENTS

3. At least every 18 months by verifying that:

   a. The cells, cell plates and battery racks show no visual indication of physical damage or abnormal deterioration.

   b. The cell-to-cell and terminal connections are clean, tight, free of corrosion and coated with anti-corrosion material.

   c. The resistance of each cell-to-cell and terminal connection is $\leq 150 \times 10^{-6}$ ohms or $\leq 20\%$ above baseline connection resistance, whichever is higher.

   d. The battery chargers will supply a load equal to the manufacturer's rating for at least 4 hours.

4. At least every 18 months, by verifying that the battery capacity is adequate to supply and maintain in OPERABLE status all of the actual or simulated emergency loads for design duty cycle when the battery is subjected to a battery service test.

5. [INTENTIONALLY BLANK]

This requirement is an open item to be addessed in the TSUP clean-up amendment

'INSERT'

INSERT

[Quad Cities - 4.9.C.5]

At least once per 60 months, verify that the battery capacity is at least the greater of either 80% of the manufacturer's rating or the minimum acceptable battery capacity from the load profile when subjected to either a performance discharge test or a modified performance discharge test. The modified performance discharge test satisfies the requirements of both the service test and the performance test and therefore, may be performed in lieu of a service test.

3.9 - LIMITING CONDITIONS FOR OPERATION

4.9 - SURVEILLANCE REQUIREMENTS

6. [INTENTIONALLY BLANK]
This requirement is an open item to be addressed in the TSUP clean-up amendment

'INSERT'

INSERT

For any battery that shows signs of degradation or has reached 85% of the service life for the expected application and delivers a capacity of less than 100% of the manufacturer's rated capacity, a performance discharge test or a modified performance test of battery capacity shall be performed at least once every 12 months or the battery shall be replaced or restored to 100% or greater of the manufacturer's rated capacity during the next refuel outage. Degradation is indicated when the battery capacity drops more than 10% from its capacity on the previous performance test, or is below 90% of the manufacturer's rating. If the battery has reached 85% of service life, delivers a capacity of 100% or greater of the manufacturer's rated capacity and has shown no signs of degradation, a performance test or a modified performance test of battery capacity shall be performed at least once every two years.

3.9 - LIMITING CONDITIONS FOR OPERATION

4.9 - SURVEILLANCE REQUIREMENTS

6. [INTENTIONALLY BLANK]
This requirement is an open item to be addressed in the TSUP clean-up amendment

INSERT

INSERT

For any battery that shows signs of degradation or has reached 85% of the service life for the expected application and delivers a capacity of less than 100% of the manufacturer's rated capacity, a performance discharge test or a modified performance test of battery capacity shall be performed at least once every 12 months or the battery shall be replaced or restored to 100% or greater of the manufacturer's rated capacity during the next refuel outage. Degradation is indicated when the battery capacity drops more than 10% from its capacity on the previous performance test, or is below 90% of the manufacturer's rating. If the battery has reached 85% of service life, delivers a capacity of 100% or greater of the manufacturer's rated capacity and has shown no signs of degradation, a performance test or a modified performance test of battery capacity shall be performed at least once every two years.

3.9 - LIMITING CONDITIONS FOR OPERATION

E. Distribution - Operating

The following power distribution systems shall be energized ~~with tie breakers open both between redundant buses within the unit and between units at the same station~~:

1. A.C. power distribution, consisting of:

   a. Both Unit engineered safety features 4160 volt buses:

      1) For Unit 2, Nos. 23-1 and 24-1,

      2) For Unit 3, Nos. 33-1 and 34-1.

   b. Both Unit engineered safety features 480 volt buses:

      1) For Unit 2, Nos. 28 and 29,

      2) For Unit 3, Nos. 38 and 39.

   c. The Unit 120 volt Essential Service Bus and Instrument Bus.

2. 250 volt D.C. power distribution, consisting of:

   a. RB MCC Nos. 2 and 3, and

   b. TB MCC Nos. 2 and 3.

3. For Unit 2, 125 volt D.C. power distribution, consisting of:

   a. TB Main Bus Nos. 2A-1and 3A,

   b. TB Res. Bus Nos. 2B and 2B-1,

   c. Reserve Bus No. 2, and

   d. RB Distribution Panel No. 2.

4.9 - SURVEILLANCE REQUIREMENTS

E. Distribution - Operating

Each of the required power distribution system divisions shall be determined energized at least once per 7 days by verifying correct breaker alignment and voltage on the busses/MCCs/panels.

3.9 - LIMITING CONDITIONS FOR OPERATION

E. Distribution - Operating

The following power distribution systems shall be energized ~~with tie breakers open both between redundant buses within the unit and between units at the same station~~:

1. A.C. power distribution, consisting of:

   a. Both Unit engineered safety features 4160 volt buses:

      1) For Unit 1, Nos. 13-1 and 14-1,

      2) For Unit 2, Nos. 23-1 and 24-1.

   b. Both Unit engineered safety features 480 volt buses:

      1) For Unit 1, Nos. 18 and 19,

      2) For Unit 2, Nos. 28 and 29, and

   c. The Unit 120 volt Essential Service Bus and Instrument Bus.

2. 250 volt D.C. power distribution, consisting of:

   a. TB MCC Nos. 1 and 2, and

   b. 1) For Unit 1, RB MCC Nos. 1A and 1B,

      2) For Unit 2, RB MCC Nos. 2A and 2B.

3. For Unit 1, 125 volt D.C. power distribution, consisting of:

   a. TB Main Bus Nos. 1A, 1A-1 and 2A,

   b. TB Reserve Bus Nos. 1B and 1B-1, and

   c. RB Distribution Panel No. 1.

4.9 - SURVEILLANCE REQUIREMENTS

E. Distribution - Operating

Each of the required power distribution system divisions shall be determined energized at least once per 7 days by verifying correct breaker alignment and voltage on the busses/MCCs/panels.

3.9 - LIMITING CONDITIONS FOR OPERATION

4. For Unit 3, 125 volt D.C. power distribution, consisting of:

    a. TB Main Bus Nos. 2A-1, 3A and 3A-1,

    b. TB Res. Bus Nos. 3B and 3B-1, and

    c. RB Distribution Panel No. 3.

5. 24/48 volt D.C. power distribution, consisting of:

    a. For Unit 2, Bus Nos. 2A and 2B.

    b. For Unit 3, Bus Nos. 3A and 3B.

APPLICABILITY:

OPERATIONAL MODE(s) 1, 2, and 3.

ACTIONS:

1. With one of the above required A.C. distribution systems not energized, re-energize the system within 8 hours or be in at least Hot Shutdown within the next 12 hours and in Cold Shutdown within the following 24 hours. CAPS CAPS

2. With one of the above required D.C. distribution systems not energized, re-energize the system within 2 hours or be in at least Hot Shutdown within the next 12 hours and in Cold Shutdown within the following 24 hours. CAPS CAPS

4.9 - SURVEILLANCE REQUIREMENTS

3.9 - LIMITING CONDITIONS FOR OPERATION

4. For Unit 2, 125 volt D.C. power distribution, consisting of:

   a. TB Main Bus Nos. 1A, 2A and 2A-1,

   b. TB Reserve Bus Nos. 2B and 2B-1, and

   c. RB Distribution Panel No. 2.

APPLICABILITY:

OPERATIONAL MODE(s) 1, 2, and 3.

ACTIONS:

1. With one of the above required A.C. distribution systems not energized, re-energize the system within 8 hours or be in at least Hot Shutdown within the next 12 hours and in Cold Shutdown within the following 24 hours. CAPS CAPS

2. With one of the above required D.C. distribution systems not energized, re-energize the system within 2 hours or be in at least Hot Shutdown within the next 12 hours and in Cold Shutdown within the following 24 hours. CAPS CAPS

4.9 - SURVEILLANCE REQUIREMENTS

BASES

The initial conditions of design basis transient and accident analyses assume Engineering Safety Features (ESF) systems are OPERABLE. The A.C. and D.C. electrical power sources are designed to provide sufficient capacity, capability, redundancy, and reliability to ensure the availability of necessary power to ESF systems so that the fuel, reactor coolant system and containment design limits are not exceeded.

The A.C. and D.C. sources are designed to permit inspection and testing of all important areas and features, especially those that have a standby function. Periodic component tests are supplemented by extensive functional testing during refueling outages under simulated accident conditions.

3/4.9.A A.C. Sources - Operating

The OPERABILITY of the A.C. electrical power sources is consistent with the initial assumptions of the accident analyses and is based upon meeting the design basis of the plant. This includes maintaining at least one of the onsite or offsite A.C. sources, D.C. power sources and associated distribution systems OPERABLE during accident conditions concurrent with an assumed loss of all offsite power and a worst-case single failure.

There are two sources of electrical energy available, i.e., the offsite transmission system and the onsite diesel generators. Two unit reserve auxiliary transformers are available to supply the Station class 1E distribution system. The reserve auxiliary transformer is sized to carry 100% of the auxiliary load. If this reserve auxiliary transformer (the normal circuit) is lost, auxiliary power from the other unit can be obtained for one division through a ~~the~~ 4160 volt bus tie (the alternate circuit). Additionally, two diesel generators are available to handle an accident. The allowable outage time takes into account the capacity and capability of the remaining A.C. sources, reasonable time for repairs, and the low probability of a design basis accident occurring during this period. Surveillance is required to ensure a highly reliable power source and no common cause failure mode for the remaining required offsite A.C. source.

Upon failure of one diesel generator, performance of appropriate surveillance requirements ensures a highly reliable power supply by checking the availability of the required offsite circuits, and the remaining required diesel generator. The initial surveillance is required to be completed regardless of how long the diesel inoperability persists, since the intent is that all diesel generator inoperabilities must be investigated for common cause failures. After the initial surveillance, an additional start test is required approximately mid-way through the allowed outage time to demonstrate continued OPERABILITY of the available onsite power sources. The diesel generator surveillance is limited to the normal start testing, since for cases in which less than a full complement of A.C. sources may be available, paralleling of two of the remaining A.C. sources may compromise the A.C. source independence. Additionally, the action provisions ensure that continued plant operation is not allowed when a complete loss of a required safety function (i.e., certain required components) would occur upon a loss of offsite power. These certain components which are critical to accomplishment of the required safety functions may be identified in advance and administratively controlled and/or evaluated on a case-by-case basis. With suitable

BASES

The initial conditions of design basis transient and accident analyses assume Engineering Safety Features (ESF) systems are OPERABLE. The A.C. and D.C. electrical power sources are designed to provide sufficient capacity, capability, redundancy, and reliability to ensure the availability of necessary power to ESF systems so that the fuel, reactor coolant system and containment design limits are not exceeded.

The A.C. and D.C. sources are designed to permit inspection and testing of all important areas and features, especially those that have a standby function. Periodic component tests are supplemented by extensive functional testing during refueling outages under simulated accident conditions.

3/4.9.A A.C. Sources - Operating

The OPERABILITY of the A.C. electrical power sources is consistent with the initial assumptions of the accident analyses and is based upon meeting the design basis of the plant. This includes maintaining at least one of the onsite or offsite A.C. sources, D.C. power sources and associated distribution systems OPERABLE during accident conditions concurrent with an assumed loss of all offsite power and a worst-case single failure.

There are two sources of electrical energy available, i.e., the offsite transmission system and the onsite diesel generators. Two unit reserve auxiliary transformers are available to supply the Station class 1E distribution system. The reserve auxiliary transformer is sized to carry 100% of the auxiliary load. If this reserve auxiliary transformer (the normal circuit) is lost, auxiliary power from the other unit can be obtained for one division through ~~the~~ a 4160 volt bus tie (the alternate circuit). Additionally, two diesel generators are available to handle an accident. The allowable outage time takes into account the capacity and capability of the remaining A.C. sources, reasonable time for repairs, and the low probability of a design basis accident occurring during this period. Surveillance is required to ensure a highly reliable power source and no common cause failure mode for the remaining required offsite A.C. source.

Upon failure of one diesel generator, performance of appropriate surveillance requirements ensures a highly reliable power supply by checking the availability of the required offsite circuits, and the remaining required diesel generator. The initial surveillance is required to be completed regardless of how long the diesel inoperability persists, since the intent is that all diesel generator inoperabilities must be investigated for common cause failures. After the initial surveillance, an additional start test is required approximately mid-way through the allowed outage time to demonstrate continued OPERABILITY of the available onsite power sources. The diesel generator surveillance is limited to the normal start testing, since for cases in which less than a full complement of A.C. sources may be available, paralleling of two of the remaining A.C. sources may compromise the A.C. source independence. Additionally, the action provisions ensure that continued plant operation is not allowed when a complete loss of a required safety function (i.e., certain required components) would occur upon a loss of offsite power. These certain components which are critical to accomplishment of the required safety functions may be identified in advance and administratively controlled and/or evaluated on a case-by-case basis.

BASES

(7/93)

Surveillance Requirements are also provided for demonstrating the OPERABILITY of the diesel generators. The specified testing is based on the guidance provided in Regulatory Guide 1.9, ~~draft~~ e Revision 3 ~~(1/91)~~, Regulatory Guide 1.108, Revision 1, and Regulatory Guide 1.137, Revision 1, as modified by plant specific analysis, diesel generator manufacturer recommendations and responses to Generic Letter 84-15.

/vendor

The diesel generators are equipped with a prelubrication system which maintains a continuous flow of oil to the diesel engine moving parts while the engine is shutdown. The purpose of this system is to increase long term diesel generator reliability by reducing the stress and wear caused by frequent dry starting of the diesel generator. The diesel generator prelube may be accomplished either through normal operation of the installed prelubrication system or by manual prelubrication of the diesel generator in accordance with the manufacturer's instructions. Performance of an idle start of the diesel generator is not considered to be a means of prelubrication.

/vendor's

A periodic "start test" of the diesel generators demonstrates proper startup from standby conditions, and verifies that the required generator voltage and frequency is attained. For this test, the diesel generator may be slow started and reach rated speed on a prescribed schedule that is selected to minimize stress and wear. In cases where this Surveillance Requirement is being used to identify a possible common ~~cause~~ failure ~~modes~~ in accordance with the action provisions, this test eliminates the risk of paralleling two of the remaining A.C. sources, which may compromise the A.C. source independence.

mode

A "load-run test" normally follows the periodic "start test" of the diesel generator to demonstrate operation at or near the continuous rating. This surveillance should only be conducted on one diesel generator at a time in order to avoid common ~~cause~~ failures that might result from offsite circuit or grid perturbations. A minimum run time of 60 minutes is required to stabilize engine temperatures. Actual run time should be in accordance with vendor recommendations with regard to good operating practice and should be sufficient to ensure that cooling and lubrication are adequate for extended periods of operation, while minimizing the time that the diesel generator is connected to the offsite source. This Surveillance Requirement may include gradual loading, as recommended by the manufacturer, so that mechanical stress and wear on the diesel engine are minimized. A load band is provided to avoid routine overloading of the diesel generators. Momentary transients outside the load band because of changing bus loads do not impact the validity of this test.

A periodic surveillance requirement is provided to assure the diesel generator is aligned to provide standby power on demand. Periodic surveillance requirements also verify that, without the aid of the refill compressor, sufficient air start capacity for each diesel generator is available. With either pair of air receiver tanks at the minimum specified pressure, there is sufficient air in the tanks to start the associated diesel generator.

Surveillance requirements provide verification that there is an adequate inventory of fuel oil in the storage tanks that is sufficient to provide time to place the facility in a safe shutdown condition and to bring in replenishment fuel from an offsite location. Additional diesel fuel can normally be obtained and delivered to the site within an eight hour period; thus a two day supply provides for

'INSERT'

INSERT

The periodicity of surveillance requirements for the shared diesel generators shall be equivalent to those required for the unit diesel generators. For example, it is not the intention to perform surveillances for the shared diesel generators twice during the specified surveillance interval in order to satisfy each unit's diesel generator surveillance requirements. By appropriately staggering the surveillance intervals between all three (3) diesel generators further ensures that for any loaded diesel generator surveillances, not more than one diesel generator is rendered inoperable at any given time in order to perform such testing.

BASES

(7/93)

Surveillance Requirements are also provided for demonstrating the OPERABILITY of the diesel generators. The specified testing is based on the guidance provided in Regulatory Guide 1.9, ~~draft~~ Revision 3 ~~(1/91)~~, Regulatory Guide 1.108, Revision 1, and Regulatory Guide 1.137, Revision 1, as modified by plant specific analysis, diesel generator manufacturer recommendations and responses to Generic Letter 84-15.

/vendor

The diesel generators are equipped with a prelubrication system which maintains a continuous flow of oil to the diesel engine moving parts while the engine is shutdown. The purpose of this system is to increase long term diesel generator reliability by reducing the stress and wear caused by frequent dry starting of the diesel generator. The diesel generator prelube may be accomplished either through normal operation of the installed prelubrication system or by manual prelubrication of the diesel generator in accordance with the manufacturer's instructions. Performance of an idle start of the diesel generator is not considered to be a means of prelubrication.

/vendor's

A periodic "start test" of the diesel generators demonstrates proper startup from standby conditions, and verifies that the required generator voltage and frequency is attained. For this test, the diesel generator may be slow started and reach rated speed on a prescribed schedule that is selected to minimize stress and wear. In cases where this Surveillance Requirement is being used to identify a possible common ~~cause~~ failure ~~modes in~~ accordance with the action provisions, this test eliminates the risk of paralleling two of the remaining A.C. sources, which may compromise the A.C. source independence.

mode

A "load-run test" normally follows the periodic "start test" of the diesel generator to demonstrate operation at or near the continuous rating. This surveillance should only be conducted on one diesel generator at a time in order to avoid common ~~cause~~ failures that might result from offsite circuit or grid perturbations. A minimum run time of 60 minutes is required to stabilize engine temperatures. Actual run time should be in accordance with vendor recommendations with regard to good operating practice and should be sufficient to ensure that cooling and lubrication are adequate for extended periods of operation, while minimizing the time that the diesel generator is connected to the offsite source. This Surveillance Requirement may include gradual loading, as recommended by the manufacturer, so that mechanical stress and wear on the diesel engine are minimized. A load band is provided to avoid routine overloading of the diesel generators. Momentary transients outside the load band because of changing bus loads do not impact the validity of this test.

A periodic surveillance requirement is provided to assure the diesel generator is aligned to provide standby power on demand. Periodic surveillance requirements also verify that, without the aid of the refill compressor, sufficient air start capacity for each diesel generator is available. With either pair of air receiver tanks at the minimum specified pressure, there is sufficient air in the tanks to start the associated diesel generator.

~~Surveillance requirements provide verification that there is an adequate inventory of fuel oil in the~~ storage tanks that is sufficient to provide time to place the facility in a safe shutdown condition and to bring in replenishment fuel from an offsite location. Additional diesel fuel can normally be obtained and delivered to the site within an eight hour period; thus a two day supply provides for

'INSERT'

INSERT

The periodicity of surveillance requirements for the shared diesel generators shall be equivalent to those required for the unit diesel generators. For example, it is not the intention to perform surveillances for the shared diesel generators twice during the specified surveillance interval in order to satisfy each unit's diesel generator surveillance requirements. By appropriately staggering the surveillance intervals between all three (3) diesel generators further ensures that for any loaded diesel generator surveillances, not more than one diesel generator is rendered inoperable at any given time in order to perform such testing.

BASES

Verifying an acceptable average temperature of ~~representative~~ battery cells is consistent with the recommendations of IEEE-450 and ensures that lower than normal temperatures do not act to inhibit or reduce battery capacity.

Verifying that the chargers will provide the manufacturer's rated current and voltage for four hours ensures that charger deterioration has not occurred and that the charger will provide the necessary capacity to restore the battery to a fully charged state.

A battery service test is a special test of the battery's capability "as found" to satisfy the design requirements of the D.C. electrical power system. The discharge rate and test length should correspond to the design duty cycle requirements.

A battery performance test is a test of constant current capacity of the battery to detect any change in the capacity determined by the acceptance test. This test is intended to determine overall battery degradation due to age and usage. A battery capacity of 80% indicates that the battery rate of deterioration is increasing, even if there is ample capacity to meet the load requirements. However, if the design margins are more limiting, the acceptable limit is based on the latest load profile.

'INSERT'

3/4.9.D D.C. Sources - Shutdown

The D.C. sources required to be OPERABLE during Cold Shutdown, Refueling, when handling irradiated fuel and during operations with a potential for draining the reactor vessel provide assurance that:

1. Systems to provide adequate coolant inventory makeup are available for the irradiated fuel in the core in case of an inadvertent drain down of the reactor vessel;

2. Systems needed to mitigate a fuel-handling accident are available;

3. Systems necessary to mitigate the effects of events that can lead to core damage during shutdown are OPERABLE;

4. Instrumentation and control capability is available for monitoring and maintaining the unit in a cold shutdown condition and refueling condition.

With one or more of the required D.C. electrical power sources inoperable, the action provisions require a suspension of activities that will preclude the occurrence of actions that could potentially initiate the postulated events. However, timely suspension of these activities is not intended to preclude completion of actions necessary to establish a safe, conservative condition.

INSERT

[Dresden - DC Operating]

A battery modified performance test is a test of the battery capacity and the battery's ability to meet the loads that exceed the constant current discharge rate of the battery (high rate short duration loads) of the battery's duty cycle. This test satisfies the requirements of both a service test and a performance test and is intended to detect any change in capacity and to determine overall battery degradation due to age and usage. The batteries have a rated capacity of 125% of the load expected at the end of their service life allowing for a minimum battery capacity of at least 80% of the manufacturer's rating. A battery capacity of 80% indicates that the battery rate of deterioration is increasing, even if there is ample capacity to meet the load requirements.

BASES

battery

Verifying an acceptable average temperature of ~~representative~~ cells is consistent with the recommendations of IEEE-450 and ensures that lower than normal temperatures do not act to inhibit or reduce battery capacity.

Verifying that the chargers will provide the manufacturer's rated current and voltage for four hours ensures that charger deterioration has not occurred and that the charger will provide the necessary capacity to restore the battery to a fully charged state.

A battery service test is a special test of the battery's capability "as found" to satisfy the design requirements of the D.C. electrical power system. The discharge rate and test length should correspond to the design duty cycle requirements.

A battery performance test is a test of constant current capacity of the battery to detect any change in the capacity determined by the acceptance test. This test is intended to determine overall battery degradation due to age and usage. A battery capacity of 80% indicates that the battery rate of deterioration is increasing, even if there is ample capacity to meet the load requirements. However, if the design margins are more limiting, the acceptable limit is based on the latest load profile.

'INSERT'

3/4.9.D D.C. Sources - Shutdown

The D.C. sources required to be OPERABLE during Cold Shutdown, Refueling, when handling irradiated fuel and during operations with a potential for draining the reactor vessel provide assurance that:

1. Systems to provide adequate coolant inventory makeup are available for the irradiated fuel in the core in case of an inadvertent drain down of the reactor vessel;

2. Systems needed to mitigate a fuel-handling accident are available;

3. Systems necessary to mitigate the effects of events that can lead to core damage during shutdown are OPERABLE;

4. Instrumentation and control capability is available for monitoring and maintaining the unit in a cold shutdown condition and refueling condition.

With one or more of the required D.C. electrical power sources inoperable, the action provisions require a suspension of activities that will preclude the occurrence of actions that could potentially initiate the postulated events. However, timely suspension of these activities is not intended to preclude completion of actions necessary to establish a safe, conservative condition.

INSERT

[Quad Cities - DC Operating]

A battery modified performance test is a test of the battery capacity and the battery's ability to meet the loads that exceed the constant current discharge rate of the battery (high rate short duration loads) of the battery's duty cycle. This test satisfies the requirements of both a service test and a performance test and is intended to detect any change in capacity and to determine overall batery degradation due to age and usage. The 125 volt batteries have a rated capacity of 125% of the load expected at the end of their service life allowing for a minimum battery capacity of at least 80% of the manufacturer's rating. A battery capacity of 80% indicates that the battery rate of deterioration is increasing, even if there is ample capacity to meet the load requirements. The 250 volt batteries do not have a rated capacity of 125% of the load expected at the end of their service therefore, the minimum allowable battery capacity is based on the capacity margin calculated from the design load profile for the battery.

3.10 - LIMITING CONDITIONS FOR OPERATION

H. Water Level - Spent Fuel Storage Pool

At least 23 feet of water shall be maintained over the top of irradiated fuel assemblies seated in the spent fuel storage pool racks.

The pool water level shall be maintained at a level of 33 feet.

APPLICABILITY:

Whenever irradiated fuel assemblies are in the spent fuel storage pool.

ACTION:

With the requirements of the above specification not satisfied, suspend all movement of fuel assemblies and crane operations with loads in the spent fuel storage pool area after placing the fuel assemblies and crane load in a safe condition. The provisions of Specification 3.0.C are not applicable.

4.10 - SURVEILLANCE REQUIREMENTS

H. Water Level - Spent Fuel Storage Pool

The water level in the spent fuel storage pool shall be determined to be at least at its minimum required depth at least once per 7 days.

3.10 - LIMITING CONDITIONS FOR OPERATION

H. Water Level - Spent Fuel Storage Pool

At least 23 feet of water shall be maintained over the top of irradiated fuel assemblies seated in the spent fuel storage pool racks.

APPLICABILITY:

Whenever irradiated fuel assemblies are in the spent fuel storage pool.

ACTION:

With the requirements of the above specification not satisfied, suspend all movement of fuel assemblies and crane operations with loads in the spent fuel storage pool area after placing the fuel assemblies and crane load in a safe condition. The provisions of Specification 3.0.C are not applicable.

4.10 - SURVEILLANCE REQUIREMENTS

H. Water Level - Spent Fuel Storage Pool

The water level in the spent fuel storage pool shall be determined to be at least at its minimum required depth at least once per 7 days.

The pool water level shall be maintained at a level of 33 feet.

TRANSIENT LINEAR HEAT GENERATION RATE

3.11 - LIMITING CONDITIONS FOR OPERATION

B. ~~Average Power Range Monitor Setpoints~~

The Average Power Range Monitor (APRM) gain or setpoints shall be set such that the FUEL DESIGN LIMITING RATIO FOR CENTERLINE MELT (FDLRC) shall be less than or equal to 1.0.

APPLICABILITY:

OPERATIONAL MODE 1, when THERMAL POWER is greater than or equal to 25% of RATED THERMAL POWER.

ACTION:

With FDLRC greater then 1.0, initiate corrective ACTION within 15 minutes and within 6 hours either:

1. Restore FDLRC to less than or equal to 1.0 ~~within its limit~~, or
2. Adjust the flow biased APRM setpoints specified in Specifications 2.2.A and 3.2.E by 1/FDLRC~~, or~~
3. Adjust[a] each ~~the~~ APRM gain such that the APRM readings are ≥100% times ~~of the~~ FRACTION OF RATED THERMAL POWER (FRTP) times FDLRC.

With the provisions of the ACTION above not met, reduce THERMAL POWER to less than 25% of RATED THERMAL POWER within the next 4 hours.

4.11 - SURVEILLANCE REQUIREMENTS

B. ~~Average Power Range Monitor Setpoints~~

The value of FDLRC shall be verified:

1. At least once per 24 hours,
2. Within 12 hours after completion of a THERMAL POWER increase of at least 15% of RATED THERMAL POWER, and
3. Initially and at least once per 12 hours when the reactor is operating with FDLRC greater than or equal to 1.0.
4. The provisions of Specification 4.0.D are not applicable.

'INSERT'

a Provided that the adjusted APRM reading does not exceed 100% of RATED THERMAL POWER and a notice of adjustment is posted on the reactor control panel.

INSERT

The TRANSIENT LINEAR HEAT GENERATION RATE (TLHGR) shall be maintained such that the FUEL DESIGN LIMITING RATIO for CENTERLINE MELT (FDLRC) is less than or equal to 1.0. Where FDLRC is equal to:

$$\frac{(LHGR)\,(1.2)}{(TLHGR)\,(FRTP)}$$

3.11 - LIMITING CONDITIONS FOR OPERATION

C. MINIMUM CRITICAL POWER RATIO

The MINIMUM CRITICAL POWER RATIO (MCPR) shall be equal to or greater than the MCPR operating limit specified in the CORE OPERATING LIMITS REPORT.

APPLICABILITY:

OPERATIONAL MODE 1, when THERMAL POWER is greater than or equal to 25% of RATED THERMAL POWER.

ACTION:

With MCPR less than the applicable MCPR limit as determined for one of the conditions specified in the CORE OPERATING LIMITS REPORT:

1. Initiate corrective ACTION within 15 minutes, and
2. Restore MCPR to within the required limit within 2 hours.

With the provisions of the ACTION above not met, reduce THERMAL POWER to less than 25% of RATED THERMAL POWER within the next 4 hours.

4.11 - SURVEILLANCE REQUIREMENTS

C. MINIMUM CRITICAL POWER RATIO

MCPR, with:

1. t_{ave} = 3.50 prior to performance of the initial scram time measurements for the cycle in accordance with Specification 4.3.D, or
2. t_{ave} determined within 72 hours of the conclusion of each scram time surveillance test required by Specification 4.3.D,

shall be determined to be equal to or greater than the applicable MCPR limit specified in the CORE OPERATING LIMITS REPORT.

1. At least once per 24 hours,
2. Within 12 hours after completion of a THERMAL POWER increase of at least 15% of RATED THERMAL POWER, and
3. Initially and at least once per 12 hours when the reactor is operating with a LIMITING CONTROL ROD PATTERN for MCPR.
4. The provisions of Specification 4.0.D are not applicable.

3.11 - LIMITING CONDITIONS FOR OPERATION

C. MINIMUM CRITICAL POWER RATIO

The MINIMUM CRITICAL POWER RATIO (MCPR) shall be equal to or greater than the MCPR operating limit specified in the CORE OPERATING LIMITS REPORT.

APPLICABILITY:

OPERATIONAL MODE 1, when THERMAL POWER is greater than or equal to 25% of RATED THERMAL POWER.

ACTION:

With MCPR less than the applicable MCPR operating limit as determined for one of the conditions specified in the CORE OPERATING LIMITS REPORT:

1. Initiate corrective ACTION within 15 minutes, and
2. Restore MCPR to within the required limit within 2 hours.

With the provisions of the ACTION above not met, reduce THERMAL POWER to less than 25% of RATED THERMAL POWER within the next 4 hours.

4.11 - SURVEILLANCE REQUIREMENTS

C. MINIMUM CRITICAL POWER RATIO

MCPR, with:

1. t_{ave} = 0.86 prior to performance of the initial scram time measurements for the cycle in accordance with Specification 4.3.D, or
2. t_{ave} determined within 72 hours of the conclusion of each scram time surveillance test required by Specification 4.3.D,

shall be determined to be equal to or greater than the applicable MCPR operating limit specified in the CORE OPERATING LIMITS REPORT.

1. At least once per 24 hours,
2. Within 12 hours after completion of a THERMAL POWER increase of at least 15% of RATED THERMAL POWER, and
3. Initially and at least once per 12 hours when the reactor is operating with a LIMITING CONTROL ROD PATTERN for MCPR.
4. The provisions of Specification 4.0.D are not applicable.

3.11 - LIMITING CONDITIONS FOR OPERATION

D. STEADY STATE LINEAR HEAT GENERATION RATE

The STEADY STATE LINEAR HEAT GENERATION RATE (SLHGR) for each type of fuel as a function of AVERAGE PLANAR EXPOSURE shall not exceed the limits specified in the CORE OPERATING LIMITS REPORT.

LINEAR HEAT GENERATION RATE (LHGR)

APPLICABILITY:

OPERATIONAL MODE 1, when THERMAL POWER is greater than or equal to 25% of RATED THERMAL POWER.

ACTION:

SLHGR

With a SLHGR exceeding the limits specified in the CORE OPERATING LIMITS REPORT:

1. Initiate corrective ACTION within 15 minutes, and
2. Restore the SLHGR to within the ~~required~~ limit within 2 hours.

SLHGR

With the provisions of the ACTION above not met, reduce THERMAL POWER to less than 25% of RATED THERMAL POWER within the next 4 hours.

4.11 - SURVEILLANCE REQUIREMENTS

D. STEADY STATE LINEAR HEAT GENERATION RATE

The SLHGR shall be determined to be equal to or less than the limit:

1. At least once per 24 hours,
2. Within 12 hours after completion of a THERMAL POWER increase of at least 15% of RATED THERMAL POWER, and
3. Initially and at least once per 12 hours when the reactor is operating with a LIMITING CONTROL ROD PATTERN for SLHGR.
4. The provisions of Specification 4.0.D are not applicable.

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3.11 - LIMITING CONDITIONS FOR OPERATION

E. TRANSIENT LINEAR HEAT GENERATION RATE

The TRANSIENT LINEAR HEAT GENERATION RATE (TLHGR) for each type of fuel as a function of AVERAGE PLANAR EXPOSURE shall not exceed the limits specified in the CORE OPERATING LIMITS REPORT.

APPLICABILITY:

OPERATIONAL MODE 1, when THERMAL POWER is greater than or equal to 25% of RATED THERMAL POWER.

ACTION:

With a TLHGR exceeding the limits specified in the CORE OPERATING LIMITS REPORT:

1. Initiate corrective ACTION within 15 minutes, and
2. Restore the TLHGR to within the required limit within 2 hours.

With the provisions of the ACTION above not met, reduce THERMAL POWER to less than 25% of RATED THERMAL POWER within the next 4 hours.

4.11 - SURVEILLANCE REQUIREMENTS

E. TRANSIENT LINEAR HEAT GENERATION RATE

The TLHGR shall be determined to be equal to or less than the limit:

1. At least once per 24 hours,
2. Within 12 hours after completion of a THERMAL POWER increase of at least 15% of RATED THERMAL POWER, and
3. Initially and at least once per 12 hours when the reactor is operating with a LIMITING CONTROL ROD PATTERN for TLHGR.
4. The provisions of Specification 4.0.D are not applicable.

BASES

3/4.11.A AVERAGE PLANAR LINEAR HEAT GENERATION RATE

This specification assures that the peak cladding temperature following the postulated design basis loss-of-coolant accident will not exceed the limit specified in 10 CFR 50.46. The specification also assures that fuel rod mechanical integrity is maintained during normal and transient operations.

The peak cladding temperature (PCT) following a postulated loss-of-coolant accident is primarily a function of the average heat generation rate of all the rods of a fuel assembly at any axial location and is dependent only secondarily on the rod-to-rod power distribution within an assembly. The peak clad temperature is calculated assuming a LINEAR HEAT GENERATION RATE (LHGR) for the highest powered rod which is equal to or less than the design LHGR corrected for densification. The APLHGR limits specified are equivalent to the LHGR of the highest powered fuel rod assumed in the LOCA analysis divided by its local peaking factor. A conservative multiplier is applied to the LHGR assumed in the LOCA analysis to account for the uncertainty associated with the measurement of the APLHGR.

The calculational procedure used to establish the maximum APLHGR values uses NRC approved calculational models which are consistent with the requirements of Appendix K of 10 CFR Part 50. The approved calculational models are listed in Specification ~~6.6.A.4~~. 6.9

The daily requirement for calculating APLHGR when THERMAL POWER is greater than or equal to 25% of RATED THERMAL POWER is sufficient since power distribution shifts are very slow when there have not been significant power or control rod changes. The requirement to calculate APLHGR within 12 hours after the completion of a THERMAL POWER increase of at least 15% of RATED THERMAL POWER ensures thermal limits are met after power distribution shifts while still allotting time for the power distribution to stabilize. The requirement for calculating APLHGR after initially determining a LIMITING CONTROL ROD PATTERN exists ensures that APLHGR will be known following a change in THERMAL POWER or power shape, that could place operation above a thermal limit.

3/4.11.B ~~APRM SETPOINTS~~ TRANSIENT LINEAR HEAT GENERATION RATE

~~The fuel cladding integrity Safety Limits of Specification 2.1 were based on a power distribution which would yield the design LHGR at RATED THERMAL POWER.~~ The flow biased neutron flux - high scram setting and control rod block functions of the APRM instruments for both two recirculation loop operation and single recirculation loop operation must be adjusted to ensure that ~~the MCPR does not become less than the fuel cladding safety limit or that~~ ≥1% plastic strain does not occur ~~in the degraded situation~~. The scram settings and rod block settings are adjusted in accordance with the formula in this specification when the value of FDLRC indicates a higher peaked power distribution to ensure that an LHGR transient would not be increased in the degraded condition.

'INSERT'

'INSERT'

INSERT

and the fuel does not experience centerline melt during anticipated operational occurrences beginning at any power and terminating at 120% of rated core thermal power.

INSERT

The daily requirement for calculating FDLRC when THERMAL POWER is greater than or equal to 25% of RATED THERMAL POWER is sufficient since power distribution shifts are very slow when there have not been significant power or control rod changes. The requirement to calculate FDLRC within 12 hours after the completion of a THERMAL POWER increase of at least 15% of RATED THERMAL POWER ensures thermal limits are met after power distribution shifts while still allotting time for the power distribution to stabilize. The requirement for calculating FDLRC after initially determining FDLRC is greater than 1.0 exists to ensure that FDLRC will be known following a change in THERMAL POWER or power shape that could place operation above a thermal limit.

The FUEL DESIGN LIMIT RATIO FOR CENTERLINE MELT (FDLRC) is defined as:

$$\text{FDLRC} = \frac{(\text{LHGR})(1.2)}{(\text{TLHGR})(\text{FRTP})} ;$$

where LHGR is the LINEAR HEAT GENERATION RATE, and TLHGR is the TRANSIENT LINEAR HEAT GENERATION RATE. The TLHGR is specified in the CORE OPERATING LIMITS REPORT.

BASES

3/4.11.A AVERAGE PLANAR LINEAR HEAT GENERATION RATE

This specification assures that the peak cladding temperature following the postulated design basis loss-of-coolant accident will not exceed the limit specified in 10 CFR 50.46. The specification also assures that fuel rod mechanical integrity is maintained during normal and transient operations.

The peak cladding temperature (PCT) following a postulated loss-of-coolant accident is primarily a function of the average heat generation rate of all the rods of a fuel assembly at any axial location and is dependent only secondarily on the rod-to-rod power distribution within an assembly. The peak clad temperature is calculated assuming a LINEAR HEAT GENERATION RATE (LHGR) for the highest powered rod which is equal to or less than the design LHGR corrected for densification. The APLHGR limits specified are equivalent to the LHGR of the highest powered fuel rod assumed in the LOCA analysis divided by its local peaking factor. A conservative multiplier is applied to the LHGR assumed in the LOCA analysis to account for the uncertainty associated with the measurement of the APLHGR.

The calculational procedure used to establish the maximum APLHGR values uses NRC approved calculational models which are consistent with the requirements of Appendix K of 10 CFR Part 50. The approved calculational models are listed in Specification ~~6.6.A.4~~. 6.9

The daily requirement for calculating APLHGR when THERMAL POWER is greater than or equal to 25% of RATED THERMAL POWER is sufficient since power distribution shifts are very slow when there have not been significant power or control rod changes. The requirement to calculate APLHGR within 12 hours after the completion of a THERMAL POWER increase of at least 15% of RATED THERMAL POWER ensures thermal limits are met after power distribution shifts while still allotting time for the power distribution to stabilize. The requirement for calculating APLHGR after initially determining a LIMITING CONTROL ROD PATTERN exists ensures that APLHGR will be known following a change in THERMAL POWER or power shape, that could place operation above a thermal limit.

3/4.11.B APRM SETPOINTS

The fuel cladding integrity Safety Limits of Specification 2.1 were based on a power distribution which would yield the design LHGR at RATED THERMAL POWER. The flow biased neutron flux - high scram setting and control rod block functions of the APRM instruments for both two recirculation loop operation and single recirculation loop operation must be adjusted to ensure that the MCPR does not become less than the fuel cladding safety limit or that ≥1% plastic strain does not occur in the degraded situation. The scram settings and rod block settings are adjusted in accordance with the formula in this specification when the value of MFLPD indicates a higher peaked power distribution to ensure that an LHGR transient would not be increased in the degraded condition.

BASES

3/4.11.C MINIMUM CRITICAL POWER RATIO

The required operating limit MCPR at steady state operating conditions as specified in Specification 3.11.C are derived from the established fuel cladding integrity Safety Limit MCPR, and an analysis of abnormal operational transients. For any abnormal operating transient analysis evaluation with the initial condition of the reactor being at the steady state operating limit, it is required that the resulting MCPR does not decrease below the Safety Limit MCPR at any time during the transient assuming instrument trip setting given in Specification 2.2.

To assure that the fuel cladding integrity Safety Limit is not exceeded during any anticipated abnormal operational transient, the most limiting transients have been analyzed to determine which result in the largest reduction in the CRITICAL POWER RATIO (CPR). The type of transients evaluated were change of flow, increase in pressure and power, positive reactivity insertion, and coolant temperature decrease. The limiting transient yields the largest delta MCPR. When added to the Safety Limit MCPR, the required minimum operating limit MCPR of Specification 3.11.C is obtained and presented in the CORE OPERATING LIMITS REPORT.

The steady state values for MCPR specified were determined using NRC-approved methodology listed in specification ~~6.6.A.4~~ 6.9.

The purpose of the reduced flow MCPR curves specified in the CORE OPERATING LIMITS REPORT are to define MCPR operating limits at other than rated core flow conditions. The reduced flow MCPR curves assure that the Safety Limit MCPR will not be violated.

Since the transient analysis takes credit for conservatism in the scram speed performance, it must be demonstrated that the specific scram speed distribution is consistent with that used in the transient analysis. The 72 hour completion time is acceptable due to the relatively minor changes in t_{ave} expected during the fuel cycle.

At THERMAL POWER levels less than or equal to 25% of RATED THERMAL POWER, the reactor will be operating at minimum recirculation pump speed and the moderator void content will be very small. For all designated control rod patterns which may be employed at this point, operating plant experience indicates that the resulting MCPR value has considerable margin. Thus, the demonstration of MCPR below this power level is unnecessary. The daily requirement for calculating MCPR when THERMAL POWER is greater than or equal to 25% of RATED THERMAL POWER is sufficient since power distribution shifts are very slow when there have not been significant power or control rod changes. The requirement for calculating MCPR after initially determining that a LIMITING CONTROL ROD PATTERN exists ensures that MCPR will be known following a change in THERMAL POWER or power shape, regardless of magnitude, that could place operation above a thermal limit.

BASES

3/4.11.C MINIMUM CRITICAL POWER RATIO

The required operating limit MCPR at steady state operating conditions as specified in Specification 3.11.C are derived from the established fuel cladding integrity Safety Limit MCPR, and an analysis of abnormal operational transients. For any abnormal operating transient analysis evaluation with the initial condition of the reactor being at the steady state operating limit, it is required that the resulting MCPR does not decrease below the Safety Limit MCPR at any time during the transient assuming instrument trip setting given in Specification 2.2.

To assure that the fuel cladding integrity Safety Limit is not exceeded during any anticipated abnormal operational transient, the most limiting transients have been analyzed to determine which result in the largest reduction in the CRITICAL POWER RATIO (CPR). The type of transients evaluated were change of flow, increase in pressure and power, positive reactivity insertion, and coolant temperature decrease. The limiting transient yields the largest delta MCPR. When added to the Safety Limit MCPR, the required minimum operating limit MCPR of Specification 3.11.C is obtained and presented in the CORE OPERATING LIMITS REPORT.

The steady state values for MCPR specified were determined using NRC-approved methodology listed in Specification ~~6.6.A.4~~. 6.9

The purpose of the MCPR multiplicative factor specified in the CORE OPERATING LIMITS REPORT is to define MCPR operating limits at other than rated core flow conditions. At less than 100% of rated flow, the required MCPR is the product of the MCPR and the off rated flow MCPR multiplier factor. The MCPR multiplier assures that the Safety Limit MCPR will not be violated.

Since the transient analysis takes credit for conservatism in the scram speed performance, it must be demonstrated that the specific scram speed distribution is consistent with that used in the transient analysis. The 72 hour completion time is acceptable due to the relatively minor changes in t_{ave} expected during the fuel cycle.

At THERMAL POWER levels less than or equal to 25% of RATED THERMAL POWER, the reactor will be operating at minimum recirculation pump speed and the moderator void content will be very small. For all designated control rod patterns which may be employed at this point, operating plant experience indicates that the resulting MCPR value has considerable margin. Thus, the demonstration of MCPR below this power level is unnecessary. The daily requirement for calculating MCPR when THERMAL POWER is greater than or equal to 25% of RATED THERMAL POWER is sufficient since power distribution shifts are very slow when there have not been significant power or control rod changes. The requirement for calculating MCPR after initially determining that a LIMITING CONTROL ROD PATTERN exists ensures that MCPR will be known following a change in THERMAL POWER or power shape, regardless of magnitude, that could place operation above a thermal limit.

BASES

3/4.11.D STEADY STATE LINEAR HEAT GENERATION RATE

This specification assures that the maximum ~~STEADY STATE~~ LINEAR HEAT GENERATION RATE in any fuel rod is less than the design STEADY STATE LINEAR HEAT GENERATION RATE even if fuel pellet densification is postulated. This provides assurance that the fuel end-of-life steady state criteria are met. The daily requirement for calculating SLHGR when THERMAL POWER is greater than or equal to 25% of RATED THERMAL POWER is sufficient since power distributions shifts are very slow when there have not been significant power or control rod changes. The requirement to calculate SLHGR within 12 hours after the completion of a THERMAL POWER increase of at least 15% of RATED THERMAL POWER ensures thermal limits are met after power distribution shifts while still allotting time for the power distribution to stabilize. The requirement for calculating SLHGR after initially determining a LIMITING CONTROL ROD PATTERN exists ensures that SLHGR will be known following a change in THERMAL POWER or power shape that could place operation above a thermal limit.

3/4.11.E TRANSIENT LINEAR HEAT GENERATION RATE

This specification provides assurance that the fuel will neither experience centerline melt nor exceed 1% plastic strain for transient overpower events beginning at any power and terminating at 120% of RATED THERMAL POWER. The daily requirement for calculating TLHGR when THERMAL POWER is greater than or equal to 25% of RATED THERMAL POWER is sufficient since power distribution shifts are very slow when there have not been significant power or control rod changes. The requirement to calculate TLHGR within 12 hours after the completion of a THERMAL POWER increase of at least 15% of RATED THERMAL POWER ensures thermal limits are met after power distribution shifts while still allotting time for the power distribution to stabilize. The requirement for calculating TLHGR after initially determining a LIMITING CONTROL ROD PATTERN exists ensures that TLHGR will be known following a change in THERMAL POWER or power shape that could place operation above a thermal limit.

6.2.B Unit Staff

The unit staff shall include the following:

1. Three non-licensed operators shall be on site at all times.

2. At least one licensed Reactor Operator shall be present in the control room when fuel is in the reactor. In addition, while the unit is in MODE(s) 1, 2, 3 or 4 at least one licensed Senior Reactor Operator shall be present in the control room.

3. Shift crew composition may be less than the minimum requirement of 10 CFR 50.54(m)(2)(i) and 6.2.B.1 and 6.2.C for a period of time not to exceed two 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.

4. A Radiation Protection Technician shall be on site when fuel is in the reactor. The position may be vacant for not more than two hours, in order to provide for unexpected absence, provided immediate action is taken to fill the required position.

5. Administrative procedures shall be developed and implemented to limit the working hours of unit staff who perform safety-related functions; e.g, senior reactor operators, reactor operators, health physicists, auxiliary operators, and key maintenance personnel.

   The amount of overtime worked by unit staff members performing safety-related functions shall be limited in accordance with the NRC Policy Statement on working hours (Generic Letter 82-12). Any deviations from the guidelines of Generic Letter 82-12 shall be authorized in advance by the Station Manager or his designee, in accordance with approved administrative procedures, or by higher levels of management, in accordance with established procedures and with documentation of the basis for granting the deviation.

6. The Operations Manager or Shift Operations Supervisor shall hold a Senior Reactor Operator License.

6.2.C Shift Technical Advisor

The Shift Technical Advisor (STA) shall provide technical advisory support to the Unit Supervisor in the areas of thermal hydraulics, reactor engineering and plant analysis with regard to the safe operation of the facility. In addition, the STA shall meet the qualifications specified by the Commission Policy Statement on Engineering Expertise on Shift. A single STA may fulfill this function for both units.

6.8 PROCEDURES AND PROGRAMS

6.8.A Written procedures shall be established, implemented, and maintained covering the activities referenced below:

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

2. The Emergency Operating Procedures required to implement the requirements of NUREG-0737 and Supplement 1 to NUREG-0737 as stated in Section 7.1 of Generic Letter No. 82-33,

3. Station Security Plan implementation,

4. Generating Station Emergency Response Plan implementation,

5. PROCESS CONTROL PROGRAM (PCP) implementation,

6. OFFSITE DOSE CALCULATION MANUAL (ODCM) implementation, and

7. Fire Protection Program implementation.

6.8.B Deleted
6.8.C Deleted

6.8.~~B~~ e The following programs shall be established, implemented, and maintained:

D

1. Reactor Coolant Sources Outside Primary Containment

   This program provides controls to minimize leakage from those portions of systems outside primary containment that could contain highly radioactive fluids during a serious transient or accident to as low as practical levels. The systems include CS, HPCI, LPCI, IC, process sampling, containment monitoring, and standby gas treatment systems. The program shall include the following:

   a. Preventive maintenance and periodic visual inspection requirements, and

   b. Leak test requirements for each system at a frequency of at least once per operating cycle.

ADMINISTRATIVE CONTROLS

6.12.B In addition to the requirements of 6.12.A, areas accessible to personnel with radiation levels greater than 1000 mrem/hr at 30 cm (12 in.) from the radiation source or from any surface which the radiation penetrates shall require the following:

1. Doors shall be locked to prevent unauthorized entry and shall not prevent individuals from leaving the area. In place of locking the door, ~~continuous,~~ direct or electronic surveillance that is capable of preventing unauthorized entry may be used. The keys shall be maintained under the administrative control of the Shift Manager on duty and/or health physics supervision.

2. Personnel access and exposure control requirements of activities being performed within these areas shall be specified by an approved RWP(or equivalent document).

3. Each person entering the area shall be provided with an alarming radiation monitoring device that continuously integrates the radiation dose rate (such as an electronic dosimeter.) ~~Continuous~~ surveillance and radiation monitoring by a Radiation Protection Technician may be substituted for an alarming dosimeter.

4. During emergency situations which involve personnel injury or actions taken to prevent major equipment damage, ~~continuous~~ surveillance and radiation monitoring of the work area by a qualified individual may be substituted for the routine RWP (or equivalent document).

5. For individual HIGH RADIATION AREAS accessible to personnel with radiation levels of greater than 1000 mrem/h at 30 cm (12 in.) that are located within large areas where no enclosure exists for purposes of locking, and where no enclosure can be reasonably constructed around the individual areas, then such individual areas shall be barricaded, conspicuously posted, and a flashing light shall be activated as a warning device.

ATTACHMENT C

REVISED TSUP PAGES FOR
DRESDEN AND QUAD CITIES NUCLEAR POWER STATIONS
LICENSE NOS. DPR-19, DPR-25, DPR-29, AND DPR-30

DRESDEN STATION

UNITS 2 & 3

(DPR-19 & DPR-25)

DEFINITIONS

DEFINITIONS

SAFETY LIMITS and LIMITING SAFETY SYSTEM SETTINGS

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LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS

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Section 6 ADMINISTRATIVE CONTROLS

ADMINISTRATIVE CONTROLS

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1.0 DEFINITIONS

CHANNEL FUNCTIONAL TEST

A CHANNEL FUNCTIONAL TEST shall be:

a. Analog CHANNEL(s) - the injection of a simulated signal into the CHANNEL as close to the sensor as practicable to verify OPERABILITY including required alarm and/or trip functions and CHANNEL failure trips.

b. Bistable CHANNEL(s) - the injection of a simulated signal into the sensor to verify OPERABILITY including required alarm and/or trip functions.

The CHANNEL FUNCTIONAL TEST may be performed by any series of sequential, overlapping or total CHANNEL steps such that the entire CHANNEL is tested.

CORE ALTERATION

CORE ALTERATION shall be the movement of any fuel, sources, or reactivity control components, within the reactor vessel with the vessel head removed and fuel in the vessel. The following exceptions are not considered to be CORE ALTERATIONS:

a. Movement of source range monitors, local power range monitors, intermediate range monitors, traversing incore probes, or special movable detectors (including undervessel replacement); and

b. Control rod movement, provided there are no fuel assemblies in the associated control cell.

Suspension of CORE ALTERATIONS shall not preclude completion of movement of a component to a safe position.

CORE OPERATING LIMITS REPORT (COLR)

The CORE OPERATING LIMITS REPORT (COLR) shall be the unit specific document that provides core operating limits for the current operating cycle. These cycle specific core operating limits shall be determined for each operating cycle in accordance with Specification 6.9. Plant operation within these operating limits is addressed in individual specifications.

CRITICAL POWER RATIO (CPR)

The CRITICAL POWER RATIO (CPR) shall be the ratio of that power in the assembly which is calculated by application of the applicable NRC approved critical power correlation to cause some point in the assembly to experience transition boiling, divided by the actual assembly power.

DOSE EQUIVALENT I-131

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

1.0 DEFINITIONS

FRACTION OF RATED THERMAL POWER (FRTP)
The FRACTION OF RATED THERMAL POWER (FRTP) shall be the measured THERMAL POWER divided by the RATED THERMAL POWER.

FREQUENCY NOTATION
The FREQUENCY NOTATION specified for the performance of Surveillance Requirements shall correspond to the intervals defined in Table 1-1.

FUEL DESIGN LIMITING RATIO (FDLRX)
The FUEL DESIGN LIMITING RATIO (FDLRX) shall be the limit used to assure that the fuel operates within the end-of-life steady-state design criteria by, among other items, limiting the release of fission gas to the cladding plenum.

FUEL DESIGN LIMITING RATIO for CENTERLINE MELT (FDLRC)
The FUEL DESIGN LIMITING RATIO for CENTERLINE MELT (FDLRC) shall be the limit used to assure that the fuel will neither experience centerline melt nor exceed 1% plastic cladding strain for transient overpower events beginning at any power and terminating at 120% of RATED THERMAL POWER.

IDENTIFIED LEAKAGE
IDENTIFIED LEAKAGE shall be: a) leakage into primary containment collection systems, such as pump seal or valve packing leaks, that is captured and conducted to a sump or collecting tank, or b) leakage into the primary containment atmosphere from sources that are both specifically located and known either not to interfere with the operation of the leakage detection systems or not to be PRESSURE BOUNDARY LEAKAGE.

LIMITING CONTROL ROD PATTERN (LCRP)
A LIMITING CONTROL ROD PATTERN (LCRP) shall be a pattern which results in the core being on a thermal hydraulic limit, i.e., operating on a limiting value for APLHGR, LHGR, or MCPR.

LINEAR HEAT GENERATION RATE (LHGR)
LINEAR HEAT GENERATION RATE (LHGR) shall be the heat generation per unit length of fuel rod. It is the integral of the heat flux over the heat transfer area associated with the unit length.

LOGIC SYSTEM FUNCTIONAL TEST (LSFT)
A LOGIC SYSTEM FUNCTIONAL TEST (LSFT) shall be a test of all required logic components, i.e., all required relays and contacts, trip units, solid state logic elements, etc, of a logic circuit, from as close to the sensor as practicable up to, but not including the actuated device, to verify OPERABILITY. The LOGIC SYSTEM FUNCTIONAL TEST may be performed by means of any series of sequential, overlapping or total system steps so that the entire logic system is tested.

1.0 DEFINITIONS

MINIMUM CRITICAL POWER RATIO (MCPR)

The MINIMUM CRITICAL POWER RATIO (MCPR) shall be the smallest CPR which exists in the core.

OFFSITE DOSE CALCULATION MANUAL (ODCM)

The OFFSITE DOSE CALCULATION MANUAL (ODCM) shall contain the methodology and parameters used in the calculation of offsite doses resulting from radioactive gaseous and liquid effluents, in the calculation of gaseous and liquid effluent monitoring Alarm/Trip Setpoints, and in the conduct of the Environmental Radiological Monitoring Program. The ODCM shall also contain (1) the Radioactive Effluent Controls and Radiological Environmental Monitoring Programs required by Specification 6.8 and (2) descriptions of the information that should be included in the Annual Radiological Environmental Operating and Semi-annual Radioactive Effluent Release Reports required by Specification 6.9.

OPERABLE - OPERABILITY

A system, subsystem, train, component, or device shall be OPERABLE or have OPERABILITY when it is capable of performing its specified safety function(s) and when all necessary attendant instrumentation, controls, normal or emergency electrical power, cooling or seal water, lubrication or other auxiliary equipment that are required for the system, subsystem, train, component or device to perform its specified safety function(s) are also capable of performing their related support function(s).

OPERATIONAL MODE

An OPERATIONAL MODE, i.e., MODE, shall be any one inclusive combination of mode switch position and average reactor coolant temperature as specified in Table 1-2.

PHYSICS TESTS

PHYSICS TESTS shall be those tests performed to measure the fundamental nuclear characteristics of the reactor core and related instrumentation and 1) described in Chapter 14 of the UFSAR, 2) authorized under the provisions of 10 CFR 50.59, or 3) otherwise approved by the Commission.

PRESSURE BOUNDARY LEAKAGE

PRESSURE BOUNDARY LEAKAGE shall be leakage through a non-isolable fault in a reactor coolant system component body, pipe wall or vessel wall.

TABLE 1-1

SURVEILLANCE FREQUENCY NOTATION

	NOTATION	FREQUENCY
1. Shift	S	At least once per 12 hours
2. Day	D	At least once per 24 hours
3. Week	W	At least once per 7 days
4. Month	M	At least once per 31 days
5. Quarter	Q	At least once per 92 days
6. Semiannual	SA	At least once per 184 days
7. Annual	A	At least once per 366 days
8. Sesquiannual	E	At least once per 18 months (550 days)
9. Startup	S/U	Prior to each reactor startup
10. Not Applicable	NA	Not applicable

2.0 SAFETY LIMITS AND LIMITING SAFETY SYSTEM SETTINGS

2.1 SAFETY LIMITS

THERMAL POWER, Low Pressure or Low Flow

2.1.A THERMAL POWER shall not exceed 25% of RATED THERMAL POWER with the reactor vessel steam dome pressure less than 785 psig or core flow less than 10% of rated flow.

APPLICABILITY: OPERATIONAL MODE(s) 1 and 2.

ACTION:

With THERMAL POWER exceeding 25% of RATED THERMAL POWER and the reactor vessel steam dome pressure less than 785 psig or core flow less than 10% of rated flow, be in at least HOT SHUTDOWN within 2 hours and comply with the requirements of Specification 6.7.

THERMAL POWER, High Pressure and High Flow

2.1.B The MINIMUM CRITICAL POWER RATIO (MCPR) shall not be less than 1.08 with the reactor vessel steam dome pressure greater than or equal to 785 psig and core flow greater than or equal to 10% of rated flow. During single recirculation loop operation, this MCPR limit shall be increased by 0.01.

APPLICABILITY: OPERATIONAL MODE(s) 1 and 2.

ACTION:

With MCPR less than the above applicable limit and the reactor vessel steam dome pressure greater than or equal to 785 psig and core flow greater than or equal to 10% of rated flow, be in at least HOT SHUTDOWN within 2 hours and comply with the requirements of Specification 6.7.

2.0 SAFETY LIMITS AND LIMITING SAFETY SYSTEM SETTINGS

Reactor Coolant System Pressure

2.1.C The reactor coolant system pressure, as measured in the reactor vessel steam dome, shall not exceed 1345 psig.

APPLICABILITY: OPERATIONAL MODE(s) 1, 2, 3 and 4.

ACTION:

With the reactor coolant system pressure, as measured in the reactor vessel steam dome, above 1345 psig, be in at least HOT SHUTDOWN with reactor coolant system pressure less than or equal to 1345 psig within 2 hours and comply with the requirements of Specification 6.7.

Reactor Vessel Water Level

2.1.D The reactor vessel water level shall be greater than or equal to twelve inches above the top of the active irradiated fuel.

APPLICABILITY: OPERATIONAL MODE(s) 3, 4 and 5.

ACTION:

With the reactor vessel water level at or below twelve inches above the top of the active irradiated fuel, manually initiate the ECCS to restore the water level, after depressurizing the reactor vessel, if required, and comply with the requirements of Specification 6.7.

3.0 - LIMITING CONDITIONS FOR OPERATION

A. Compliance with the Limiting Conditions for Operation contained in the succeeding Specifications is required during the OPERATIONAL MODE(s) or other conditions specified therein; except that upon failure to meet the Limiting Conditions for Operation, the associated ACTION requirements shall be met.

B. Noncompliance with a Specification shall exist when the requirements of the Limiting Condition for Operation and associated ACTION requirements are not met within the specified time intervals. If the Limiting Condition for Operation is restored prior to expiration of the specified time intervals, completion of the ACTION requirements is not required.

C. When a Limiting Condition for Operation is not met, except as provided in the associated ACTION requirements, within one hour ACTION shall be initiated to place the unit in an OPERATIONAL MODE in which the Specification does not apply by placing it, as applicable, in:

1. At least HOT SHUTDOWN within the next 12 hours, and
2. At least COLD SHUTDOWN within the subsequent 24 hours.

Where corrective measures are completed that permit operation under the ACTION requirements, the ACTION may be taken in accordance with the specified time limits as measured from the time of failure to meet the Limiting Condition for Operation. Exceptions to these requirements are stated in the individual Specifications.

This Specification is not applicable in OPERATIONAL MODE 4 or 5.

D. When an LCO is not met, entry into a MODE or other specified condition in the Applicability shall not be made except when the associated ACTIONS to be entered permit continued operation in the MODE or other specified condition in the Applicability for an unlimited period of time. This Specification shall not prevent changes in MODES or other specified conditions in the Applicability that are required to comply with ACTIONS. Exceptions to these requirements are stated in the individual Specifications.

4.0 - SURVEILLANCE REQUIREMENTS

A. Surveillance Requirements shall be met during the reactor OPERATIONAL MODE(s) or other conditions specified for individual Limiting Conditions for Operation unless otherwise stated in an individual Surveillance Requirement.

B. Each Surveillance Requirement shall be performed within the specified surveillance interval with a maximum allowable extension not to exceed 25 percent of the surveillance interval.

C. Failure to perform a Surveillance Requirement within the allowed surveillance interval, defined by Specification 4.0.B, shall constitute noncompliance with the OPERABILITY requirements for a Limiting Condition for Operation. The time limits of the ACTION requirements are applicable at the time it is identified that a Surveillance Requirement has not been performed. The ACTION requirements may be delayed for up to 24 hours to permit the completion of the surveillance when the allowable outage time limits of the ACTION requirements are less than 24 hours. Surveillance requirements do not have to be performed on inoperable equipment.

D. Entry into an OPERATIONAL MODE or other specified applicable condition shall not be made unless the Surveillance Requirement(s) associated with the Limiting Condition for Operation have been performed within the applicable surveillance interval or as otherwise specified. This provision shall not prevent passage through or to OPERATIONAL MODE(s) as required to comply with ACTION requirements.

E. Surveillance Requirements for inservice inspection and testing of ASME Code Class 1, 2, and 3 components shall be applicable as follows:

1. Inservice Inspection of ASME Code Class 1, 2, and 3 components and inservice testing of ASME Code Class 1, 2, and 3 pumps and valves shall be performed in accordance with Section XI of the ASME Boiler and Pressure Vessel Code and applicable Addenda as required by 10 CFR Part 50, Section 50.55a(g) and 50.55a(f), respectively, except where specific written relief has been granted by the Commission pursuant to 10 CFR Part 50, Section 50.55a(g)(6)(i) or 50.55a(f)(6)(i), respectively.

4.0 - SURVEILLANCE REQUIREMENTS

2. Surveillance intervals specified in Section XI of the ASME Boiler and Pressure Vessel Code and applicable Addenda for the inservice inspection and testing activities required by the ASME Boiler and Pressure Vessel Code and applicable Addenda shall be applicable as follows in these Technical Specifications:

ASME Boiler and Pressure Vessel Code and applicable Addenda terminology for inservice inspection and testing activities	Required Frequencies for performing inservice inspection and testing activities
Weekly	At least once per 7 days
Monthly	At least once per 31 days
Quarterly or every 3 months	At least once per 92 days
Semiannually or every 6 months	At least once per 184 days
Every 9 months	At least once per 276 days
Yearly or annually	At least once per 366 days
Biennially or every 2 years	At least once per 731 days

3. The provisions of Specification 4.0.B are applicable to the above required frequencies for performing inservice inspection and testing activities.

4. Performance of the above inservice inspection and testing activities shall be in addition to other specified Surveillance Requirements.

5. Nothing in the ASME Boiler and Pressure Vessel Code shall be construed to supersede the requirements of any Technical Specification.

6. The Inservice Inspection Program for piping identified in NRC Generic Letter 88-01 shall be performed in accordance with the staff positions on schedule, methods, and personnel and sample expansion included in Generic Letter 88-01 or in accordance with alternate measures approved by the NRC staff.

3.1 - LIMITING CONDITIONS FOR OPERATION

A. Reactor Protection System (RPS)

The reactor protection system (RPS) instrumentation CHANNEL(s) shown in Table 3.1.A-1 shall be OPERABLE.

APPLICABILITY:

As shown in Table 3.1.A-1.

ACTION:

1. With the number of OPERABLE CHANNEL(s) less than required by the Minimum CHANNEL(s) per TRIP SYSTEM requirement for one TRIP SYSTEM, place the inoperable CHANNEL(s) and/or that TRIP SYSTEM in the tripped condition[a] within 1 hour.
2. With the number of OPERABLE CHANNEL(s) less than required by the Minimum CHANNEL(s) per TRIP SYSTEM requirement for both TRIP SYSTEM(s), place at least one TRIP SYSTEM in the tripped condition[b] within 1 hour and take the ACTION required by Table 3.1.A-1.

4.1 - SURVEILLANCE REQUIREMENTS

A. Reactor Protection System

1. Each reactor protection system instrumentation CHANNEL shall be demonstrated OPERABLE by the performance of the CHANNEL CHECK, CHANNEL FUNCTIONAL TEST and CHANNEL CALIBRATION operations for the OPERATIONAL MODE(s) and at the frequencies shown in Table 4.1.A-1.
2. LOGIC SYSTEM FUNCTIONAL TEST(s) of all CHANNEL(s) shall be performed at least once per 18 months.
3. The response time of each reactor trip functional unit shown in Table 3.1.A-1 shall be demonstrated at least once per 18 months. Each test shall include at least one CHANNEL per TRIP SYSTEM such that all CHANNEL(s) are tested at least once every N times 18 months where N is the total number of redundant CHANNEL(s) in a specific reactor TRIP SYSTEM.

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a An inoperable CHANNEL need not be placed in the tripped condition when this would cause the trip function to occur. In these cases, the inoperable CHANNEL shall be restored to OPERABLE status within 2 hours or the ACTION required by Table 3.1.A-1 for that trip function shall be taken.

b The TRIP SYSTEM need not be placed in the tripped condition if this would cause the trip function to occur. When a TRIP SYSTEM can be placed in the tripped condition without causing the trip function to occur, place the TRIP SYSTEM with the most inoperable CHANNEL(s) in the tripped condition; if both systems have the same number of inoperable CHANNEL(s), place either TRIP SYSTEM in the tripped condition.

TABLE 4.1.A-1

REACTOR PROTECTION SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

Functional Unit	Applicable OPERATIONAL MODES	CHANNEL CHECK	CHANNEL FUNCTIONAL TEST	CHANNEL[a] CALIBRATION
1. Intermediate Range Monitor:				
a. Neutron Flux - High	2	S[b]	S/U[c], W[o]	E[o,r]
	3, 4, 5	S	W[o]	E[o,r]
b. Inoperative	2, 3, 4, 5	NA	W[o]	NA
2. Average Power Range Monitor[f]:				
a. Setdown Neutron Flux - High	2	S[b]	S/U[c], W[o]	SA[o]
	3, 5[m]	S	W	SA
b. Flow Biased Neutron Flux - High	1	S, D[g]	W	W[d,e], SA
c. Fixed Neutron Flux - High	1	S	W	W[d], SA
d. Inoperative	1, 2, 3, 5[m]	NA	W	NA
3. Reactor Vessel Steam Dome Pressure - High	1, 2[i]	NA	M	Q
4. Reactor Vessel Water Level - Low	1, 2	D	M	E[h]
5. Main Steam Line Isolation Valve - Closure	1, 2[p]	NA	M	E
6. Main Steam Line Radiation - High	1, 2[l]	S	M	E[q]
7. Drywell Pressure - High	1, 2[n]	NA	M	Q

TABLE 4.1.A-1 (Continued)

REACTOR PROTECTION SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

TABLE NOTATION

(a) Neutron detectors may be excluded from the CHANNEL CALIBRATION.

(b) The IRM and SRM channels shall be determined to overlap for at least (½) decades during each startup after entering OPERATIONAL MODE 2 and the IRM and APRM channels shall be determined to overlap for at least (½) decades during each controlled shutdown, if not performed within the previous 7 days.

(c) Within 24 hours prior to startup, if not performed within the previous 7 days. The weekly CHANNEL FUNCTIONAL TEST may be used to fulfill this requirement.

(d) This calibration shall consist of the adjustment of the APRM CHANNEL to conform, within 2% of RATED THERMAL POWER, to the power values calculated by a heat balance during OPERATIONAL MODE 1 when THERMAL POWER is ≥25% of RATED THERMAL POWER. This adjustment must be accomplished: a) within 2 hours if the APRM CHANNEL is indicating lower power values than the heat balance, or b) within 12 hours if the APRM CHANNEL is indicating higher power values than the heat balance. Until any required APRM adjustment has been accomplished, notification shall be posted on the reactor control panel.

Any APRM CHANNEL gain adjustment made in compliance with Specification 3.11.B shall not be included in determining the above difference. This calibration is not required when THERMAL POWER is <25% of RATED THERMAL POWER. The provisions of Specification 4.0.D are not applicable.

(e) This calibration shall consist of the adjustment of the APRM flow biased channel to conform to a calibrated flow signal.

(f) The LPRMs shall be calibrated at least once per 2000 effective full power hours (EFPH).

(g) Verify measured recirculation loop flow to be greater than or equal to established recirculation loop flow at the existing pump speed.

(h) Trip units are calibrated at least once per 31 days and transmitters are calibrated at the frequency identified in the table.

(i) This function is not required to be OPERABLE when the reactor pressure vessel head is unbolted or removed per Specification 3.12.A.

(j) With any control rod withdrawn. Not applicable to control rods removed per Specification 3.10.I or 3.10.J.

(k) This function may be bypassed, provided a control rod block is actuated, for reactor protection system reset in Refuel and Shutdown positions of the reactor mode switch.

TABLE 4.1.A-1 (Continued)

REACTOR PROTECTION SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

(l) This function not required to be OPERABLE when THERMAL POWER is less than 45% of RATED THERMAL POWER.

(m) Required to be OPERABLE only prior to and during required SHUTDOWN MARGIN demonstrations performed per Specification 3.12.B.

(n) This function is not required to be OPERABLE when PRIMARY CONTAINMENT INTEGRITY is not required.

(o) The provisions of Specification 4.0.D are not applicable to the CHANNEL FUNCTIONAL TEST and CHANNEL CALIBRATION surveillances for a period of 24 hours after entering OPERATIONAL MODE 2 or 3 when shutting down from OPERATIONAL MODE 1.

(p) This function is not required to be OPERABLE when reactor pressure is less than 600 psig.

(q) A current source provides an instrument channel alignment every 3 months.

(r) The CHANNEL CALIBRATION surveillance requirements shall be performed if not performed within the previous seven days.

3.2 - LIMITING CONDITIONS FOR OPERATION

A. Isolation Actuation

The isolation actuation instrumentation CHANNEL(s) shown in Table 3.2.A-1 shall be OPERABLE with their trip setpoints set consistent with the values shown in the Trip Setpoint column.

APPLICABILITY:

As shown in Table 3.2.A-1.

ACTION:

1. With an isolation actuation instrumentation CHANNEL trip setpoint less conservative than the value shown in the Trip Setpoint column of Table 3.2.A-1, declare the CHANNEL inoperable until the CHANNEL is restored to OPERABLE status with its trip setpoint adjusted consistent with the Trip Setpoint value.

2. With the number of OPERABLE CHANNEL(s) less than required by the Minimum CHANNEL(s) per TRIP SYSTEM requirement for one TRIP SYSTEM, place the inoperable CHANNEL(s) and/or TRIP SYSTEM in the tripped condition[a] within one hour.

4.2 - SURVEILLANCE REQUIREMENTS

A. Isolation Actuation

1. Each isolation actuation instrumentation CHANNEL shall be demonstrated OPERABLE by the performance of the CHANNEL CHECK, CHANNEL FUNCTIONAL TEST and CHANNEL CALIBRATION operations for the OPERATIONAL MODE(s) and at the frequencies shown in Table 4.2.A-1.

2. LOGIC SYSTEM FUNCTIONAL TEST(s) of all CHANNEL(s) shall be performed at least once per 18 months.

---

a An inoperable CHANNEL need not be placed in the tripped condition where this would cause the trip function to occur. In these cases, the inoperable CHANNEL shall be restored to OPERABLE status within 2 hours or the ACTION required by Table 3.2.A-1 for that trip function shall be taken.

TABLE 3.2.A-1

ISOLATION ACTUATION INSTRUMENTATION

Functional Unit	Trip Setpoint[i]	Minimum CHANNEL(s) per TRIP SYSTEM[a]	Applicable OPERATIONAL MODE(s)	ACTION
1. PRIMARY CONTAINMENT ISOLATION				
a. Reactor Vessel Water Level - Low	≥144 inches	2	1, 2, 3	20
b. Drywell Pressure - High[d]	≤2 psig	2	1, 2, 3	20
c. Drywell Radiation - High	≤100 R/hr	1	1, 2, 3	23
2. SECONDARY CONTAINMENT ISOLATION				
a. Reactor Vessel Water Level - Low[c]	≥144 inches	2	1, 2, 3 & *	24
b. Drywell Pressure - High[c,d]	≤2 psig	2	1, 2, 3	24
c. Reactor Building Ventilation Exhaust Radiation - High[c]	≤4 mR/hr	2	1, 2, 3 & * *	24
d. Refueling Floor Radiation - High[c]	≤100 mR/hr	2	1, 2, 3 & * *	24
3. MAIN STEAM LINE (MSL) ISOLATION				
a. Reactor Vessel Water Level - Low Low	≥84 inches	2	1, 2, 3	21
b. MSL Tunnel Radiation - High[b]	≤3[g] x normal background	2	1, 2, 3	21
c. MSL Pressure - Low	≥825 psig	2	1	22
d. MSL Flow - High	≤120% of rated	2/line	1, 2, 3	21
e. MSL Tunnel Temperature - High	≤200°F	2 of 4 in each of 2 sets	1, 2, 3	21

TABLE 3.2.A-1 (Continued)

ISOLATION ACTUATION INSTRUMENTATION

Functional Unit	Trip Setpoint[l]	Minimum CHANNEL(s) per TRIP SYSTEM[a]	Applicable OPERATIONAL MODE(s)	ACTION
4. REACTOR WATER CLEANUP SYSTEM ISOLATION				
a. Standby Liquid Control System Initiation[f]	NA	NA	1, 2, 3	23
b. Reactor Vessel Water Level - Low	≥144 inches	2	1, 2, 3	23
5. ISOLATION CONDENSER ISOLATION				
a. Steam Flow - High	≤300% of rated steam flow	1	1, 2, 3	23
b. Return Flow - High	≤32 (Unit 2)/ ≤14.8 (Unit 3) inches water diff.	1	1, 2, 3	23
6. HIGH PRESSURE COOLANT INJECTION ISOLATION				
a. Steam Flow - High	≤300% of rated steam flow [h]	1	1, 2, 3	23
b. Reactor Vessel Pressure - Low	≥80 psig	2	1, 2, 3	23
c. Area Temperature - High	≤200°F	4[j]	1, 2, 3	23

TABLE 3.2.A-1 (Continued)

ISOLATION ACTUATION INSTRUMENTATION

TABLE NOTATION

* During CORE ALTERATIONS or operations with a potential for draining the reactor vessel.

** When handling irradiated fuel in the secondary containment.

(a) A CHANNEL may be placed in an inoperable status for up to 2 hours for required surveillance without placing the CHANNEL in the tripped condition provided the Functional Unit maintains isolation actuation capability.

(b) Also trips the mechanical vacuum pump and isolates the steam jet air ejectors.

(c) Isolates the reactor building ventilation system and actuates the standby gas treatment system.

(d) This function is not required to be OPERABLE when PRIMARY CONTAINMENT INTEGRITY is not required.

(e) Only one TRIP SYSTEM.

(f) Closes only reactor water cleanup system isolation valves.

(g) Normal background is as measured during full power operation without hydrogen being injected. With Unit 2 operating above 20% RATED THERMAL POWER and hydrogen being injected into the feedwater, this Unit 2 setting may be as measured during full power operation with hydrogen being injected.

(h) Includes a time delay of $3 \leq t \leq 9$ seconds.

(i) Reactor vessel water level settings are expressed in inches above the top of active fuel (which is 360 inches above vessel zero).

(j) All four switches in either of 2 groups for each trip system.

TABLE 4.2.A-1

ISOLATION ACTUATION INSTRUMENTATION SURVEILLANCE REQUIREMENTS

Functional Unit	CHANNEL CHECK	CHANNEL FUNCTIONAL TEST	CHANNEL CALIBRATION	Applicable OPERATIONAL MODE(s)
1. PRIMARY CONTAINMENT ISOLATION				
a. Reactor Vessel Water Level - Low	S	M	E[a]	1, 2, 3
b. Drywell Pressure - High[b]	NA	M	Q	1, 2, 3
c. Drywell Radiation - High	S	M	E	1, 2, 3
2. SECONDARY CONTAINMENT ISOLATION				
a. Reactor Vessel Water Level - Low[c]	S	M	E[a]	1, 2, 3 & *
b. Drywell Pressure - High[b,c]	NA	M	Q	1, 2, 3
c. Reactor Building Ventilation Exhaust Radiation - High[c]	S	M	E	1, 2, 3 & * *
d. Refueling Floor Radiation - High[c]	S	M	Q	1, 2, 3 & * *
3. MAIN STEAM LINE (MSL) ISOLATION				
a. Reactor Vessel Water Level - Low Low	S	M	E[a]	1, 2, 3
b. MSL Tunnel Radiation - High	S	M	E[d]	1, 2, 3
c. MSL Pressure - Low	NA	M	Q	1
d. MSL Flow - High	S	M	E	1, 2, 3
e. MSL Tunnel Temperature - High	NA	E	E	1, 2, 3

TABLE 4.2.A-1 (Continued)

ISOLATION ACTUATION INSTRUMENTATION SURVEILLANCE REQUIREMENTS

Functional Unit	CHANNEL CHECK	CHANNEL FUNCTIONAL TEST	CHANNEL CALIBRATION	Applicable OPERATIONAL MODE(s)
4. REACTOR WATER CLEANUP SYSTEM ISOLATION				
a. Standby Liquid Control System Initiation	NA	E	NA	1, 2, 3
b. Reactor Vessel Water Level - Low	S	M	E[a]	1, 2, 3
5. ISOLATION CONDENSER				
a. Steam Flow - High	NA	M	Q	1, 2, 3
b. Return Flow - High	NA	M	Q	1, 2, 3
6. HIGH PRESSURE COOLANT INJECTION ISOLATION				
a. Steam Flow - High	NA	M	E[a]	1, 2, 3
b. Reactor Vessel Pressure - Low	NA	M	E[a]	1, 2, 3
c. Area Temperature - High	NA	E	E	1, 2, 3
7. SHUTDOWN COOLING ISOLATION				
a. Reactor Vessel Water Level - Low	S	M	E[a]	3, 4, 5
b. Recirculation Line Water Temperature - High (Cut-in Permissive)	NA	M	Q	1, 2, 3

TABLE 4.2.A-1 (Continued)

ISOLATION ACTUATION INSTRUMENTATION
SURVEILLANCE REQUIREMENTS

TABLE NOTATION

* During CORE ALTERATIONS or operations with a potential for draining the reactor vessel.

** When handling irradiated fuel in the secondary containment.

(a) Trip units are calibrated at least once per 31 days and transmitters are calibrated at the frequency identified in the table.

(b) This function is not required to be OPERABLE when PRIMARY CONTAINMENT INTEGRITY is not required.

(c) Isolates the reactor building ventilation system and actuates the standby gas treatment system.

(d) These instrument channels will be calibrated using simulated electrical signals once every three months. In addition, calibration including the sensors will be performed every 18 months.

3.2 - LIMITING CONDITIONS FOR OPERATION

B. Emergency Core Cooling Systems (ECCS) Actuation

The ECCS actuation instrumentation CHANNEL(s) shown in Table 3.2.B-1 shall be OPERABLE with their trip setpoints set consistent with the values shown in the Trip Setpoint column.

APPLICABILITY:

As shown in Table 3.2.B-1.

ACTION:

1. With an ECCS actuation instrumentation CHANNEL trip setpoint less conservative than the value shown in the Trip Setpoint column of Table 3.2.B-1, declare the CHANNEL inoperable until the CHANNEL is restored to OPERABLE status with its trip setpoint adjusted consistent with the Trip Setpoint value.
2. With one or more ECCS actuation instrumentation CHANNEL(s) inoperable, take the ACTION required by Table 3.2.B-1.
3. With either ADS TRIP SYSTEM inoperable, restore the inoperable TRIP SYSTEM to OPERABLE status within:

   a. 7 days provided that both the HPCI and IC are OPERABLE, or

   b. 72 hours.

   With the above provisions of this ACTION not met, be in at least HOT

4.2 - SURVEILLANCE REQUIREMENTS

B. ECCS Actuation

1. Each ECCS actuation instrumentation CHANNEL shall be demonstrated OPERABLE by the performance of the CHANNEL CHECK, CHANNEL FUNCTIONAL TEST and CHANNEL CALIBRATION operations for the OPERATIONAL MODE(s) and at the frequencies shown in Table 4.2.B-1.
2. LOGIC SYSTEM FUNCTIONAL TEST(s) of all CHANNEL(s) shall be performed at least once per 18 months.

TABLE 3.2.B-1 (Continued)

ECCS ACTUATION INSTRUMENTATION

Functional Unit	Trip Setpoint(h)	Minimum CHANNEL(s) per Trip Function(a)	Applicable OPERATIONAL MODE(s)	ACTION
3. HIGH PRESSURE COOLANT INJECTION (HPCI) SYSTEM(d)				
a. Reactor Vessel Water Level - Low Low	≥84 inches	4	1, 2, 3	35
b. Drywell Pressure - High(f)	≤2 psig	4	1, 2, 3	35
c. Condensate Storage Tank Level - Low(i)	≥10,000 gal	2	1, 2, 3	35
d. Suppression Chamber Water Level - High(i)	≤15' 5" above bottom of chamber	2	1, 2, 3	35
e. Reactor Vessel Water Level - High (Trip)	≤194 inches	1	1, 2, 3	31
f. HPCI Pump Discharge Flow - Low (Bypass)	≥600 gpm	1	1, 2, 3	33
g. Manual Initiation	NA	1/system	1, 2, 3	34
4. AUTOMATIC DEPRESSURIZATION SYSTEM - TRIP SYSTEM 'A'(d)				
a. Reactor Vessel Water Level - Low Low	≥84 inches	2	1, 2, 3	30
b. Drywell Pressure - High(f)	≤2 psig	2	1, 2, 3	30
c. Initiation Timer	≤120 sec	1	1, 2, 3	31
d. Low Low Level Timer	≤10 min	1	1, 2, 3	31
e. CS Pump Discharge Pressure - High (Permissive)	≥100 psig & ≤150 psig	1/pump	1, 2, 3	31
f. LPCI Pump Discharge Pressure - High (Permissive)	≥100 psig & ≤150 psig	1/pump	1, 2, 3	31

TABLE 3.2.B-1 (Continued)

ECCS ACTUATION INSTRUMENTATION

Functional Unit	Trip Setpoint[h]	Minimum CHANNEL(s) per Trip Function[a]	Applicable OPERATIONAL MODE(s)	ACTION
5. AUTOMATIC DEPRESSURIZATION SYSTEM - TRIP SYSTEM 'B'[d]				
a. Reactor Vessel Water Level - Low Low	≥84 inches	2	1, 2, 3	30
b. Drywell Pressure - High[f]	≤2 psig	2	1, 2, 3	30
c. Initiation Timer	≤120 sec	1	1, 2, 3	31
d. Low Low Level Timer	≤10 min	1	1, 2, 3	31
e. CS Pump Discharge Pressure - High (Permissive)	≥100 psig & ≤150 psig	1/pump	1, 2, 3	31
f. LPCI Pump Discharge Pressure - High (Permissive)	≥100 psig & ≤150 psig	1/pump	1, 2, 3	31
6. LOSS OF POWER				
a. 4.16 kv Emergency Bus Undervoltage (Loss of Voltage)	2930 ± 146 volts decreasing voltage	2/bus	1, 2, 3, 4[e], 5[e]	36
b. 4.16 kv Emergency Bus Undervoltage (Degraded Voltage)	≥ 3784 volts (Unit 2)[g][j] ≥ 3832 volts (Unit 3)[g][j]	2/bus	1, 2, 3, 4[e], 5[e]	36

TABLE 4.2.B-1

ECCS ACTUATION INSTRUMENTATION SURVEILLANCE REQUIREMENTS

Functional Unit	CHANNEL CHECK	CHANNEL FUNCTIONAL TEST	CHANNEL CALIBRATION	Applicable OPERATIONAL MODE(s)
1. CORE SPRAY (CS) SYSTEM				
a. Reactor Vessel Water Level - Low Low	S	M	Q	1, 2, 3, 4[b], 5[b]
b. Drywell Pressure - High[d]	NA	M	Q	1, 2, 3
c. Reactor Vessel Pressure - Low (Permissive)	NA	M	Q	1, 2, 3, 4[b], 5[b]
d. CS Pump Discharge Flow - Low (Bypass)	NA	M	Q[e]	1, 2, 3, 4[b], 5[b]
2. LOW PRESSURE COOLANT INJECTION (LPCI) SUBSYSTEM				
a. Reactor Vessel Water Level - Low Low	S	M	Q	1, 2, 3, 4[b], 5[b]
b. Drywell Pressure - High[d]	NA	M	Q	1, 2, 3
c. Reactor Vessel Pressure - Low (Permissive)	NA	M	Q	1, 2, 3, 4[b], 5[b]
d. LPCI Pump Discharge Flow - Low (Bypass)	NA	M	Q[e]	1, 2, 3, 4[b], 5[b]
3. HIGH PRESSURE COOLANT INJECTION (HPCI) SYSTEM[a]				
a. Reactor Vessel Water Level - Low Low	S	M	Q	1, 2, 3
b. Drywell Pressure - High[d]	NA	M	Q	1, 2, 3
c. Condensate Storage Tank Level - Low	NA	M	NA	1, 2, 3
d. Suppression Chamber Water Level - High	NA	M	NA	1, 2, 3
e. Reactor Vessel Water Level - High (Trip)	NA	M	E	1, 2, 3
f. HPCI Pump Discharge Flow - Low (Bypass)	NA	M	Q	1, 2, 3
g. Manual Initiation	NA	E	NA	1, 2, 3

3.2 - LIMITING CONDITIONS FOR OPERATION

C. ATWS - RPT

The anticipated transient without scram recirculation pump trip (ATWS - RPT) instrumentation CHANNEL(s) shown in Table 3.2.C-1 shall be OPERABLE with their trip setpoints set consistent with the values shown in the Trip Setpoint column.

APPLICABILITY:

OPERATIONAL MODE 1.

ACTION:

1. With an ATWS - RPT instrumentation CHANNEL trip setpoint less conservative than the value shown in the Trip Setpoint column of Table 3.2.C-1, declare the CHANNEL inoperable until the CHANNEL is restored to OPERABLE status with its trip setpoint adjusted consistent with the Trip Setpoint value.
2. With one level CHANNEL or one pressure CHANNEL inoperable in one or both TRIP SYSTEM(s), within 14 days, either restore the inoperable CHANNEL to OPERABLE status or place the inoperable CHANNEL in the tripped condition[a]. Otherwise, be in STARTUP within the next 6 hours.
3. With two level CHANNELS or two pressure CHANNELS inoperable in one or both TRIP SYSTEM(s), declare the TRIP SYSTEM(s) inoperable.

4.2 - SURVEILLANCE REQUIREMENTS

C. ATWS - RPT

1. Each ATWS - RPT instrumentation CHANNEL shall be demonstrated OPERABLE by the performance of the CHANNEL CHECK, CHANNEL FUNCTIONAL TEST and CHANNEL CALIBRATION operations at the frequencies shown in Table 4.2.C-1.
2. LOGIC SYSTEM FUNCTIONAL TEST(s) of all CHANNEL(s) shall be performed at least once per 18 months.

a The inoperable CHANNEL(s) need not be placed in the tripped condition where this would cause the Trip Function to occur.

3.2 - LIMITING CONDITIONS FOR OPERATION

4. With one level CHANNEL and one pressure CHANNEL inoperable in one or both TRIP SYSTEM(s), restore at least one inoperable CHANNEL to OPERABLE status within 14 days or be in STARTUP within the next 6 hours.

5. With one TRIP SYSTEM inoperable, restore the inoperable TRIP SYSTEM to OPERABLE status within 72 hours or be in at least STARTUP within the next 6 hours.

6. With both TRIP SYSTEM(s) inoperable, restore at least one TRIP SYSTEM to OPERABLE status within one hour or be in at least STARTUP within the next 6 hours.

4.2 - SURVEILLANCE REQUIREMENTS

3.2 - LIMITING CONDITIONS FOR OPERATION

D: Isolation Condenser Actuation

The isolation condenser actuation instrumentation CHANNEL(s) shown in Table 3.2.D-1 shall be OPERABLE with their trip setpoints set consistent with the values shown in the Trip Setpoint column.

APPLICABILITY:

OPERATIONAL MODE(s) 1, 2 and 3 with the reactor steam dome pressure >150 psig.

ACTION:

1. With an isolation condenser actuation instrumentation CHANNEL trip setpoint less conservative than the value shown in the Trip Setpoint column of Table 3.2.D-1, declare the CHANNEL inoperable until the CHANNEL is restored to OPERABLE status with its trip setpoint adjusted consistent with the Trip Setpoint value.
2. With one or more isolation condenser system actuation instrumentation CHANNEL(s) inoperable, take the ACTION required by Table 3.2.D-1.

4.2 - SURVEILLANCE REQUIREMENTS

D. Isolation Condenser Actuation

1. Each isolation condenser actuation instrumentation CHANNEL shall be demonstrated OPERABLE by the performance of the CHANNEL CHECK, CHANNEL FUNCTIONAL TEST and CHANNEL CALIBRATION operations at the frequencies shown in Table 4.2.D-1.
2. LOGIC SYSTEM FUNCTIONAL TEST(s) of all CHANNEL(s) shall be performed at least once per 18 months.

TABLE 3.2.D-1

ISOLATION CONDENSER ACTUATION INSTRUMENTATION

Functional Unit	Trip Setpoint	Minimum CHANNEL(s) per TRIP SYSTEM[a]	ACTION
Reactor Vessel Pressure - High	≤1070 psig	2	40

ACTION

ACTION 40 - With the number of OPERABLE CHANNEL(s) less than required by the Minimum CHANNEL(s) per TRIP SYSTEM requirement:

a. With one CHANNEL inoperable, place the inoperable CHANNEL in the tripped condition within one hour or declare the isolation condenser system inoperable.

b. With more than one CHANNEL inoperable, declare the isolation condenser system inoperable.

---

a A CHANNEL may be placed in an inoperable status for up to 2 hours for required surveillance without placing the TRIP SYSTEM in the tripped condition provided at least one OPERABLE CHANNEL in the same TRIP SYSTEM is monitoring that parameter.

TABLE 4.2.D-1

ISOLATION CONDENSER ACTUATION INSTRUMENTATION
SURVEILLANCE REQUIREMENTS

Functional Unit	CHANNEL CHECK	CHANNEL FUNCTIONAL TEST	CHANNEL CALIBRATION
Reactor Vessel Pressure - High	NA	M	Q

TABLE 3.2.E-1

CONTROL ROD BLOCK INSTRUMENTATION

Functional Unit	Trip Setpoint	Minimum CHANNEL(s) per Trip Function[i]	Applicable OPERATIONAL MODE(s)	ACTION
1. ROD BLOCK MONITORS[a]				
a. Upscale	As specified in the COLR	2	1[e]	50
b. Inoperative	NA	2	1[e]	50
c. Downscale	≥5/125 of full scale	2	1[e]	50
2. AVERAGE POWER RANGE MONITORS				
a. Flow Biased Neutron Flux - High				
1. Dual Recirculation Loop Operation	≤(0.58W+50)[g]	4	1	51
2. Single Recirculation Loop Operation	≤(0.58W+46.5)[g]	4	1	51
b. Inoperative	NA	4	1, 2, 5[h]	51
c. Downscale	≥3/125 of full scale	4	1	51
d. Startup Neutron Flux - High	≤12/125 of full scale	4	2, 5[h]	51

TABLE 3.2.E-1 (Continued)

CONTROL ROD BLOCK INSTRUMENTATION

Functional Unit	Trip Setpoint	Minimum CHANNEL(s) per Trip Function[i]	Applicable OPERATIONAL MODE(s)	ACTION
3. SOURCE RANGE MONITORS				
a. Detector not full in[b]	NA	3	2	51
		2	5	51
b. Upscale[c]	≤1 x 10^5 cps	3	2	51
		2	5	51
c. Inoperative[c]	NA	3	2	51
		2	5	51
4. INTERMEDIATE RANGE MONITORS				
a. Detector not full in[d]	NA	6	2, 5	51
b. Upscale	≤108/125 of full scale	6	2, 5	51
c. Inoperative	NA	6	2, 5	51
d. Downscale[d]	≥5/125 of full scale	6	2, 5	51

TABLE 3.2.E-1 (Continued)

CONTROL ROD BLOCK INSTRUMENTATION

Functional Unit	Trip Setpoint	Minimum CHANNEL(s) per Trip Function[i]	Applicable OPERATIONAL MODE(s)	ACTION
5. SCRAM DISCHARGE VOLUME (SDV)				
a. Water Level - High	(Unit 2) ≤29 gal (Unit 3) ≤25 gal	1 per bank	1, 2, 5[f]	52
b. SDV Switch in Bypass	NA	1	5[f]	52

TABLE 3.2.E-1 (Continued)

CONTROL ROD BLOCK INSTRUMENTATION

TABLE NOTATION

(a) The RBM shall be automatically bypassed when a peripheral control rod is selected or the reference APRM channel indicates less than 30% of RATED THERMAL POWER.

(b) This function shall be automatically bypassed if detector count rate is >100 cps or the IRM channels are on range 3 or higher.

(c) This function shall be automatically bypassed when the associated IRM channels are on range 8 or higher.

(d) This function shall be automatically bypassed when the IRM channels are on range 1.

(e) With THERMAL POWER $\geq$30% of RATED THERMAL POWER.

(f) With more than one control rod withdrawn. Not applicable to control rods removed per Specification 3.10.I or 3.10.J.

(g) The Average Power Range Monitor rod block function is varied as a function of recirculation drive flow (W). The trip setting of this function must be maintained in accordance with Specification 3.11.B. W is equal to the percentage of the drive flow required to produce a rated core flow of 98 $\times 10^6$ lbs/hr.

(h) Required to be OPERABLE only during SHUTDOWN MARGIN demonstrations performed per Specification 3.12.B.

(i) A CHANNEL may be placed in an inoperable status for up to 2 hours for required surveillance without placing the CHANNEL in the tripped condition provided the Functional Unit maintains control rod block capability.

TABLE 4.2.E-1

CONTROL ROD BLOCK INSTRUMENTATION SURVEILLANCE REQUIREMENTS

Functional Unit	CHANNEL CHECK	CHANNEL FUNCTIONAL TEST	CHANNEL CALIBRATION[a]	Applicable OPERATIONAL MODE(s)
1. ROD BLOCK MONITORS				
a. Upscale	NA	S/U[b,c], M[c]	Q	1[d]
b. Inoperative	NA	S/U[b,c], M[c]	NA	1[d]
c. Downscale	NA	S/U[b,c], M[c]	Q	1[d]
2. AVERAGE POWER RANGE MONITORS				
a. Flow Biased Neutron Flux - High				
1. Dual Recirculation Loop Operation	NA	S/U[b], M	SA	1
2. Single Recirculation Loop Operation	NA	S/U[b], M	SA	1
b. Inoperative	NA	S/U[b], M	NA	1, 2, 5[j]
c. Downscale	NA	S/U[b], M	Q	1
d. Startup Neutron Flux - High	NA	S/U[b], M	SA[k]	2, 5[j]
3. SOURCE RANGE MONITORS				
a. Detector not full in[f]	NA	S/U[b], W	E	2,[i] 5
b. Upscale[g]	NA	S/U[b], W	E	2,[i] 5
c. Inoperative[g]	NA	S/U[b], W	NA	2,[i] 5

TABLE 4.2.E-1 (Continued)

CONTROL ROD BLOCK INSTRUMENTATION
SURVEILLANCE REQUIREMENTS

Functional Unit	CHANNEL CHECK	CHANNEL FUNCTIONAL TEST	CHANNEL CALIBRATION[a]	Applicable OPERATIONAL MODE(s)
4. INTERMEDIATE RANGE MONITORS				
a. Detector not full in[h]	NA	S/U[b], W	E	2[i], 5
b. Upscale	NA	S/U[b], W	E[k]	2[i], 5
c. Inoperative	NA	S/U[b], W	NA	2[i], 5
d. Downscale[h]	NA	S/U[b], W	E[k]	2[i], 5
5. SCRAM DISCHARGE VOLUME (SDV)				
a. Water Level - High	NA	Q	NA	1, 2, 5[e]
b. SDV Switch in Bypass	NA	E	NA	5[e]

TABLE 4.2.E-1 (Continued)

CONTROL ROD BLOCK INSTRUMENTATION
SURVEILLANCE REQUIREMENTS

TABLE NOTATION

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

(b) Within 7 days prior to startup.

(c) Includes reactor manual control "relay select matrix" system input.

(d) With THERMAL POWER ≥30% of RATED THERMAL POWER.

(e) With more than one control rod withdrawn. Not applicable to control rods removed per Specification 3.10.I or 3.10.J.

(f) This function shall be automatically bypassed if detector count rate is >100 cps or the IRM channels are on range 3 or higher.

(g) This function shall be automatically bypassed when the associated IRM channels are on range 8 or higher.

(h) This function shall be automatically bypassed when the IRM channels are on range 1.

(i) The provisions of Specification 4.0.D are not applicable to the CHANNEL FUNCTIONAL TEST and CHANNEL CALIBRATION surveillances for entry into the applicable OPERATIONAL MODE(s) from OPERATIONAL MODE 1 provided the surveillances are performed within 12 hours after such entry

(j) Required to be OPERABLE only during SHUTDOWN MARGIN demonstrations performed per Specification 3.12.B.

(k) The CHANNEL CALIBRATION surveillance requirements shall be performed within 12 hours upon each entry into any OPERATIONAL MODE(s) from OPERATIONAL MODE 1 if not performed within the previous seven days.

TABLE 3.2.F-1

ACCIDENT MONITORING INSTRUMENTATION

INSTRUMENTATION	Required CHANNEL(s)	Minimum CHANNEL(s)	Applicable OPERATIONAL MODE(s)	ACTION
1. Reactor Vessel Pressure	2	1	1, 2	60
2. Reactor Vessel Water Level	2	1	1, 2	60
3 Torus Water Level	2	1	1, 2	60
4. Torus Water Temperature	2	1	1, 2	60
5. Drywell Pressure - Wide Range	2	1	1, 2	60
6. Drywell Pressure - Narrow Range	2	1	1, 2	60
7. Drywell Air Temperature	2	1	1, 2	60
8. Drywell Oxygen Concentration - Analyzer and Monitor	2	1	1, 2	62
9. Drywell Hydrogen Concentration - Analyzer and Monitor	2	1	1, 2	62
10 Safety & Relief Valve Position Indicators - Acoustic & Temperature	2/valve (1 each)	1/valve	1, 2	63
11. (Source Range) Neutron Monitors	2	2	1,2	60
12. Drywell Radiation Monitors	2	2	1, 2, 3	61
13. Torus Pressure	2[a]	1	1, 2	60

a This function is shared with Drywell Pressure-Wide Range and Drywell Pressure-Narrow Range.

TABLE 3.2.F-1 (Continued)

ACCIDENT MONITORING INSTRUMENTATION

ACTION

ACTION 60 -

a. With the number of OPERABLE accident monitoring instrumentation CHANNEL(s) less than the Required CHANNEL(s) shown in Table 3.2.F-1, restore the inoperable CHANNEL(s) to OPERABLE status within 30 days or be in at least HOT SHUTDOWN within the next 12 hours.

b. With the number of OPERABLE accident monitoring instrumentation CHANNEL(s) less than the Minimum CHANNEL(s) shown in Table 3.2.F-1, restore the inoperable CHANNEL(s) to OPERABLE status within 48 hours or be in at least HOT SHUTDOWN within the next 12 hours.

ACTION 61- With the number of OPERABLE accident monitoring instrumentation CHANNEL(s) less than the Minimum CHANNEL(s) shown in Table 3.2.F-1, initiate the preplanned alternate method of monitoring the appropriate parameter(s) within 72 hours, and:

a. Either restore the inoperable CHANNEL(s) to OPERABLE status within 7 days of the event, or

b. Prepare and submit a Special Report to the Commission pursuant to Specification 6.9.B within 30 days following the event outlining the action taken, the cause of the inoperability and the plans and schedule for restoring the system to OPERABLE status.

ACTION 62-

a. With the number of OPERABLE accident monitoring instrumentation CHANNEL(s) one less than the Required CHANNEL(s) shown in Table 3.2.F-1, restore the inoperable CHANNEL(s) to OPERABLE status within 30 days or be in at least HOT SHUTDOWN within the next 12 hours.

b. With the number of OPERABLE accident monitoring instrumentation CHANNEL(s) less than the Minimum CHANNEL(s) shown in Table 3.2.F-1; and provided the high radiation sampling system (HRSS) combustible gas monitoring capability for the drywell is OPERABLE; restore the inoperable CHANNEL(s) to OPERABLE status within 30 days or be in at least HOT SHUTDOWN within the next 12 hours.

c. With the number of OPERABLE accident monitoring instrumentation CHANNEL(s) less than the Minimum CHANNEL(s) shown in Table 3.2.F-1; and the HRSS combustible gas monitoring capability for the drywell inoperable; restore at least one inoperable CHANNEL to OPERABLE status within 7 days or be in at least HOT SHUTDOWN within the next 12 hours.

TABLE 3.2.F-1 (Continued)

ACCIDENT MONITORING INSTRUMENTATION

ACTION 63 -

a. With the number of OPERABLE accident monitoring instrumentation CHANNEL(s) less than the Required CHANNEL(s) shown in Table 3.2.F-1, restore the inoperable CHANNEL(s) to OPERABLE status prior to startup from a COLD SHUTDOWN of longer than 72 hours.

b. With the number of OPERABLE accident monitoring instrumentation CHANNEL(s) less than the Minimum CHANNEL(s) shown in Table 3.2.F-1, restore at least one of the inoperable CHANNEL(s) to OPERABLE status within 30 days or be in at least HOT SHUTDOWN within the next 12 hours.

TABLE 4.2.F-1

ACCIDENT MONITORING INSTRUMENTATION
SURVEILLANCE REQUIREMENTS

INSTRUMENTATION	CHANNEL CHECK	CHANNEL CALIBRATION	Applicable OPERATIONAL MODE(s)
1. Reactor Vessel Pressure	M	SA	1, 2
2. Reactor Vessel Water Level	M	SA	1, 2
3 Torus Water Level	M	A	1, 2
4. Torus Water Temperature	M	A	1, 2
5. Drywell Pressure - Wide Range	M	E	1, 2
6. Drywell Pressure - Narrow Range	M	Q	1, 2
7. Drywell Air Temperature	M	E	1, 2
8. Drywell Hydrogen/Oxygen Concentration - Analyzer and Monitor	M	Q	1, 2
9. Safety/Relief Valve Position Indicators - Acoustic & Temperature	M[c]	E	1, 2
10. (Source Range) Neutron Monitors	M	Q[b]	1, 2, 3
11. Drywell Radiation Monitors	M	E[a]	1, 2
12. Torus Pressure	M	Q	1, 2

TABLE 4.2.F-1 (Continued)

ACCIDENT MONITORING INSTRUMENTATION
SURVEILLANCE REQUIREMENTS

TABLE NOTATION

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

(b) Neutron detectors may be excluded from the CHANNEL CALIBRATION.

(c) CHANNEL CHECK of the Acoustic Monitors shall consist of verifying the instrument threshold levels.

3.2 - LIMITING CONDITIONS FOR OPERATION

G. Source Range Monitoring

At least the following source range monitor (SRM) channels shall be OPERABLE:

a. In OPERATIONAL MODE 2[a], three.

b. In OPERATIONAL MODE 3 and 4, two.

APPLICABILITY:

OPERATIONAL MODE(s) 2[a], 3, and 4.

ACTION:

1. In OPERATIONAL MODE 2[a] with one of the above required source range monitor CHANNEL(s) inoperable, at least 3 source range monitor CHANNEL(s) shall be restored to OPERABLE status within 4 hours or the reactor shall be in at least HOT SHUTDOWN within the next 12 hours.

2. In OPERATIONAL MODE(s) 3 or 4 with one or more of the above required source range monitor CHANNEL(s) inoperable, verify all insertable control rods to be fully inserted in the core and lock the reactor mode switch in the Shutdown position within one hour.

4.2 - SURVEILLANCE REQUIREMENTS

G. Source Range Monitoring

Each of the required source range monitor CHANNEL(s) shall be demonstrated OPERABLE by:

1. Verifying, prior to withdrawal of the control rods, that the SRM count rate is ≥3 cps with the detector fully inserted.

2. Performance of a CHANNEL CHECK at least once per:

   a. 12 hours in OPERATIONAL MODE 2[a], and

   b. 24 hours in OPERATIONAL MODE(s) 3 or 4.

3. Performance of a CHANNEL FUNCTIONAL TEST:

   a. Within 7 days prior to startup, and

   b. At least once per 31 days[b].

4. Performance of a CHANNEL CALIBRATION[c] at least once per 18 months[b].

---

a With IRM's on range 2 or below.

b The provisions of Specification 4.0.D are not applicable for entry into the applicable OPERATIONAL MODE(s) from OPERATIONAL MODE 1, provided the surveillance is performed within 12 hours after such entry.

c Neutron detectors may be excluded from the CHANNEL CALIBRATION.

3.2 - LIMITING CONDITIONS FOR OPERATION

H. Explosive Gas Monitoring

The explosive gas monitoring instrumentation CHANNEL(s) shown in Table 3.2.H-1 shall be OPERABLE with their alarm/trip setpoints set to ensure that the limits of Specification 3.8.H are not exceeded.

APPLICABILITY:

During offgas holdup system operation.

ACTION:

1. With an explosive gas monitoring instrumentation CHANNEL alarm/trip setpoint less conservative than required by the above specification, declare the CHANNEL inoperable and take the ACTION shown in Table 3.2.H-1.
2. With less than the minimum number of explosive gas monitoring instrumentation CHANNEL(s) OPERABLE, take the ACTION shown in Table 3.2.H-1. Restore the inoperable instrumentation to OPERABLE status within 30 days and, if unsuccessful, prepare and submit a Special Report to the Commission pursuant to Specification 6.9.B to explain why this inoperability was not corrected in a timely manner.
3. The provisions of Specification 3.0.C are not applicable.

4.2 - SURVEILLANCE REQUIREMENTS

H. Explosive Gas Monitoring

Each explosive gas monitoring instrumentation CHANNEL shall be demonstrated OPERABLE by the performance of the CHANNEL CHECK, CHANNEL FUNCTIONAL TEST and CHANNEL CALIBRATION operations at the frequencies shown in Table 4.2.H-1.

TABLE 4.2.H-1

EXPLOSIVE GAS MONITORING INSTRUMENTATION
SURVEILLANCE REQUIREMENTS

Functional Unit	CHANNEL CHECK	CHANNEL FUNCTIONAL TEST	CHANNEL CALIBRATION
MAIN CONDENSER OFFGAS TREATMENT SYSTEM EXPLOSIVE GAS MONITORING SYSTEM			
Hydrogen Monitor	D	M	Q

3.2 - LIMITING CONDITIONS FOR OPERATION

I. Suppression Chamber and Drywell Spray Actuation

The suppression chamber and drywell spray actuation instrumentation CHANNEL(s) shown in Table 3.2.I-1 shall be OPERABLE with their trip setpoints set consistent with the values shown in the Trip Setpoint column of Table 3.2.I-1.

APPLICABILITY:

OPERATIONAL MODE(s) 1, 2 and 3.

ACTION:

With a suppression chamber and drywell spray actuation instrumentation CHANNEL trip setpoint less conservative than the value shown in the Trip Setpoint column of Table 3.2.I-1, declare the CHANNEL inoperable and take the ACTION shown in Table 3.2.I-1.

4.2 - SURVEILLANCE REQUIREMENTS

I. Suppression Chamber and Drywell Spray Actuation

1. Each suppression chamber and drywell spray actuation instrumentation CHANNEL shall be demonstrated OPERABLE by the performance of the CHANNEL CHECK, CHANNEL FUNCTIONAL TEST and CHANNEL CALIBRATION operations at the frequencies shown in Table 4.2.I-1.

2. LOGIC SYSTEM FUNCTIONAL TEST(s) of all CHANNEL(s) shall be performed at least once per 18 months.

TABLE 3.2.I-1

SUPPRESSION CHAMBER AND DRYWELL SPRAY ACTUATION INSTRUMENTATION

Functional Unit	Trip Setpoint[a]	Minimum CHANNEL(s) per TRIP SYSTEM[c]	ACTION
1. Drywell Pressure - High (Permissive)	$0.5 \leq \rho \leq 1.5$ psig	2	80
2. Reactor Vessel Water Level -Low (Permissive)	≥ -48 inches	1	80

ACTION

ACTION 80 -

a. With the number of OPERABLE CHANNEL(s) less than required by the Minimum OPERABLE CHANNEL(s) per TRIP SYSTEM requirement for one TRIP SYSTEM, place at least one inoperable CHANNEL in the tripped condition[b] within one hour or declare the suppression chamber and drywell sprays inoperable.

b. With the number of OPERABLE CHANNEL(s) less than required by the Minimum OPERABLE CHANNEL(s) per TRIP SYSTEM requirement for both TRIP SYSTEM(s), declare the suppression chamber and drywell sprays inoperable.

---

a Reactor vessel water level settings are expressed in inches above the top of active fuel (which is 360 inches above vessel zero).

b If an instrument is inoperable, it shall be placed (or simulated) in a tripped condition so that it will not prevent a containment spray.

c A CHANNEL may be placed in an inoperable status for up to 2 hours for required surveillance without placing the CHANNEL in the tripped condition provided the Functional Unit maintains Suppression Chamber and Drywell Spray Actuation capability.

3.2 - LIMITING CONDITIONS FOR OPERATION

J. Feedwater Pump Trip

The feedwater pump trip instrumentation CHANNEL(s) shown in Table 3.2.J-1 shall be OPERABLE with their trip setpoints set consistent with the values shown in the Trip Setpoint column of Table 3.2.J-1.

APPLICABILITY:

OPERATIONAL MODE 1.

ACTION:

With a feedwater pump trip instrumentation CHANNEL trip setpoint less conservative than value shown in the Trip Setpoint column of Table 3.2.J-1, declare the CHANNEL inoperable and take the ACTION shown in Table 3.2.J-1.

4.2 - SURVEILLANCE REQUIREMENTS

J. Feedwater Pump Trip

1. Each feedwater pump trip instrumentation CHANNEL shall be demonstrated OPERABLE by the performance of the CHANNEL CHECK, CHANNEL FUNCTIONAL TEST and CHANNEL CALIBRATION operations at the frequencies shown in Table 4.2.J-1.

2. LOGIC SYSTEM FUNCTIONAL TEST(s) of all CHANNEL(s) shall be performed at least once per 18 months.

TABLE 3.2.J-1

FEEDWATER PUMP TRIP INSTRUMENTATION

Functional Unit	Trip Setpoint[a]	Minimum CHANNEL(s)[b]	ACTION
Reactor Vessel Water Level -High	≤ 201 inches	2	90

ACTION

ACTION 90 - a. With the number of OPERABLE CHANNEL(s) one less than required by the Minimum CHANNEL(s) requirement, restore the inoperable CHANNEL to OPERABLE status within 7 days or place the inoperable CHANNEL in the tripped condition within the next 8 hours.

b. With the number of OPERABLE CHANNEL(s) two less than required by the Minimum CHANNEL(s) requirement, restore at least one of the inoperable CHANNEL(s) to OPERABLE status within 72 hours or be in at least STARTUP within the next 8 hours.

---

a Reactor vessel water level settings are expressed in inches above the top of active fuel (which is 360 inches above vessel zero).

b A CHANNEL may be placed in an inoperable status for up to 2 hours for required surveillance without placing the CHANNEL in the tripped condition.

3/4.2.E Control Rod Block Actuation Instrumentation

The control rod block functions are provided to prevent excessive control rod withdrawal so that the MINIMUM CRITICAL POWER RATIO (MCPR) does not go below the MCPR fuel cladding integrity Safety Limit. During shutdown conditions, control rod block instrumentation initiates withdrawal blocks to ensure that all control rods remain inserted to prevent inadvertent criticality.

The trip logic for this function is one-out-of-n; e.g., any trip of one of the six average power range monitors (APRMs), eight intermediate range monitors (IRMs), or four source range monitors (SRMs), will result in a rod block. The minimum instrument CHANNEL requirements assure sufficient instrumentation to assure that the single failure criterion is met. The minimum instrument CHANNEL requirements for the rod block monitor may be reduced by one for a short period of time to allow for maintenance, testing, or calibration.

The APRM rod block function is flow-biased and prevents a significant reduction in MCPR, especially during operation at reduced flow. The APRM provides gross core protection, i.e., limits the gross withdrawal of control rods in the normal withdrawal sequence.

In the REFUEL and STARTUP/HOT STANDBY OPERATIONAL MODE(s), the APRM rod block function setpoint is significantly reduced to provide the same type of protection in the REFUEL and STARTUP/HOT STANDBY OPERATIONAL MODE(s) as the APRM flow-biased rod block does in the RUN OPERATIONAL MODE, i.e., prevents control rod withdrawal before a scram is reached.

The rod block monitor (RBM) function provides local protection of the core, i.e., the prevention of transition boiling in a local region of the core for a single rod withdrawal error. The trip setting is flow-biased. At low power, the worst-case withdrawal of a single control rod without rod block action will not violate the fuel cladding integrity Safety Limit. Thus the RBM rod block function is not required below the specified power level. The worst-case single control rod withdrawal error is analyzed for each reload to assure that, with the specific trip settings, rod withdrawal is blocked before the MCPR reaches the fuel cladding integrity Safety Limit.

The IRM rod block function provides local as well as gross core protection. The scaling arrangement is such that the trip setting is less than a factor of ten above the indicated level. Analysis of the worst-case accident results in rod block action before MCPR approaches the MCPR fuel cladding integrity Safety Limit.

A downscale indication on an APRM is an indication that the instrument has failed or is not sufficiently sensitive. In either case, the instrument will not respond to changes in control rod motion, and the control rod motion is thus prevented.

The SRM rod blocks of low count rate and the detector not fully inserted assure that the SRMs are not withdrawn from the core prior to commencing rod withdrawal for startup. The scram discharge volume, and high water level rod block provide annunciation for operator action. The alarm setpoint has been selected to provide adequate time to allow for the determination of the cause for the level increase and corrective action prior to automatic scram initiation.

BASES

3/4.2.F Accident Monitoring Instrumentation

Instrumentation is provided to monitor sufficient accident conditions to adequately assess important variables and provide the operators with the necessary information to complete the appropriate mitigation actions. OPERABILITY of the instrumentation listed provides adequate monitoring of the containment following a loss-of-coolant accident. Information from this instrumentation will provide the operator with a detailed knowledge of the conditions resulting from the accident; based on this information, the operator can make logical decisions regarding post accident recovery. Allowable outage times are based on diverse instrumentation availability for guiding the operator should an accident occur, and on the low probability of an instrument being out-of-service concurrent with an accident. As noted in the surveillance requirements, the instrumentation CHANNEL(s) associated with Torus Pressure provides a dual function and is shared in common with the Drywell Pressure (Narrow and Wide Ranges) instrumentation. This instrumentation is identified in response to Generic Letter 82-33 and the associated NRC Safety Evaluation Report, and some instrumentation is included in accordance with the response to Generic Letter 83-36.

3/4.2.G Source Range Monitoring Instrumentation

The source range monitors (SRM) provide the operator with the status of the neutron flux in the core at very low power levels during startup and shutdown. The consequences of reactivity accidents are functions of the initial neutron flux. Therefore, the requirements for a minimum count rate assures that any transient, should it occur, begins at or above the initial value used in the analyses of transients from cold conditions. Two OPERABLE SRM CHANNEL(s) are adequate to monitor the approach to criticality using homogeneous patterns of scattered control rod withdrawal. Three OPERABLE SRMs provide an added conservatism. When the intermediate range monitors are on scale, adequate information is available without the SRMs and they can be retracted.

3/4.2.H Explosive Gas Monitoring Instrumentation

Instrumentation is provided to monitor the concentrations of potentially explosive mixtures in the off-gas (waste) holdup system to prevent a possible uncontrolled release via this pathway. This instrumentation is included in accordance with Generic Letter 89-01.

3/4.2.I Suppression Chamber and Drywell Spray Actuation Instrumentation

Instrumentation is provided to monitor the parameters which are necessary to permit initiation of the suppression chamber and drywell spray mode of the low pressure coolant injection/ containment cooling system to condense steam in the containment atmosphere. The spray mode does not significantly affect the rise of drywell pressure following a loss of coolant accident, but does result in quicker depressurization following completion of the blowdown.

3.3 - LIMITING CONDITIONS FOR OPERATION

A. SHUTDOWN MARGIN (SDM)

The SHUTDOWN MARGIN (SDM) shall be equal to or greater than:

1. 0.38% Δk/k with the highest worth control rod analytically determined, or
2. 0.28% Δk/k with the highest worth control rod determined by test.

APPLICABILITY:

OPERATIONAL MODE(s) 1, 2, 3, 4, and 5.

ACTION:

With the SHUTDOWN MARGIN less than specified:

1. In OPERATIONAL MODE 1 or 2, restore the required SHUTDOWN MARGIN within 6 hours or be in at least HOT SHUTDOWN within the next 12 hours.
2. In OPERATIONAL MODE 3 or 4, immediately verify all insertable control rods to be fully inserted and suspend all activities that could reduce the SHUTDOWN MARGIN. In OPERATIONAL MODE 4, establish SECONDARY CONTAINMENT INTEGRITY within 8 hours.
3. In OPERATIONAL MODE 5, suspend CORE ALTERATION(s) and other activities that could reduce the SHUTDOWN MARGIN and fully insert all insertable control rods within 1 hour. Establish SECONDARY CONTAINMENT INTEGRITY within 8 hours.

4.3 - SURVEILLANCE REQUIREMENTS

A. SHUTDOWN MARGIN

The SHUTDOWN MARGIN shall be determined to be equal to or greater than that specified at any time during the operating cycle:

1. By demonstration, prior to or during the first startup after each refueling outage.
2. Within 24 hours after detection of a withdrawn control rod that is immovable, as a result of excessive friction or mechanical interference, or known to be unscrammable. The required SHUTDOWN MARGIN shall be verified acceptable with an increased allowance for the withdrawn worth of the immovable or unscrammable control rod.
3. By calculation, prior to each fuel movement during the fuel loading sequence.

3.3 - LIMITING CONDITIONS FOR OPERATION

C. Control Rod OPERABILITY

All control rods shall be OPERABLE.

APPLICABILITY:

OPERATIONAL MODE(s) 1 and 2.

ACTION:

1. With one control rod inoperable due to being immovable as a result of excessive friction or mechanical interference, or known to be unscrammable:
   a. Within one hour:
      1) Verify that the inoperable control rod, if withdrawn, is separated from all other inoperable withdrawn control rods by at least two control cells in all directions.
      2) Disarm the associated directional control valves[a] either:
         a) Electrically, or
         b) Hydraulically by closing the drive water and exhaust water isolation valves.
   b. With the provisions of ACTION 1.a above not met, be in at least HOT SHUTDOWN within the next 12 hours.

4.3 - SURVEILLANCE REQUIREMENTS

C. Control Rod OPERABILITY

1. When above the low power setpoint of the RWM, all withdrawn control rods not required to have their directional control valves disarmed electrically or hydraulically shall be demonstrated OPERABLE by moving each control rod at least one notch:
   a. At least once per 7 days, and
   b. Within 24 hours when any control rod is immovable as a result of excessive friction or mechanical interference, or known to be unscrammable.
2. All control rods shall be demonstrated OPERABLE by performance of Surveillance Requirements 4.3.D, 4.3.F, 4.3.G, 4.3.H and 4.3.I.

a May be rearmed intermittently, under administrative control, to permit testing associated with restoring the control rod to OPERABLE status.

3.3 - LIMITING CONDITIONS FOR OPERATION

D. Maximum Scram Insertion Times

The maximum scram insertion time of each control rod from the fully withdrawn position to 90% insertion, based on de-energization of the scram pilot valve solenoids as time zero, shall not exceed 7 seconds.

APPLICABILITY:

OPERATIONAL MODE(s) 1 and 2.

ACTION:

With the maximum scram insertion time of one or more control rods exceeding 7 seconds:

1. Declare the control rod(s) exceeding the above maximum scram insertion time inoperable, and
2. When operation is continued with three or more control rods with maximum scram insertion times in excess of 7 seconds, perform Surveillance Requirement 4.3.D.3 at least once per 60 days of power operation.

With the provisions of the ACTION(s) above not met, be in at least HOT SHUTDOWN within 12 hours.

4.3 - SURVEILLANCE REQUIREMENTS

D. Maximum Scram Insertion Times

The maximum scram insertion time of the control rods shall be demonstrated through measurement with reactor coolant pressure greater than 800 psig and, during single control rod scram time tests, with the control rod drive pumps isolated from the accumulators:

1. For all control rods prior to THERMAL POWER exceeding 40% of RATED THERMAL POWER:
   a. following CORE ALTERATION(s), or
   b. after a reactor shutdown that is greater than 120 days,
2. For specifically affected individual control rods[a] following maintenance on or modification to the control rod or control rod drive system which could affect the scram insertion time of those specific control rods, and
3. For at least 10% of the control rods, on a rotating basis, at least once per 120 days of power operation.

a The provisions of Specification 4.0.D are not applicable provided this surveillance is conducted prior to exceeding 40% of RATED THERMAL POWER.

3.3 - LIMITING CONDITIONS FOR OPERATION

H. Control Rod Drive Coupling

All control rods shall be coupled to their drive mechanisms.

APPLICABILITY:

OPERATIONAL MODE(s) 1, 2, and 5[a].

ACTION:

1. In OPERATIONAL MODE 1 or 2 with one control rod not coupled to its associated drive mechanism, within 2 hours:

   a. If permitted by the RWM, insert the control rod drive mechanism to accomplish recoupling and verify recoupling by withdrawing the control rod, and:

      1) Observing any indicated response of the nuclear instrumentation, and

      2) Demonstrating that the control rod will not go to the overtravel position.

   b. If not permitted by the RWM or, if recoupling is not accomplished in accordance with ACTION 1.a above, then declare the control rod inoperable, fully insert the control rod and disarm the associated directional control valves[b] either:

      1) Electrically, or

4.3 - SURVEILLANCE REQUIREMENTS

H. Control Rod Drive Coupling

Each affected control rod shall be demonstrated to be coupled to its drive mechanism by verifying that the control rod drive does not go to the overtravel position:

1. Deleted.

2. Anytime the control rod is withdrawn to the "Full out" position, and

3. Following maintenance on or modification to the control rod or control rod drive system which could have affected the control rod drive coupling integrity.

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a In OPERATIONAL MODE 5, this Specification is applicable for withdrawn control rods and is not applicable to control rods removed per Specification 3.10.I or 3.10.J.

b May be rearmed intermittently, under administrative control, to permit testing associated with restoring the control rod to OPERABLE status.

3.3 - LIMITING CONDITIONS FOR OPERATION

I. Control Rod Position Indication System

All control rod position indicators shall be OPERABLE.

APPLICABILITY:

OPERATIONAL MODE(s) 1, 2, and 5[a].

ACTION:

1. In OPERATIONAL MODE 1 or 2 with one or more control rod position indicators inoperable, within one hour either:

   a. Determine the position of the control rod by an alternate method, or

   b. Move the control rod to a position with an OPERABLE position indicator, or

   c. Declare the control rod inoperable, fully insert the inoperable withdrawn control rod(s), and disarm the associated directional control valves[b] either:

      1) Electrically, or

      2) Hydraulically by closing the drive water and exhaust water isolation valves.

4.3 - SURVEILLANCE REQUIREMENTS

I. Control Rod Position Indication System

The control rod position indication system shall be determined OPERABLE by verifying:

1. At least once per 24 hours that the position of each control rod is indicated.

2. That the indicated control rod position changes during the movement of the control rod drive when performing Surveillance Requirement 4.3.C.1.

3. Deleted.

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a In OPERATIONAL MODE 5, this Specification is applicable for withdrawn control rods and is not applicable to control rods removed per Specification 3.10.I or 3.10.J.

b May be rearmed intermittently, under administrative control, to permit testing associated with restoring the control rod(s) to OPERABLE status.

3.3 - LIMITING CONDITIONS FOR OPERATION

L. Rod Worth Minimizer (RWM)

The rod worth minimizer (RWM) shall be OPERABLE.

APPLICABILITY:

OPERATIONAL MODE(s) 1 and 2[a], when THERMAL POWER is less than or equal to 20% of RATED THERMAL POWER.

ACTION:

With the RWM inoperable, verify control rod movement and compliance with the prescribed control rod pattern by a second licensed operator or technically qualified individual who is present at the reactor control console. Otherwise, control rod movement may be made only by actuating the manual scram or placing the reactor mode switch in the Shutdown position.

4.3 - SURVEILLANCE REQUIREMENTS

L. Rod Worth Minimizer (RWM)

The RWM shall be demonstrated OPERABLE:

1. By verifying that the control rod patterns and sequence input to the RWM computer are correctly loaded following any loading of the program into the computer.
2. In OPERATIONAL MODE 2 within 8 hours prior to withdrawal of control rods for the purpose of making the reactor critical:
   a. by verifying proper indication of the selection error of at least one out-of-sequence control rod.
   b. by verifying the rod block function.
3. In OPERATIONAL MODE 1 prior to reducing THERMAL POWER below 20% of RATED THERMAL POWER:
   a. by verifying proper indication of the selection error of at least one out-of-sequence control rod.
   b. by verifying the rod block function.

a Entry into OPERATIONAL MODE 2 and withdrawal of selected control rods is permitted for the purpose of determining the OPERABILITY of the RWM prior to withdrawal of control rods for the purpose of bringing the reactor to criticality.

BASES

3/4.3.A SHUTDOWN MARGIN

A sufficient SHUTDOWN MARGIN ensures that 1) the reactor can be made subcritical from all operating conditions, 2) the reactivity transients associated with postulated accident conditions are controllable within acceptable limits, and 3) the reactor will be maintained sufficiently subcritical to preclude inadvertent criticality in the shutdown condition.

The SHUTDOWN MARGIN limitation is a restriction to be applied principally to a new refueling pattern. Satisfaction of the limitation must be determined at the time of loading and must be such that it will apply to the entire subsequent fuel cycle. This determination is provided by core design calculations and administrative control of fuel loading patterns. These procedures include restrictions to allow only those intermediate fuel assembly configurations that have been shown to provide the required SHUTDOWN MARGIN.

Since core reactivity values will vary through core life as a function of fuel depletion and poison burnup, the demonstration of SHUTDOWN MARGIN will be performed during xenon free conditions and adjusted to 68°F to accommodate the current moderator temperature. The generalized form is that the reactivity of the core loading will be limited so the core can be made subcritical by at least $R + 0.38\% \Delta k/k$ or $R + 0.28\% \Delta k/k$, as appropriate, with the strongest control rod fully withdrawn and all others fully inserted. Two different values are supplied in the Limiting Condition for Operation to provide for the different methods of determination of the highest control rod worth, either analytically or by test. This is due to the reduced uncertainty in the SHUTDOWN MARGIN test when the highest worth control rod is determined by demonstration. When SHUTDOWN MARGIN is determined by calculations not associated with a test, additional margin must be added to the specified SHUTDOWN MARGIN limit to account for uncertainties in the calculation.

The value of R in units of % $\Delta k/k$ is the difference between the calculated beginning-of-life core reactivity, at the beginning of the operating cycle, and the calculated value of maximum core reactivity at any time later in the operating cycle, where it would be greater than at the beginning. The value of R shall include the potential SHUTDOWN MARGIN loss assuming full B_4C settling in all inverted poison tubes present in the core. R must be a positive quantity or zero and a new value of R must be determined for each new fuel cycle.

The value of % $\Delta k/k$ in the above expression is provided as a finite, demonstrable, subcriticality margin. This margin is verified using an in-sequence control rod withdrawal at the beginning-of-life fuel cycle conditions. This assures subcriticality with not only the strongest fully withdrawn but at least an $R + 0.28\% \Delta k$ (or $0.38\% \Delta k$) margin beyond this condition. This reactivity characteristic has been a basic assumption in the analysis of plant performance and can be best demonstrated at the time of fuel loading, but the margin must also be determined anytime a control rod is incapable of insertion following a scram signal. Any control rod that is immovable as a result of excessive friction or mechanical interference, or is known to be unscrammable, per Specification 3.3.C, is considered to be incapable of insertion following a scram signal. It is important to note that a control rod can be electrically immovable, but scrammable, and no increase in SHUTDOWN MARGIN is required for these control rods.

BASES

During MODE 5, adequate SDM is required to ensure that the reactor does not reach criticality during control rod withdrawals. An evaluation of each in-vessel fuel movement during fuel loading (including shuffling fuel within the core) is required to ensure adequate SDM is maintained during refueling. This evaluation ensures that the intermediate loading patterns are bounded by the safety analyses for the final core loading pattern. For example, bounding analyses that demonstrate adequate SDM for the most reactive configurations during the refueling may be performed to demonstrate acceptability of the entire fuel movement sequence. These bounding analyses include additional margins to the associated uncertainties. Spiral offload/reload sequences inherently satisfy the SR, provided the fuel assemblies are reloaded in the same configuration analyzed for the new cycle. Removing fuel from the core will always result in an increase in SDM.

3/4.3.B Reactivity Anomalies

During each fuel cycle, excess operating reactivity varies as fuel depletes and as any burnable poison in supplementary control is burned. The magnitude of this excess reactivity may be inferred from the critical rod configuration. As fuel burnup progresses, anomalous behavior in the excess reactivity may be detected by comparison of the critical rod pattern selected base states to the predicted rod inventory at that state. Power operating base conditions provide the most sensitive and directly interpretable data relative to core reactivity. Furthermore, using power operating base conditions permits frequent reactivity comparisons. Requiring a reactivity comparison at the specified frequency assures that a comparison will be made before the core reactivity change exceeds 1% $\Delta k/k$. Deviations in core reactivity greater than 1% $\Delta k/k$ are not expected and require thorough evaluation. A 1% $\Delta k/k$ reactivity limit is considered safe since an insertion of the reactivity into the core would not lead to transients exceeding design conditions of the reactor system.

3/4.3.C Control Rod OPERABILITY

Control rods are the primary reactivity control system for the reactor. In conjunction with the Reactor Protection System, the control rods provide the means for reliable control of reactivity changes to ensure the specified acceptable fuel design limits are not exceeded. This specification, along with others, assures that the performance of the control rods in the event of an accident or transient, meets the assumptions used in the safety analysis. Of primary concern is the trippability of the control rods. Other causes for inoperability are addressed in other Specifications following this one. However, the inability to move a control rod which remains trippable does not prevent the performance of the control rod's safety function.

The specification requires that a rod be taken out-of-service if it cannot be moved with drive pressure. Damage within the control rod drive mechanism could be a generic problem, therefore with a control rod immovable because of excessive friction or mechanical interference, operation of the reactor is limited to a time period which is reasonable to determine the cause of the inoperability and at the same time prevent operation with a large number of inoperable control rods. Control rods that are inoperable due to exceeding allowed scram times, but are movable by

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control rod drive pressure, need not be disarmed electrically if the shutdown margin provisions are met for each position of the affected rod(s).

If the rod is fully inserted and then disarmed electrically or hydraulically, it is in a safe position of maximum contribution to shutdown reactivity. (Note: To disarm the drive electrically, four amphenol-type plug connectors are removed from the drive insert and withdrawal solenoids, rendering the drive immovable. This procedure is equivalent to valving out the drive and is preferred, as drive water cools and minimizes crud accumulation in the drive.). If it is disarmed electrically in a non-fully inserted position, that position shall be consistent with the SHUTDOWN MARGIN limitation stated in Specification 3.3.A. This assures that the core can be shut down at all times with the remaining control rods, assuming the strongest OPERABLE control rod does not insert. The occurrence of more than eight inoperable control rods could be indicative of a generic control rod drive problem which requires prompt investigation and resolution.

In order to reduce the potential for Control Rod Drive (CRD) damage and more specifically, collet housing failure, a program of disassembly and inspection of CRDs is conducted during or after each refueling outage. This program follows the recommendations of General Electric SIL-139 with nondestructive examination results compiled and reported to General Electric on collet housing cracking problems.

The required surveillance intervals are adequate to determine that the rods are OPERABLE and not so frequent as to cause excessive wear on the system components.

<u>3/4.3.D</u> <u>Control Rod Maximum Scram Insertion Times;</u>

<u>3/4.3.E</u> <u>Control Rod Average Scram Insertion Times; and</u>

<u>3/4.3.F</u> <u>Four Control Rod Group Scram Insertion Times</u>

These specifications ensure that the control rod insertion times are consistent with those used in the safety analyses. The control rod system is analyzed to bring the reactor subcritical at a rate fast enough to prevent fuel damage, i.e., to prevent the MCPR from becoming less than the fuel cladding integrity Safety Limit. The analyses demonstrate that if the reactor is operated within the limitation set in Specification 3.11.C, the negative reactivity insertion rates associated with the scram performance (as adjusted for statistical variation in the observed data) result in protection of the MCPR Safety Limit.

Analysis of the limiting power transient shows that the negative reactivity rates, resulting from the scram with the average response of all the drives, as given in the above specification, provide the required protection, and MCPR remains greater than the fuel cladding integrity SAFETY LIMIT. In the analytical treatment of most transients, 290 milliseconds are allowed between a neutron sensor reaching the scram point and the start of motion of the control rods. This is adequate and conservative when compared to the typically observed time delay of about 210 milliseconds. Approximately 90 milliseconds after neutron flux reaches the trip point, the pilot scram valve

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solenoid de-energizes and 120 milliseconds later the control rod motion is estimated to actually begin. However, 200 milliseconds rather than 120 milliseconds is conservatively assumed for this time interval in the transient analyses and is also included in the allowable scram insertion times specified in Specifications 3.3.D, 3.3.E, and 3.3.F. In the statistical treatment of the limiting transients, a statistical distribution of total scram delay is used rather than the bounding value described above.

The performance of the individual control rod drives is monitored to assure that scram performance is not degraded. Observed plant data or Technical Specification limits were used to determine the average scram performance used in the transient analyses, and the results of each set of control rod scram tests performed during the current cycle are compared against earlier results to verify that the performance of the control rod insertion system has not changed significantly.

If test results should be determined to fall outside of the statistical population defining the scram performance characteristics used in the transient analyses, a re-determination of thermal margin requirements is undertaken as required by Specification 3.11.C. A smaller test sample than that required by these specifications is not statistically significant and should not be used in the re-determination of thermal margins. Individual control rod drives with excessive scram times can be fully inserted into the core and de-energized in the manner of an inoperable rod drive provided the allowable number of inoperable control rod drives is not exceeded. In this case, the scram speed of the drive shall not be used as a basis in the re-determination of thermal margin requirements. For excessive average scram insertion times, only the individual control rods in the two-by-two array which exceed the allowed average scram insertion time are considered inoperable.

The scram times for all control rods are measured at the time of each refueling outage. Experience with the plant has shown that control drive insertion times vary little through the operating cycle; hence no re-assessment of thermal margin requirements is expected under normal conditions. The history of drive performance accumulated to date indicates that the 90% insertion times of new and overhauled drives approximate a normal distribution about the mean which tends to become skewed toward longer scram times as operating time is accumulated. The probability of a drive not exceeding the mean 90% insertion time by 0.75 seconds is greater than 0.999 for a normal distribution. The measurement of the scram performance of the drives surrounding a drive, which exceeds the expected range of scram performance, will detect local variations and also provide assurance that local scram time limits are not exceeded. Continued monitoring of other drives exceeding the expected range of scram times provides surveillance of possible anomalous performance.

The test schedule provides reasonable assurance of detection of slow drives before system deterioration beyond the limits of Specification 3.3.C. The program was developed on the basis of the statistical approach outlined above and judgement. The occurrence of scram times within the limits, but significantly longer than average, should be viewed as an indication of a systematic problem with control rod drives, especially if the number of drives exhibiting such scram times exceeds eight, which is the allowable number of inoperable rods.

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3/4.3.G Control Rod Scram Accumulators

The control rod scram accumulators are part of the control rod drive system and are provided to ensure that the control rods scram under varying reactor conditions. The control rod scram accumulators store sufficient energy to fully insert a control rod at any reactor vessel pressure. The accumulator is a hydraulic cylinder with a free floating piston. The piston separates the water used to scram the control rods from the nitrogen which provides the required energy. The scram accumulators are necessary to scram the control rods within the required insertion times.

Control rods with inoperable accumulators are declared inoperable and Specification 3.3.C then applies. This prevents a pattern of inoperable accumulators that would result in less reactivity insertion on a scram than has been analyzed even though control rods with inoperable accumulators may still be inserted with normal drive water pressure. OPERABILITY of the accumulator ensures that there is a means available to insert the control rods even under the most unfavorable depressurization of the reactor.

3/4.3.H Control Rod Drive Coupling

Control rod dropout accidents can lead to significant core damage. If coupling integrity is maintained, the possibility of a rod drop accident is eliminated. Neutron instrumentation response to rod movement may provide verification that a rod is following its drive. Absence of such response to drive movement may indicate an uncoupled condition, or may be due to the lack of proximity of the drive to the instrumentation. However, the overtravel position feature provides a positive check, as only uncoupled drives may reach this position.

3/4.3.I Control Rod Position Indication System (RPIS)

In order to ensure that the control rod patterns can be followed and therefore that other parameters are within their limits, the control rod position indication system must be OPERABLE. Normal control rod position is displayed by two-digit indication to the operator from position 00 to 48. Each even number is a latching position, whereas each odd number provides information while the rod is in-motion and inputs for rod drift annunciation. The ACTION statement provides for the condition where no positive information is displayed for a large portion or all of the rod's travel. Usually, only one digit of one or two of a rod's positions is unavailable with a faulty RPIS, and the control rod may be located in a known position. However, there are several alternate methods for determining control rod position including the full core display, the four rod display, the rod worth minimizer, and the process computer. Another method to determine position would be to move the control rod, by single notch movement, to a position with an OPERABLE position indicator. The original position would then be established and the control rod could be returned to its original position by single notch movement. As long as no control rod drift alarms are received, the position of the control rod would then be known.

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3/4.3.J Control Rod Drive Housing Support

The control rod housing support restricts the outward movement of a control rod to less than 3 inches in the extremely remote event of a housing failure. The amount of reactivity which could be added by this small amount of rod withdrawal, which is less than a normal single withdrawal increment, will not contribute to any damage to the primary coolant system. The design basis is given in Section 4.6.3.5 of the UFSAR. This support is not required if the reactor coolant system is at atmospheric pressure, since there would then be no driving force to rapidly eject a drive housing.

3/4.3.K Scram Discharge Volume Vent and Drain Valves

The scram discharge volume is required to be OPERABLE so that it will be available when needed to accept discharge water from the control rods during a reactor scram and will isolate the reactor coolant system from the containment when required. The operability of the scram discharge volume vent and drain valves assures the proper venting and draining of the volume, so that water accumulation in the volume does not occur. These specifications designate the minimum acceptable level of scram discharge volume vent and drain valve OPERABILITY, provide for the periodic verification that the valves are open, and for the testing of these valves under reactor scram conditions during each refueling outage.

3/4.3.L Rod Worth Minimizer

Control rod withdrawal and insertion sequences are established to assure that the maximum insequence individual control rod or control rod segments which are withdrawn at any time during the fuel cycle could not be worth enough to result in a peak fuel enthalpy greater than 280 cal/gm in the event of a control rod drop accident. These sequences are developed prior to initial operation of the unit following any refueling outage and the requirement that an operator follow these sequences is supervised by the RWM or a second technically qualified individual. These sequences are developed to limit reactivity worth of control rods and, together with the integral rod velocity limiters and the action of the control rod drive system, limit potential reactivity insertion such that the results of a control rod drop accident will not exceed a maximum fuel energy content of 280 cal/gm. The peak fuel enthalpy of 280 cal/gm is below the energy content at which rapid fuel dispersal and primary system damage have been found to occur based on experimental data. Therefore, the energy deposited during a postulated rod drop accident is significantly less than that required for rapid fuel dispersal.

The analysis of the control rod drop accident was originally presented in Sections 7.9.3, 14.2.1.2, and 14.2.1.4 of the original SAR. Improvements in analytical capability have allowed a more refined analysis of the control rod drop accident which is discussed below.

Every operating cycle the peak fuel rod enthalpy rise is determined by comparing cycle specific parameters with the results of parametric analyses. This peak fuel rod enthalpy is then compared

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to the analysis limit of 280 cal/gm to demonstrate compliance for that operating cycle. If the cycle specific parameters are outside the range used in the parametric study, an extension of the enthalpy may be required. Some of the cycle specific parameters used in the analysis are: maximum control rod worth, Doppler coefficient, effective delayed neutron fraction and maximum four bundle local peaking factor. The NRC approved methodology listed in Specification 6.9.A.6 provides a detailed description of the methodology used in performing the rod drop analyses.

The rod worth minimizer provides automatic supervision to assure that out-of-sequence control rods will not be withdrawn or inserted, i.e., it limits operator deviations from planned withdrawal sequences (reference UFSAR Section 7.7.2). It serves as a backup to procedural control of control rod worth. In the event that the rod worth minimizer is out-of-service when required, a second licensed operator or other technically qualified individual who is present at the reactor console can manually fulfill the control rod pattern conformance function of the rod worth minimizer. In this case, the normal procedural controls are backed up by independent procedural controls to assure conformance.

3/4.3.M Rod Block Monitor

The rod block monitor (RBM) is designed to automatically prevent fuel damage in the event of erroneous rod withdrawal from locations of high power density during high power operation. Two channels are provided, and one of these may be bypassed from the console for maintenance and/or testing. Tripping of one of the channels will block erroneous rod withdrawal soon enough to prevent fuel damage. This system backs up the operator, who withdraws control rods according to a written sequence. The specified restrictions with one channel out-of-service conservatively assure that fuel damage will not occur due to rod withdrawal errors when this condition exists.

3/4.3.N Economic Generation Control System

Operation of the facility with the economic generation control system (EGC) (automatic flow control) is limited to the range of 65% to 100% of rated core flow. In this flow range and above 20% of RATED THERMAL POWER, the reactor could safely tolerate a rate of change of load of 8 MWe/sec (reference UFSAR Section 7.7.3.2). Limits within the EGC and the flow control system prevent rates of change greater than approximately 4 MWe/sec. When EGC is in operation, this fact will be indicated on the main control room console.

3.5 - LIMITING CONDITIONS FOR OPERATION

A. Emergency Core Cooling System - Operating

The emergency core cooling systems (ECCS) shall be OPERABLE with:

1. The core spray (CS) system consisting of two subsystems with each subsystem comprised of:
    a. One OPERABLE CS pump, and
    b. An OPERABLE flow path capable of taking suction from the suppression chamber and transferring the water through the spray sparger to the reactor vessel.
2. The low pressure coolant injection (LPCI) subsystem comprised of:
    a. Four OPERABLE LPCI pumps, and
    b. An OPERABLE flow path capable of taking suction from the suppression chamber and transferring the water to the reactor vessel.
3. The high pressure cooling injection (HPCI) system consisting of:
    a. One OPERABLE HPCI pump, and
    b. An OPERABLE flow path capable of taking suction from the suppression chamber and transferring the water to the reactor vessel.
4. The automatic depressurization system (ADS) with at least 5 OPERABLE ADS valves.

4.5 - SURVEILLANCE REQUIREMENTS

A. Emergency Core Cooling System - Operating

The ECCS shall be demonstrated OPERABLE by:

1. At least once per 31 days:
    a. For the CS system, the LPCI subsystem and the HPCI system:
        1) Verifying that the system piping from the pump discharge valve to the system isolation valve is filled with water.
        2) 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[a] position.
    b. For the HPCI system, verifying that the HPCI pump flow controller is in the correct position.
2. Verifying that, when tested pursuant to Specification 4.0.E:
    a. The CS pump in each subsystem develop a flow of at least 4500 gpm against a test line pressure corresponding to a reactor vessel pressure of ≥90 psig.

a Except that an automatic valve capable of automatic return to its ECCS position when an ECCS signal is present may be in position for another mode of operation.

3.5 - LIMITING CONDITIONS FOR OPERATION

APPLICABILITY:

OPERATIONAL MODE(s) 1, 2[b] and 3[b].

ACTION:

1. For the core spray system:

   a. With one CS subsystem inoperable, provided that the LPCI subsystem is OPERABLE, restore the inoperable CS subsystem to OPERABLE status within 7 days, or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

   b. With both CS subsystems inoperable, be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

2. For the LPCI subsystem:

   a. With one LPCI pump inoperable[d], provided that both CS subsystems are OPERABLE, restore the inoperable LPCI pump to OPERABLE status within 30 days, or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

4.5 - SURVEILLANCE REQUIREMENTS

   b. Three LPCI pumps together develop a flow of at least 14,500 gpm against a test line pressure corresponding to a reactor vessel pressure of ≥20 psig.

   c. The HPCI pump develops a flow of at least 5000 gpm against a system head corresponding to reactor vessel pressure, when steam is being supplied to the turbine between 920 and 1005 psig[c].

3. At least once per 18 months:

   a. For the CS system, the LPCI subsystem, and the HPCI system, verify each system/subsystem actuates on an actual or simulated automatic initiation signal. Actual injection of coolant into the reactor vessel may be excluded from this test.

   b. For the HPCI system, verifying that:

      1) The system develops a flow of ≥5000 gpm against a system head corresponding to reactor vessel pressure, when steam is being supplied to the turbine between 150 and 350 psig[c].

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b The HPCI system and ADS are not required to be OPERABLE when reactor steam dome pressure is ≤150 psig.

d The provisions of Specification 3.9.A, Actions 4.a or 6.b are applicable to the LPCI subsystem such that with an inoperable diesel generator, for the remaining OPERABLE diesel generator, both LPCI pumps (and their associated flow path) associated with that OPERABLE diesel generator, shall be OPERABLE.

c The provisions of Specification 4.0.D are not applicable provided the surveillance is performed within 12 hours after reactor steam pressure is adequate to perform the test.

3.5 - LIMITING CONDITIONS FOR OPERATION

b. With the LPCI subsystem otherwise inoperable[d], provided that both CS subsystems are OPERABLE, restore the LPCI subsystem to OPERABLE status within 7 days or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

c. With the LPCI subsystem and one or both CS subsystems inoperable, be in at least HOT SHUTDOWN within 12 hours and in COLD SHUTDOWN within the next 24 hours.

3. With the HPCI system inoperable, provided both CS subsystems, the LPCI subsystem, the ADS and the Isolation Condenser (IC) system are OPERABLE, restore the HPCI system to OPERABLE status within 14 days or be in at least HOT SHUTDOWN within the next 12 hours and reduce reactor steam dome pressure to ≤150 psig within the following 24 hours.

4. For the ADS:

a. With one of the above required ADS valves inoperable, provided the HPCI system, both CS subsystems and three LPCI pumps are OPERABLE, restore the inoperable ADS valve to OPERABLE

4.5 - SURVEILLANCE REQUIREMENTS

2) The pump suction is automatically transferred from the condensate storage tank to the suppression chamber on a condensate storage tank water level - low signal and on a suppression chamber water level - high signal.

c. Performing a CHANNEL CALIBRATION of the CS and LPCI system discharge line "keep filled" alarm instrumentation.

d. Deleted.

4. At least once per 18 months for the ADS:

a. Verify the ADS actuates on an actual or simulated automatic initiation signal. Actual valve actuation may be excluded from this test.

b. Manually opening each ADS valve when the reactor steam dome pressure is ≥150 psig[c] and observing that either:

1) The turbine control valve or turbine bypass valve position responds accordingly, or

2) There is a corresponding change in the measured steam flow.

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d The provisions of Specification 3.9.A, Actions 4.a or 6.b are applicable to the LPCI subsystem such that with an inoperable diesel generator, for the remaining OPERABLE diesel generator, <u>both</u> LPCI pumps (and their associated flow path) associated with that OPERABLE diesel generator, shall be OPERABLE.

c The provisions of Specification 4.0.D are not applicable provided the surveillance is performed within 12 hours after reactor steam pressure is adequate to perform the test.

3.5 - LIMITING CONDITIONS FOR OPERATION

status within 14 days or be in at least HOT SHUTDOWN within the next 12 hours and reduce reactor steam dome pressure to ≤150 psig within the following 24 hours.

b. With two or more of the above required ADS valves inoperable, be in at least HOT SHUTDOWN within 12 hours and reduce reactor steam dome pressure to ≤150 psig within the following 24 hours.

5. With an ECCS discharge line "keep filled" pressure alarm instrumentation CHANNEL inoperable, perform Surveillance Requirement 4.5.A.1.a.1) for CS and LPCI at least once per 24 hours.

6. Deleted.

7. In the event an ECCS system 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.B 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.

4.5 - SURVEILLANCE REQUIREMENTS

3.5 - LIMITING CONDITIONS FOR OPERATION

D. Isolation Condenser

The isolation condenser (IC) system shall be OPERABLE.

APPLICABILITY:

OPERATIONAL MODE(s) 1, 2 and 3 with reactor steam dome pressure >150 psig.

ACTION:

With the IC system inoperable, operation may continue provided the HPCI system is OPERABLE; restore the IC system to OPERABLE status within 14 days or be in at least HOT SHUTDOWN within the next 12 hours and reduce reactor steam dome pressure to ≤150 psig within the following 24 hours.

4.5 - SURVEILLANCE REQUIREMENTS

D. Isolation Condenser

The IC system shall be demonstrated OPERABLE:

1. At least once per 24 hours by verifying the shell side water volume and the shell side water temperature to be within limits.
2. 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.
3. At least once per 18 months by verifying the IC system actuates on an actual or simulated automatic initiation signal.
4. At least once per 5 years by verifying the system heat removal capability.

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With the HPCI system inoperable, adequate core cooling is assured by the OPERABILITY of the redundant and diversified Automatic Depressurization System and both the CS system and LPCI subsystem. In addition, the Isolation Condenser (IC) system, a system for which no credit is taken in the safety analysis, will automatically initiate on a sustained reactor high pressure condition. The HPCI out-of-service period of 14 days is based on the demonstrated OPERABILITY of redundant and diversified low pressure core cooling systems and the IC system.

The surveillance requirements provide adequate assurance that the HPCI system will be OPERABLE when required. Although all active components are testable and full flow can be demonstrated by recirculation through a test loop during reactor operation, a complete system functional test requires a reactor shutdown. The pump discharge piping is maintained full to prevent water hammer damage and to provide cooling at the earliest moment.

Upon failure of the HPCI system to function properly after a small break loss-of-coolant, the Automatic Depressurization System (ADS) automatically causes all OPERABLE main steamline relief valves to open, depressurizing the reactor so that flow from the low pressure core cooling systems can enter the core in time to limit fuel cladding temperature to less than 2200°F. ADS is conservatively required to be OPERABLE whenever reactor vessel pressure exceeds 150 psig. This pressure is substantially below that for which the low pressure core cooling systems can provide adequate core cooling for events requiring ADS.

ADS automatically controls the five main steamline relief valves although safety analyses support a minimum of 4 OPERABLE valves. It is therefore appropriate to permit one valve to be out-of-service for up to 14 days without materially reducing system reliability. A manual actuation of each ADS valve is performed to verify that the valve and solenoid are functioning properly and that no blockage exists in the ADS discharge lines. This is demonstrated by the response of the turbine control or bypass valve or by a change in the measured steam flow or by any other method suitable to verify steam flow. Adequate reactor steam dome pressure must be available to perform this test to avoid damaging the valve. Sufficient time is therefore allowed after the required pressure is achieved to perform this test once only. The pressure specified for this test is that pressure recommended by the valve manufacturer. Reactor startup is allowed prior to performing this test because valve OPERABILITY and the setpoints for overpressure protection are verified, per ASME requirements, prior to valve installation. Thus, a footnote is included in this SR to indicate that 4.0.D does not apply.

To preserve single failure criteria, a minimum of two independent OPERABLE low-pressure ECCS subsystems/loops are required in OPERATIONAL MODE(s) 4 and 5 to ensure adequate vessel inventory makeup in the event of an inadvertent vessel draindown. Only a single LPCI pump is required per loop because of the large injection capacity. All of the ECCS may be inoperable provided the reactor head is removed, the reactor cavity is flooded, the spent fuel gates are removed, and the water level is maintained within the limits required by the Refueling Operations specifications.

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3/4.5.C Suppression Chamber

The suppression chamber is required to be OPERABLE as part of the ECCS to ensure that a sufficient supply of water is available to the HPCI and CS systems and the LPCI subsystem in the event of a LOCA. This limit on suppression chamber minimum water volume ensures that sufficient water is available to permit recirculation cooling flow to the core. The OPERABILITY of the suppression chamber in OPERATIONAL MODE(s) 1, 2 or 3 is also required by Specification 3.7.K.

Repair work might require making the suppression chamber inoperable. This specification will permit those repairs to be made and concurrently provide assurance that the irradiated fuel has an adequate cooling water supply when the suppression chamber must be made inoperable, including draining, in OPERATIONAL MODE(s) 4 or 5.

In OPERATIONAL MODE(s) 4 and 5 the suppression chamber minimum required water volume is reduced because the reactor coolant is maintained at or below 212°F. Since pressure suppression is not required below 212°F, the minimum water volume is based on net positive suction head (NPSH), recirculation volume and vortex prevention plus a safety margin for conservatism. With the suppression chamber water level less than the required limit, all ECCS subsystems are inoperable unless they are aligned to an OPERABLE condensate storage tank. When the suppression chamber level is less than 8 feet, the CS system or the LPCI subsystem is considered OPERABLE only if it can take suction from the condensate storage tank, and the condensate storage tank water level is sufficient to provide the required NPSH for the CS or LPCI pumps. Therefore, a verification that either the suppression chamber water level is greater than or equal to 8 feet or that CS or LPCI is aligned to take suction from the condensate storage tank and the condensate storage tank contains greater than or equal to 140,000 gallons of water, ensures CS or LPCI can supply at least 50,000 gallons of make-up water to the reactor pressure vessel. The CS suction is uncovered at the 90,000 gallon level.

3/4.5.D Isolation Condenser

The isolation condenser is provided for core decay heat removal following reactor isolation from the main condenser and reactor scram. The isolation condenser has a heat removal capacity (252.5 x 10^6 BTU/hour) sufficient to handle the decay heat production at 300 seconds following a scram. Following a reactor scram and an isolation from the main condenser, water will be lost from the reactor vessel through the relief valves during the first 300 seconds. This represents a minor loss relative to the vessel inventory.

The system may be manually initiated at any time. The system is automatically initiated on high reactor pressure in excess of 1070 psig sustained for 17 seconds. The time delay is provided to prevent unnecessary actuation of the system during anticipated turbine trips. Automatic initiation is provided to minimize the coolant loss following isolation from the main condenser. To be considered OPERABLE, the shell side of the isolation condenser must contain at least 20,000 gallons of water. Make-up water to the shell side of the isolation condenser is provided by the

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condensate transfer pumps from the condensate storage tank. The condensate transfer pumps are OPERABLE from on-site power. The preferred source of make-up water for the Isolation Condenser is the clean demineralized water system. The fire protection system is also available as make-up water.

3.6 - LIMITING CONDITIONS FOR OPERATION

B. Jet Pumps

All jet pumps shall be OPERABLE and flow indication shall be OPERABLE on at least 19 jet pumps.

APPLICABILITY:

OPERATIONAL MODE(s) 1 and 2.

ACTION:

1. With one or more jet pumps inoperable for other than inoperable flow indication, be in at least HOT SHUTDOWN within 12 hours.

2. With flow indication inoperable for two or more jet pumps, flow indication shall be restored such that at least 19 jet pumps have OPERABLE flow indication within 12 hours or be in at least HOT SHUTDOWN within the next 12 hours.

4.6 - SURVEILLANCE REQUIREMENTS

B. Jet Pumps

All jet pumps shall be demonstrated OPERABLE as follows:

1. During two loop operation, at least once per 24 hours while greater than 25% of RATED THERMAL POWER by determining recirculation loop flow, total core flow and individual jet pump flow for each jet pump and verifying that no two of the following conditions occur when both recirculation pumps are operating in accordance with Specification 3.6.C:

   a. The indicated recirculation pump flow differs by >10% from the established speed-flow characteristics.

   b. The indicated total core flow differs by >10% from the established total core flow value derived from established core plate ΔP/core flow relationships.

   c. The indicated flow of any individual jet pump differs from the established patterns by >10%.

   d. The provisions of Specification 4.0.D are not applicable provided that the surveillance is performed within 24 hours after exceeding 25% of RATED THERMAL POWER.

3.6 - LIMITING CONDITIONS FOR OPERATION

D. Idle Recirculation Loop Startup

An idle recirculation loop shall not be started unless the temperature differential between the reactor pressure vessel and the bottom head coolant temperature is within limits[a], and:

1. When both loops have been idle, unless the temperature differential between the reactor coolant within the idle loop to be started up and the coolant in the reactor pressure vessel is within limits, or

2. When only one loop has been idle, unless the temperature differential between the reactor coolant within the idle and operating recirculation loops is within limits.

APPLICABILITY:

OPERATIONAL MODE(s) 1, 2, 3 and 4.

ACTION:

With temperature differences and/or flow rates exceeding the above limits, suspend startup of any recirculation loop, restore the parameter(s) to within limits within 30 minutes, and determine if the reactor coolant system is acceptable for continued operation within 72 hours.

Otherwise, be in HOT SHUTDOWN in 12 hours and COLD SHUTDOWN within the following 24 hours.

4.6 - SURVEILLANCE REQUIREMENTS

D. Idle Recirculation Loop Startup

The temperature differentials and flow rate shall be determined to be within the limits within 15 minutes prior to startup of an idle recirculation loop.

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a Below 25 psig reactor pressure, this temperature differential is not applicable.

3.6 - LIMITING CONDITIONS FOR OPERATION

E. Safety Valves

The safety valve function of the 9 reactor coolant system safety valves shall be OPERABLE in accordance with the specified code safety valve function lift settings[a] established as:

1 safety valve[b] @1135 psig ±1%
2 safety valves @1240 psig ±1%
2 safety valves @1250 psig ±1%
4 safety valves @1260 psig ±1%

APPLICABILITY:

OPERATIONAL MODE(s) 1, 2 and 3.

ACTION:

1. With the safety valve function of one or more of the above required safety valves inoperable, be in at least HOT SHUTDOWN within 12 hours and in COLD SHUTDOWN within the next 24 hours.

2. Deleted.

4.6 - SURVEILLANCE REQUIREMENTS

E. Safety Valves

1. Deleted.

2. At least once per 18 months, 1/2 of the safety valves shall be removed, set pressure tested and reinstalled or replaced with spares that have been previously set pressure tested and stored in accordance with manufacturer's recommendations. At least once per 40 months, the safety valves shall be rotated such that all 9 safety valves are removed, set pressure tested and reinstalled or replaced with spares that have been previously set pressure tested and stored in accordance with manufacturer's recommendations.

a The lift setting pressure shall correspond to ambient conditions of the valves at nominal operating temperatures and pressures.

b Target Rock combination safety/relief valve.

3.6 - LIMITING CONDITIONS FOR OPERATION

F. Relief Valves

5 reactor coolant system relief valves and the reactuation time delay of two relief valves shall be OPERABLE with the following settings:

Relief Function Setpoint (psig)
Open
≤ 1112 psig
≤ 1112 psig
≤ 1135 psig
≤ 1135 psig
≤ 1135 psig[a]

APPLICABILITY:

OPERATIONAL MODE(s) 1, 2 and 3.

ACTION:

1. With one or more relief valves open, provided that suppression pool average water temperature is <110°F, take action to close the open relief valve(s); if suppression pool average water temperature is ≥110°F place the reactor mode switch in the Shutdown position.

4.6 - SURVEILLANCE REQUIREMENTS

F. Relief Valves

1. The relief valve function and the reactuation time delay function instrumentation shall be demonstrated OPERABLE by performance of a:
   a. CHANNEL FUNCTIONAL TEST of the relief valve function at least once per 18 months, and a
   b. CHANNEL CALIBRATION and LOGIC SYSTEM FUNCTIONAL TEST of the entire system at least once per 18 months.
2. Deleted.

a Target Rock combination safety/relief valve.

3.6 - LIMITING CONDITIONS FOR OPERATION

2. With the relief valve function and/or the reactuation time delay of one of the above required reactor coolant system relief valves inoperable, restore the inoperable relief valve function and the reactuation time delay function to OPERABLE status within 14 days or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

3. With the relief valve function and/or the reactuation time delay of more than one of the above required reactor coolant system relief valves inoperable, be in at least HOT SHUTDOWN within 12 hours and in COLD SHUTDOWN within the next 24 hours.

4. Deleted.

4.6 - SURVEILLANCE REQUIREMENTS

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3/4.6.A Recirculation Loops

3/4.6.B Jet Pumps

3/4.6.C Recirculation Pumps

3/4.6.D Idle Recirculation Loop Startup

The reactor coolant recirculation system is designed to provide a forced coolant flow through the core to remove heat from the fuel. The reactor coolant recirculation system consists of two recirculation pump loops external to the reactor vessel. These loops provide the piping path for the driving flow of water to the reactor vessel jet pumps. The operation of the reactor coolant recirculation system is an initial condition assumed in the design basis loss-of-coolant accident (LOCA). During a LOCA caused by a recirculation loop pipe break, the intact loop is assumed to provide coolant flow during the first few seconds of the accident. The analyses assumes both loops are operating at the same flow prior to the accident. If a LOCA occurs with a flow mismatch between the two loops, the analysis conservatively assumes the pipe break is in the loop with the higher flow.

A plant specific analysis has been performed assuming only one operating recirculation loop. This analysis has demonstrated that in the event of a LOCA caused by a pipe break in the operating recirculation loop, the ECCS response will provide adequate core cooling. The transient analyses of Chapter 15 of the FSAR have also been performed for single recirculation loop operation and demonstrate sufficient flow coastdown characteristics to maintain fuel thermal margins during the abnormal operational transients analyzed provided the MCPR fuel cladding integrity Safety Limit is increased as noted by Specification 2.1.B. The Reactor Protection System APRM scram and control rod block setpoints are also required to be adjusted to account for the different response of the reactor and different relationships between recirculation drive flow and reactor core flow. During single loop operation for greater than 24 hours, the idle recirculation pump is electrically prohibited from starting until ready to resume two loop operation. This is done to prevent a cold water injection transient caused by an inadvertent pump startup.

Jet pump OPERABILITY is an explicit assumption in the design basis LOCA analysis. The capability of reflooding the core to two-thirds core height is dependent upon the structural integrity of the jet pumps. If a beam holding a jet pump in place fails, the jet pump suction and mixer sections could become displaced, resulting in a larger flow area through the jet pump and a lower core flooding elevation. This could adversely affect the water level in the core during the reflood phase of a LOCA as well as the assumed blowdown flow during a LOCA.

The surveillance requirements for jet pumps are designed to detect a significant degradation in jet pump performance that precedes a jet pump failure. Significant degradation is indicated if more than one of the three specified criteria confirms unacceptable deviations from established patterns or relationships. A break in a jet pump decreases the flow resistance characteristic of the external piping loop causing the recirculation pump to operate at a higher flow condition when compared to previous operation. The agreement of indicated core plate dp and core flow relationships provides

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assurance that the recirculation flow is not bypassing the core through inactive or broken jet pumps. The change in the flow rate of the failed jet pump produces a change in the indicated flow rate of that pump relative to the other pumps in that loop. Comparison of the data with a normal pattern provides the indication necessary to detect a failed jet pump. Allowable deviations from the established patterns have been developed based on operation. Since refueling activities (fuel assembly replacement or shuffle, as well as any modification to fuel support orifice size or core plate bypass flow) can affect the relationship between core flow, jet pump flow, and recirculation loop flow, these relationships may need to be reestablished each cycle. Similarly, initial entry into extended single loop operation may also require establishment of these relationships. During the initial weeks of operation under such conditions, while base-lining new "established patterns," engineering judgement of the daily surveillance results is used to detect significant abnormalities which could indicate jet pump failure.

The accuracy of the core flow measurement system is assumed in the derivation of the Safety Limit MINIMUM CRITICAL POWER RATIO. An analysis assuming a loss of flow indication for three jet pumps resulted in uncertainties within the values assumed for the core flow measurement system in the Safety Limit MINIMUM CRITICAL POWER RATIO calculation for both two loop operation and single loop operation. Therefore, plant operation with loss of flow indication in up to two jet pumps is acceptable as long as each jet pump is on a separate riser and no more than one calibrated double tap jet pump per loop is affected.

Recirculation pump speed mismatch limits are in compliance with the ECCS LOCA analysis design criteria. For some limited low probability events with the recirculation loop operating with large speed differences, it is possible for the LPCI loop selection logic to select the wrong loop for injection. Above 80% of RATED THERMAL POWER, the LPCI selection logic is expected to function at a speed differential of 15%. Below 80% of RATED THERMAL POWER, the loop select logic would be expected to function at a speed differential of 20%. Therefore, this specification provides a margin of 5% in pump speed differential before a problem could arise.

In order to prevent undue stress on the vessel nozzles and bottom head region, the recirculation loop temperatures shall be within the limit specified in the Dresden Administrative Technical Requirements prior to startup of an idle loop. The loop temperature must also be within the limit specified in the Dresden Administrative Technical Requirements to prevent thermal shock to the recirculation pump and recirculation nozzles. Since the coolant in the bottom of the vessel is at a lower temperature than the coolant in the upper regions of the core, undue stress on the vessel would result if the temperature difference was greater than the limit specified in the Dresden Administrative Technical Requirements. Additionally, asymmetric speed operation of the recirculation pumps during idle loop startup induces levels of jet pump riser vibration that are higher than normal. The specific limitation of the rated pump speed limit specified in the Dresden Administrative Technical Requirements for the operating recirculation pump prior to the start of the idle recirculation pump ensures that the recirculation pump speed mismatch requirements are maintained.

In addition to suspending startup of an idle recirculation loop not meeting the temperature limits, the temperature parameters must be restored within 30 minutes. The 30 minute completion time

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reflects the urgency of restoring the parameters to within the analyzed range. Most violations will not be severe, and the activity can be accomplished in this time in a controlled manner.

Besides restoring operation within limits, an evaluation is required to determine if operation can continue. The evaluation must verify the reactor coolant system integrity remains acceptable and must be completed if continued operation is desired. Several methods may be used, including comparison with pre-analyzed transients in the stress analyses, new analyses, or inspection of the components.

The 72 hour completion time is reasonable to accomplish the evaluation of a mild violation. More severe violations may require special, event specific stress analyses or inspections. A favorable evaluation must be completed if continued operation is desired.

3/4.6.E Safety Valves

3/4.6.F Relief Valves

The American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code requires the reactor pressure vessel be protected from overpressure during upset conditions by self-actuated safety valves. As part of the nuclear pressure relief system, the size and number of safety valves are selected such that peak pressure in the nuclear system will not exceed the ASME Code limits for the reactor coolant pressure boundary. The overpressure protection system must accommodate the most severe pressurization transient. Evaluations have determined that the most severe transient is the closure of all the main steam line isolation valves followed by a reactor scram on high neutron flux. The analysis results demonstrate that the design safety valve capacity is capable of maintaining reactor pressure below the ASME Code limit of 110% of the reactor pressure vessel design pressure.

The relief valve function is not assumed to operate in response to any accident, but are provided to remove the generated steam flow upon turbine stop valve closure coincident with failure of the turbine bypass system. The relief valve opening pressure settings are sufficiently low to prevent the need for safety valve actuation following such a transient.

Each of the five relief valves discharge to the suppression chamber via a dedicated relief valve discharge line. Steam remaining in the relief valve discharge line following closure can condense, creating a vacuum which may draw suppression pool water up into the discharge line. This condition is normally alleviated by the vacuum breakers; however, subsequent actuation in the presence of an elevated water leg can result in unacceptably high thrust loads on the discharge piping. To prevent this, the relief valves have been designed to ensure that each valve which closes will remain closed until the normal water level in the relief valve discharge line is restored. The opening and closing setpoints are set such that all pressure induced subsequent actuation are limited to the two lowest set valves. These two valves are equipped with additional logic which functions in conjunction with the setpoints to inhibit valve reopening during the elevated water leg duration time following each closure.

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Each safety/relief valve is equipped with diverse position indicators which monitor the tailpipe acoustic vibration and temperature. Either of these provide sufficient indication of safety/relief valve position for normal operation.

3/4.6.G Leakage Detection Systems

The RCS leakage detection systems required by this specification are provided to monitor and detect leakage from the reactor coolant pressure boundary. Limits on leakage from the reactor coolant pressure boundary are required so that appropriate action can be taken before the integrity of the reactor coolant pressure boundary is impaired. Leakage detection systems for the reactor coolant system are provided to alert the operators when leakage rates above the normal background levels are detected and also to supply quantitative measurement of leakage rates. Leakage from the reactor coolant pressure boundary inside the drywell is detected by at least one or two independently monitored variables, such as sump level changes and drywell atmosphere radioactivity levels. The means of quantifying leakage in the drywell is the drywell floor drain sump pumps. With the drywell floor drain sump pump system inoperable, no other form of monitoring can provide the equivalent information. However, primary containment atmosphere sampling for radioactivity can provide indication of changes in leakage rates.

3/4.6.H Operational Leakage

The allowable leakage rates from the reactor coolant system have been based on the predicted and experimentally observed behavior of cracks in pipes. The normally expected background leakage due to equipment design and the detection capability of the instrumentation for determining system leakage was also considered. The evidence obtained from experiments suggests that for leakage somewhat greater than that specified for UNIDENTIFIED LEAKAGE the probability is small that the imperfection or crack associated with such leakage would grow rapidly. However, in all cases, if the leakage rates exceed the values specified or the leakage is located and known to be PRESSURE BOUNDARY LEAKAGE, the reactor will be shutdown to allow further investigation and corrective action.

An UNIDENTIFIED LEAKAGE increase of more than 2 gpm within a 24 hour period is an indication of a potential flaw in the reactor coolant pressure boundary and must be quickly evaluated. Although the increase does not necessarily violate the absolute UNIDENTIFIED LEAKAGE limit, IGSCC susceptible components must be determined not to be the source of the leakage within the required completion time.

3/4.6.I Chemistry

The water chemistry limits of the reactor coolant system are established to prevent damage to the reactor materials in contact with the coolant. Chloride limits are specified to prevent stress

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corrosion cracking of the stainless steel. The effect of chloride is not as great when the oxygen concentration in the coolant is low, thus the 0.2 ppm limit on chlorides is permitted during POWER OPERATION.

Conductivity measurements are required on a continuous basis since changes in this parameter are an indication of abnormal conditions. When the conductivity is within limits, the pH, chlorides and other impurities affecting conductivity must also be within their acceptable limits. With the conductivity meter inoperable, additional samples must be analyzed to ensure that the chlorides are not exceeding the limits.

Action 1 permits temporary operation with chemistry limits outside of the limits required in OPERATIONAL MODE 1 without requiring Commission notification. The surveillance requirements provide adequate assurance that concentrations in excess of the limits will be detected in sufficient time to take corrective action.

3/4.6.J Specific Activity

The limitations on the specific activity of the primary coolant ensure that the 2 hour thyroid and whole body doses resulting from a main steam line failure outside the containment during steady state operation will not exceed small fractions of the dose guidelines of 10 CFR 100. The values for the limits on specific activity represent interim limits based upon a parametric evaluation by the NRC of typical site locations. These values are conservative in that specific site parameters, such as site boundary location and meteorological conditions, were not considered in this evaluation.

The ACTION statement permitting POWER OPERATION to continue for limited time periods with the primary coolant's specific activity greater than 0.2 microcuries per gram DOSE EQUIVALENT I-131, but less than or equal to 4.0 microcuries per gram DOSE EQUIVALENT I-131, accommodates possible iodine spiking phenomenon which may occur following changes in THERMAL POWER. Information obtained on iodine spiking will be used to assess the parameters associated with spiking phenomena. A reduction in frequency of isotopic analysis following power changes may be permissible if justified by the data obtained.

Closing the main steam line isolation valves prevents the release of activity to the environs should a steam line rupture occur outside containment. The surveillance requirements provide adequate assurance that excessive specific activity levels in the reactor coolant will be detected in sufficient time to take corrective action.

3/4.6.K Pressure/Temperature Limits

All components in the reactor coolant system are designed to withstand the effects of cyclic loads due to system temperature and pressure changes. These cyclic loads are introduced by normal load transients, reactor trips, and startup and shutdown operations. The various categories of load cycles used for design purposes are provided in Section 4 of the FSAR. During startup and

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shutdown, the rates of temperature and pressure changes are limited so that the maximum specified heatup and cooldown rates are consistent with the design assumptions and satisfy the stress limits for cyclic operation.

During heatup, the thermal gradients in the reactor vessel wall produce thermal stresses which vary from compressive at the inner wall to tensile at the outer wall. These thermal induced compressive stresses tend to alleviate the tensile stresses induced by the internal pressure. Therefore, a pressure temperature curve based on steady state conditions, i.e., no thermal stresses, represents a lower bound of all similar curves for finite heatup rates when the inner wall of the vessel is treated as the governing location.

The heatup analysis also covers the determination of pressure-temperature limitations for the case in which the outer wall of the vessel becomes the controlling location. The thermal gradients established during heatup produce tensile stresses which are already present. The thermal induced stresses at the outer wall of the vessel are tensile and are dependent on both the rate of heatup and the time along the heatup ramp; therefore, a lower bound curve similar to that described for the heatup of the inner wall cannot be defined. Subsequently, for the cases in which the outer wall of the vessel becomes the stress controlling location, each heatup rate of interest must be analyzed on an individual basis.

The pressure-temperature limit lines shown in Figure 3.6.K-1, for operating conditions; Inservice Hydrostatic Testing (curve A), Non-Nuclear Heatup/Cooldown (curve B), and Core Critical Operation (curve C). The curves have been established to be in conformance with Appendix G to 10 CFR Part 50 and Regulatory Guide 1.99 Revision 2, and take into account the change in reference nil-ductility transition temperature (RT_{NDT}) as a result of neutron embrittlement. The adjusted reference temperature (ART) of the limiting vessel material is used to account for irradiation effects.

Three vessel regions are considered for the development of the pressure-temperature curves: 1) the core beltline region; 2) the non-beltline region (other than the closure flange region); and 3) the closure flange region. The beltline region is defined as that region of the reactor vessel that directly surrounds the effective height of the reactor core and is subject to an RT_{NDT} adjustment to account for radiation embrittlement. The non-beltline and closure flange regions receive insufficient fluence to necessitate an RT_{NDT} adjustment. These regions contain components which include; the reactor vessel nozzles, closure flanges, top and bottom head plates, control rod drive penetrations, and shell plates that do not directly surround the reactor core. Although the closure flange region is a non-beltline region, it is treated separately for the development of the pressure-temperature curves to address 10CFR Part 50 Appendix G requirements.

In evaluating the adequacy of the steel which comprises the reactor vessel, it is necessary that the following be established: 1) the RT_{NDT} for all vessel and adjoining materials; 2) the relationship between RT_{NDT} and integrated neutron flux (fluence, at energies greater than one Mev); and 3) the fluence at the location of a postulated flaw.

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Boltup Temperature

The initial RT_{NDT} of the main closure flanges, the shell and head materials connecting to these flanges, and connecting welds is 10°F; however, the vertical electroslag welds which terminate immediately below the vessel flange have an RT_{NDT} of 40°F. Therefore, the minimum allowable boltup temperature is established as 100°F (RT_{NDT} + 60°F) which includes a 60°F conservatism required by the original ASME Code of construction.

Curve A - Hydrotesting

As indicated in curve A of Figure 3.6.K-1 for system hydrotesting, the minimum metal temperature of the reactor vessel shell is 100°F for reactor pressures less than 312 psig. This 100°F minimum boltup temperature is based on a RT_{NDT} of 40°F for the electroslag weld immediately below the vessel flange and a 60°F conservatism required by the original ASME Code of construction. At reactor pressures greater than 312 psig, the minimum vessel metal temperature is established as 130°F. The 130°F minimum temperature is based on a closure flange region RT_{NDT} of 40°F and a 90°F conservatism required by 10CFR Part 50 Appendix G for pressure in excess of 20% of the preservice hydrostatic test pressure (1563 psig). At approximately 650 psig the effects of pressurization are more limiting than the boltup stresses at the closure flange region, hence a family of non-linear curves intersect the 130°F vertical line. Beltline as well as non-beltline curves have been provided to allow separate monitoring of the two regions. Beltline curves as a function of vessel exposure for 12, 14 and 16 effective full power years (EFPY) are presented to allow the use of the appropriate curve up to 16 EFPY of operation.

A typical sequence involved in pressure testing is a heatup to the required temperature and then pressurization to the required pressure for the inspection. During the heatup, at 100°F/hour or less, Curve B is the governing curve. Since the vessel is not pressurized during the heatup, Curves A and B are the same. When temperatures are stabilized to within 20°F/hour rates, at temperatures above those required by curve A, pressurization begins, at which point Curve A is the governing curve. During the inspection period with the vessel at the required pressure, temperature changes are limited to 20°F/hour.

Curve B - Non-Nuclear Heatup/Cooldown

Curve B of Figure 3.6.K-1 applies during heatups with non-nuclear heat (e.g., recirculation pump heat) and during cooldowns when the reactor is not critical (e.g., following a scram). The curve provides the minimum reactor vessel metal temperatures based on the most limiting vessel stress. As indicated by the vertical 100°F line, the boltup stresses at the closure flange region are most limiting for reactor pressures below approximately 110 psig. For reactor pressures greater than approximately 110 psig, pressurization and thermal stresses become more limiting than the boltup stresses, which is reflected by the nonlinear portion of curve B. The non-linear portion of the curve is dependent on non-beltline and beltline regions, with the beltline region temperature limits having been adjusted to account for vessel irradiation (up to a

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vessel exposure of 16 EFPY). The non-beltline region is limiting between approximately 110 psig and 830 psig. Above approximately 803 psig, the beltline region becomes limiting.

Curve C - Core Critical Operation

Curve C, the core critical operation curve shown in Figure 3.6.K-1, is generated in accordance with 10CFR Part 50 Appendix G which requires core critical pressure-temperature limits to be 40°F above any curve A or B limits. Since curve B is more limiting, (curve C is curve B plus 40°F.

The actual shift in RT_{NDT} of the vessel material will be established periodically during operation by removing and evaluating, in accordance with ASTM E185-73 and 10CFR Part 50, Appendix H, irradiated reactor vessel material specimens installed near the inside wall of the reactor vessel in the core area. The irradiated specimens can be used with confidence in predicting reactor vessel material transition temperature shift. The operating limit curves of Figure 3.6.K-1 shall be adjusted, as required, on the basis of the specimen data and recommendations of Regulatory Guide 1.99, Revision 2.

3/4.6.L Reactor Steam Dome Pressure

The reactor steam dome pressure is an assumed initial condition of Design Basis Accidents and transients and is also an assumed value in the determination of compliance with reactor pressure vessel overpressure protection criteria. The reactor steam dome pressure of ≤1005 psig is an initial condition of the vessel overpressure protection analysis. This analysis assumes an initial maximum reactor steam dome pressure and evaluates the response of the pressure relief system, primarily the safety valves, during the limiting pressurization transient. The determination of compliance with the overpressure criteria is dependent on the initial reactor steam dome pressure; therefore, the limit on this pressure ensures that the assumptions of the overpressure protection analysis are conserved.

3/4.6.M Main Steam Line Isolation Valves

Double isolation valves are provided on each of the main steam lines to minimize the potential leakage paths from the containment in case of a line break. Only one valve in each line is required to maintain the integrity of the containment, however, single failure considerations require that two valves be OPERABLE. The surveillance requirements are based on the operating history of this type of valve. The maximum closure time has been selected to contain fission products and to ensure the core is not uncovered following line breaks. The minimum closure time is consistent with the assumptions in the safety analyses to prevent pressure surges.

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3/4.6.N Structural Integrity

The inspection programs for ASME Code Class 1, 2 and 3 components ensure that the structural integrity of these components will be maintained at an acceptable level throughout the life of the plant.

The inservice inspection program for ASME Code Class 1, 2 and 3 components will be performed in accordance with Section XI of the ASME Boiler and Pressure Vessel Code and applicable addenda as required by 10 CFR Part 50.55a(g) except where specific written relief has been granted by the NRC pursuant to 10 CFR Part 50.55a(g)(6)(i).

3/4.6.O Shutdown Cooling - HOT SHUTDOWN

3/4.6.P Shutdown Cooling - COLD SHUTDOWN

Irradiated fuel in the reactor pressure vessel generates decay heat during normal and abnormal shutdown conditions, potentially resulting in an increase in the temperature of the reactor coolant. This decay heat is required to be removed such that the reactor coolant temperature can be reduced in preparation for performing refueling, maintenance operations or for maintaining the reactor in cold shutdown conditions. Systems capable of removing decay heat are therefore required to perform these functions.

A single shutdown cooling mode loop provides sufficient heat removal capability for removing core decay heat and mixing to assure accurate temperature indication, however, single failure considerations require that two loops be OPERABLE or that alternate methods capable of decay heat removal be demonstrated and that an alternate method of coolant mixing be in operation.

3.7 - LIMITING CONDITIONS FOR OPERATION

A. PRIMARY CONTAINMENT INTEGRITY

PRIMARY CONTAINMENT INTEGRITY shall be maintained.

APPLICABILITY:

OPERATIONAL MODE(s) 1, 2[a] and 3.

ACTION:

Without PRIMARY CONTAINMENT INTEGRITY, restore PRIMARY CONTAINMENT INTEGRITY within 1 hour or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

4.7 - SURVEILLANCE REQUIREMENTS

A. PRIMARY CONTAINMENT INTEGRITY

PRIMARY CONTAINMENT INTEGRITY shall be demonstrated:

1. After each closing of each penetration subject to Type B testing, except the primary containment air locks, if opened following Type A or B test, by leak rate testing the seals with gas at $\geq P_a$ (48 psig), and verifying that when the measured leakage rate for these seals is added to the leakage rates determined pursuant to Surveillance Requirement 4.7.B.4 for all other Type B and C penetrations, the combined leakage rate is $\leq 0.60\ L_a$.

2. At least once per 31 days by verifying that all primary containment penetrations[b] not capable of being closed by OPERABLE containment automatic isolation valves and required to be closed during accident conditions are closed, except for valves that are open under administrative control as permitted by Specification 3.7.D.

3. By verifying each primary containment air lock is in compliance with the requirements of Specification 3.7.C.

4. By verifying the suppression chamber is in compliance with the requirements of Specification 3.7.K.

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a See Special Test Exception 3.12.A.

b Except valves, blind flanges, and deactivated automatic valves which are located inside the containment. Valves and blind flanges in high radiation areas may be verified by use of administrative controls. These penetrations shall be verified closed during each COLD SHUTDOWN except such verification need not be performed when the primary containment has not been de-inerted since the last verification or more often than once per 92 days.

3.7 - LIMITING CONDITIONS FOR OPERATION

B. Primary Containment Leakage

Primary containment leakage rates shall be limited to:

1. An overall integrated leakage rate of $\leq L_a$ which is defined as 1.6 percent by weight of the containment air per 24 hours at P_a (48 psig).

2. A combined leakage rate of $\leq 0.60\ L_a$ for all primary containment penetrations, except[a] for main steam line isolation valves, subject to Type B and C tests when pressurized to P_a (48 psig).

3. ≤ 11.5 scfh for any one main steam line isolation valve when tested at P_t (25 psig)[a].

APPLICABILITY:

When PRIMARY CONTAINMENT INTEGRITY is required per Specification 3.7.A.

ACTION:

With the measured combined leakage rate for all primary containment penetrations subject to Type B and C tests $>0.60\ L_a$, restore the combined leakage rate to $\leq 0.60\ L_a$, within 1 hour. Otherwise, be in HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

4.7 - SURVEILLANCE REQUIREMENTS

B. Primary Containment Leakage

The primary containment leakage rates shall be demonstrated at the following test schedule and shall be determined in conformance with the criteria, methods and provisions specified in Appendix J of 10CFR Part 50, as modified by approved exemptions:

1. Three Type A overall integrated containment leakage rate tests shall be conducted at approximately equal intervals during shutdown at $\geq P_a$ (48 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.

2. If the results of any periodic Type A test are $>0.75\ L_a$, the test schedule for subsequent Type A tests shall be reviewed and approved by the Commission. If the results of two consecutive Type A tests are $>0.75\ L_a$, a Type A test shall be performed at intervals in accordance with 10 CFR Part 50, Appendix J, as modified by approved exemptions, until the results of two consecutive Type A tests are $\leq 0.75\ L_a$, at which time the above test schedule may be resumed.

3. The accuracy of each Type A test shall be verified by a supplemental test which:

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

a Exemption from Appendix J to 10CFR Part 50.

3.7 - LIMITING CONDITIONS FOR OPERATION

4.7 - SURVEILLANCE REQUIREMENTS

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

c. Requires the quantity of gas to be bled from the containment during the supplemental test to be equivalent to at least 25% of the total measured leakage at $\geq P_a$ (48 psig).

4. Type B and C tests shall be conducted with gas at $\geq P_a$ (48 psig) at intervals in accordance with 10 CFR 50, Appendix J, as modified by approved exemptions, except for tests involving:

   a. Air locks which shall be leak tested in accordance with Surveillance Requirement 4.7.C,

   b. Main steam line isolation valves[a] which shall be leak tested at $\geq P_t$ (25 psig)[a], and

   c. Bolted double-gasketed seals which shall be leak tested at $\geq P_a$ (48 psig) following each closure of the seal and at intervals in accordance with 10 CFR 50, Appendix J, as modified by approved exemptions.

a Exemption from Appendix J to 10CFR Part 50.

3.7 - LIMITING CONDITIONS FOR OPERATION

c. Otherwise, be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

2. With the primary containment air lock interlock mechanism inoperable, restore the air lock interlock mechanism to OPERABLE status within 24 hours, or lock at least one air lock door closed and verify that the door is locked closed at least once per 31 days. Personnel entry and exit through the airlock is permitted provided one OPERABLE air lock door remains locked closed at all times and an individual is dedicated to assure that both air lock doors are not opened simultaneously.

3. With the primary containment air lock inoperable, except as a result of an inoperable air lock door or interlock mechanism, maintain at least one air lock door closed; restore the inoperable air lock to OPERABLE status within 24 hours or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

4.7 - SURVEILLANCE REQUIREMENTS

3.7 - LIMITING CONDITIONS FOR OPERATION

H. Drywell - Suppression Chamber Differential Pressure

Differential pressure between the drywell and the suppression chamber shall be ≥1.0 psid[a].

APPLICABILITY:

OPERATIONAL MODE 1, during the time period:

1. Beginning within 24 hours after THERMAL POWER is >15% of RATED THERMAL POWER following startup, and
2. Ending within 24 hours prior to reducing THERMAL POWER to ≤15% of RATED THERMAL POWER preliminary to a scheduled reactor shutdown.

ACTION:

1. With the drywell - suppression chamber differential pressure less than the above limit, restore the required differential pressure within 24 hours or reduce THERMAL POWER to <15% RATED THERMAL POWER within the next 8 hours.
2. With the drywell - suppression chamber differential pressure instrumentation CHANNEL inoperable, restore the inoperable CHANNEL to OPERABLE status within 30 days or reduce THERMAL POWER to <15% RATED THERMAL POWER within the next 8 hours.

4.7 - SURVEILLANCE REQUIREMENTS

H. Drywell - Suppression Chamber Differential Pressure

1. The drywell - suppression chamber differential pressure shall be demonstrated to be within limits by verifying the differential pressure at least once per 12 hours.
2. At least one drywell - suppression chamber differential pressure instrumentation CHANNEL, and at least one drywell pressure and one suppression chamber pressure instrumentation CHANNEL shall be demonstrated OPERABLE by performance of a:
   a. CHANNEL CHECK at least once per 24 hours,
   b. CHANNEL CALIBRATION at least once every 31 days.

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a Except for up to 4 hours for required surveillance which reduces the differential pressure.

3.7 - LIMITING CONDITIONS FOR OPERATION

I. DELETED

4.7 - SURVEILLANCE REQUIREMENTS

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3.7 - LIMITING CONDITIONS FOR OPERATION

within 1 hour or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

2. In OPERATIONAL MODE(s) 1 or 2 with the suppression pool average water temperature >95°F, except as permitted above, restore the average temperature to ≤95°F within 24 hours or reduce THERMAL POWER to ≤1% RATED THERMAL POWER within the next 12 hours.

3. With the suppression pool average water temperature >105°F during testing which adds heat to the suppression pool, except as permitted above, stop all testing which adds heat to the suppression pool and restore the average temperature to ≤95°F within 24 hours or reduce THERMAL POWER to ≤1% RATED THERMAL POWER within the next 12 hours.

4. With the suppression pool average water temperature >110°F, immediately place the reactor mode switch in the Shutdown position and operate at least one low pressure coolant injection loop in the suppression pool cooling mode.

5. With the suppression pool average water temperature >120°F, depressurize the reactor pressure vessel to <150 psig (reactor steam dome pressure) within 12 hours.

4.7 - SURVEILLANCE REQUIREMENTS

3. Deleted.

4. Deleted.

5. At least once per 18 months by conducting a drywell to suppression chamber bypass leak test at an initial differential pressure of 1.0 psid and verifying that the measured leakage is within the specified limit. If any drywell to suppression chamber bypass leak test fails to meet the specified limit, the test schedule for subsequent tests shall be reviewed and approved by the Commission. If two consecutive tests fail to meet the specified limit, a test shall be performed at least every 9 months until two consecutive tests meet the specified limit, at which time the 18 month test schedule may be resumed.

3.7 - LIMITING CONDITIONS FOR OPERATION

N. SECONDARY CONTAINMENT INTEGRITY

SECONDARY CONTAINMENT INTEGRITY shall be maintained.

APPLICABILITY:

OPERATIONAL MODE(s) 1, 2, 3 and *.

ACTION:

1. Without SECONDARY CONTAINMENT INTEGRITY in OPERATIONAL MODES(s) 1, 2 or 3, restore SECONDARY CONTAINMENT INTEGRITY within 4 hours or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

2. Without SECONDARY CONTAINMENT INTEGRITY in OPERATIONAL MODE *, suspend handling of irradiated fuel in the secondary containment, CORE ALTERATION(s), and operations with a potential for draining the reactor vessel. The provisions of Specification 3.0.C are not applicable.

4.7 - SURVEILLANCE REQUIREMENTS

N. SECONDARY CONTAINMENT INTEGRITY

SECONDARY CONTAINMENT INTEGRITY shall be demonstrated by:

1. Verifying at least once per 24 hours that the pressure within the secondary containment is ≥0.25 inches of vacuum water gauge.

2. Verifying at least once per 31 days that:

   a. At least one door in each secondary containment air lock is closed.

   b. All secondary containment penetrations[a] not capable of being closed by OPERABLE secondary containment automatic isolation dampers and required to be closed during accident conditions are closed.

3. At least once per 18 months by operating one standby gas treatment subsystem at a flow rate ≤4000 cfm for one hour and maintaining ≥0.25 inches of vacuum water gauge in the secondary containment.

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* When handling irradiated fuel in the secondary containment, during CORE ALTERATION(s), and operations with a potential for draining the reactor vessel.

a Valves and blind flanges in high-radiation areas may be verified by use of administrative controls. Normally locked or sealed-closed penetrations may be opened intermittently under administrative controls.

3.7 - LIMITING CONDITIONS FOR OPERATION

P. Standby Gas Treatment System

Two independent standby gas treatment subsystems shall be OPERABLE.

APPLICABILITY:

OPERATIONAL MODE(s) 1, 2, 3 and *.

ACTION:

1. With one standby gas treatment subsystem inoperable, restore the inoperable subsystem to OPERABLE status within 7 days, or:
   a. In OPERATIONAL MODE(s) 1,2 or 3, be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.
   b. In OPERATIONAL MODE *, suspend handling of irradiated fuel in the secondary containment, CORE ALTERATION(s), and operations with a potential for draining the reactor vessel. The provisions of Specification 3.0.C are not applicable.
2. Deleted.

4.7 - SURVEILLANCE REQUIREMENTS

P. Standby Gas Treatment System

Each standby gas treatment subsystem shall be demonstrated OPERABLE:

1. At least once per 31 days by initiating, from the control room, flow through the HEPA filters and charcoal adsorbers and verifying that the subsystem operates for at least 10 hours with the heaters operating.
2. 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 subsystem by:
   a. Verifying that the subsystem satisfies the in-place penetration and bypass leakage testing acceptance criteria of <1% 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 4000 cfm ±10%.
   b. 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 ASTM-D-3803-89, for a methyl iodide penetration of <10%, when tested at 30°C and 70% relative humidity; and

* When handling irradiated fuel in the secondary containment, during CORE ALTERATION(s), and operations with a potential for draining the reactor vessel.

3.7 - LIMITING CONDITIONS FOR OPERATION

3. Deleted.

4.7 - SURVEILLANCE REQUIREMENTS

c. Verifying a subsystem flow rate of 4000 cfm ±10% during system operation when tested in accordance with ANSI N510-1980.

3. After every 1440 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 ASTM-D-3803-89, for a methyl iodide penetration of <10%, when tested at 30°C and 70% relative humidity.

4. At least once per 18 months by:

   a. Verifying that the pressure drop across the combined HEPA filters and charcoal adsorber banks is <6 inches water gauge while operating the filter train at a flow rate of 4000 cfm ±10%.

   b. Verifying that the filter train starts and isolation dampers open on each of the following test signals:

      1) Manual initiation from the control room, and

      2) Simulated automatic initiation signal.

   c. Verifying that the heaters dissipate 30 ±3 kw when tested in accordance with ANSI N510-1989. This reading shall include the appropriate correction for variations from 480 volts at the bus.

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which temporarily increases the suppression chamber pressure and reduces the differential pressure. Only one direct suppression chamber to drywell differential pressure instrumentation CHANNEL is provided. However, any pair of the redundant drywell and suppression chamber pressure instrumentation CHANNEL(s) are sufficient to determine the differential pressure.

3/4.7.I DELETED

3/4.7.J Primary Containment Oxygen Concentration

All nuclear reactors must be designed to withstand events that generate hydrogen either due to the zirconium metal-water reaction in the core or due to radiolysis. The primary method to control hydrogen is to inert the primary containment. With the primary containment inerted, that is, oxygen concentration less than 4.0 volume percent, a combustible mixture cannot be present in the primary containment for any hydrogen concentration. The Design Basis Accident (DBA) loss-of-coolant accident (LOCA) analysis assumes that the primary containment is inerted when the DBA occurs. Thus, the hydrogen assumed to be released to the primary containment as a result of a metal-water reaction in the reactor core will not produce combustible gas mixtures in the primary containment.

The primary containment oxygen concentration must be within the specified limit when primary containment is inerted, except as allowed by the relaxations during startup and shutdown. The primary containment must be inert in OPERATIONAL MODE 1, since this is the condition with the highest probability of an event that could produce hydrogen. Inerting the primary containment is an operational problem because it prevents containment access without an appropriate breathing apparatus. Therefore, the primary containment is inerted as late as possible in the plant startup and de-inerted as soon as possible in the plant shutdown. As long as reactor power is below 15% of RATED THERMAL POWER, the potential for an event that generates significant hydrogen is low and the primary containment does not need to be inert. Furthermore, the probability of an event that generates hydrogen occurring within the first 24 hours of a reactor startup or within the last 24 hours before a shutdown is low enough that these windows, when the primary containment is not inerted, are also justified. The 24 hour time frame is a reasonable amount of time to allow plant personnel to perform inerting or de-inerting.

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3/4.7.N SECONDARY CONTAINMENT INTEGRITY

The function of the secondary containment is to isolate and contain fission products that escape from primary containment following a Design Basis Accident (DBA), to confine the postulated release of radioactive material within the requirements of 10CFR Part 100, and to isolate and contain fission products that are released during certain operations that take place inside primary containment, when primary containment is not required to be OPERABLE, or that take place outside of primary containment. The reactor building and associated structures provide secondary containment during normal operation when the drywell is sealed and in service. At other times the drywell may be open and, when required, secondary containment integrity is specified. There are two principal accidents for which credit is taken for secondary containment OPERABILITY. These are a LOCA and fuel-handling accident inside secondary containment. The secondary containment performs no active function in response to each of these limiting events; however, its leak tightness is required to limit offsite radiation doses to below those required by 10CFR Part 100. Maintaining secondary containment OPERABLE ensures that the release of radioactive materials from the primary containment is restricted to those leakage paths and associated leakage rates assumed in the accident analysis and that fission products entrapped within the secondary containment structure will be treated prior to discharge to the environment. Establishing and maintaining a vacuum in the reactor building with the standby gas treatment system during testing, along with the surveillance of the doors, hatches, dampers and valves, is adequate to ensure that there are no violations of the integrity of the secondary containment. This surveillance is normally conducted during periods of calm winds (<5 mph), but may be conducted under higher wind conditions with appropriate application of correction factors.

Valves and blind flanges located in high radiation areas may be verified by use of administrative controls. Allowing verification by administrative controls is considered acceptable, since access to these areas is typically restricted during MODES 1, 2, and 3 for ALARA reasons. Therefore, the probability of misalignment of these SCIVs, once they have been verified to be in the proper position, is low. Normally locked or sealed closed penetrations may be opened intermittently under administrative controls. These administrative controls consist of stationing a dedicated operator, who is in continuous communication with the control room, at the controls of the penetration. In this way, the penetration can be rapidly isolated when a valid secondary containment isolation signal is indicated.

3/4.7.O Secondary Containment Automatic Isolation Dampers

The function of the secondary containment ventilation system automatic isolation dampers, in combination with other accident-mitigation systems, is to limit fission-product release during and following postulated Design Basis Accidents (DBA) such that offsite radiation exposures are maintained within the requirements of 10CFR Part 100. Secondary containment isolation ensures that fission products that escape from primary containment following a DBA, or which are released during certain operations when primary containment is not required, or take place outside primary containment, are maintained within applicable limits. The OPERABILITY requirements for the secondary containment ventilation system isolation dampers help ensure that adequate secondary

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containment leak tightness is maintained during and after an accident by minimizing potential paths to the environment.

3/4.7.P Standby Gas Treatment System

The standby gas treatment system (SBGT) is required to ensure that radioactive materials that leak from the primary containment into the secondary containment following a Design Basis Accident (DBA) are filtered and adsorbed prior to exhausting to the environment. This system reduces the potential releases of radioactive material, principally iodine, to within values specified in 10CFR Part 100.

The standby gas treatment system is designed to filter and exhaust the reactor building atmosphere to the main chimney during secondary containment isolation conditions, with a minimum release of radioactive materials from the reactor building to the environment. One standby gas treatment fan is designed to automatically start upon secondary containment isolation and to maintain the reactor building pressure to approximately a negative ¼ inch water gauge pressure; all leakage should be in-leakage. Should the fan fail to start, the redundant alternate fan and filter subsystem is designed to start automatically.

The OPERABILITY of the standby gas treatment system reduces the potential release of radioactive material, principally iodine, following a design basis accident. The reduction in containment iodine inventory reduces the resulting site boundary radiation doses associated with containment leakage. The operation of this system and resultant iodine removal capacity are consistent with the assumptions used in the LOCA analyses. Periodic operation of the system with the heaters is sufficient to reduce the buildup of moisture on the adsorbers and HEPA filters.

Since the standby gas treatment subsystems are shared by both units, one subsystem is powered by the unit diesel generator power source of each unit. The emergency power supply OPERABILITY requirements for the standby gas treatment system are addressed within Specification 3.9.A, Actions. For example, if conducting the alternate offsite power source cross-tie surveillance were to require the inoperability of both unit diesel generator power sources, neither of the standby gas treatment subsystems would have an OPERABLE diesel generator power source and the appropriate ACTION would have to be entered.

3.8 - LIMITING CONDITIONS FOR OPERATION

A. Containment Cooling Service Water System

At least the following independent containment cooling service water (CCSW) subsystems, with each subsystem comprised of:

1. Two OPERABLE CCSW pumps, and

2. An OPERABLE flow path capable of taking suction from the ultimate heat sink and transferring the water:

   a. Through one LPCI heat exchanger, and separately,

   b. To the associated safety related equipment,

shall be OPERABLE:

1. In OPERATIONAL MODE(s) 1, 2 and 3, two subsystems.

2. In OPERATIONAL MODE *, the subsystem(s) associated with subsystems/loops and components required OPERABLE by Specification 3.8.D.

APPLICABILITY:

OPERATIONAL MODE(s) 1, 2, 3 and *.

4.8 - SURVEILLANCE REQUIREMENTS

A. Containment Cooling Service Water System

Each of the required CCSW subsystems shall be demonstrated OPERABLE at least once per 31 days by verifying that each valve, manual or power operated, in the flow path that is not locked, sealed or otherwise secured in position, is in its correct position.

* When handling irradiated fuel in the secondary containment, during CORE ALTERATION(s), and operations with a potential for draining the reactor vessel.

3.8 - LIMITING CONDITIONS FOR OPERATION

2. In OPERATIONAL MODE * with the CCSW subsystem which is associated with the safety related equipment required OPERABLE by Specification 3.8.D inoperable, declare the associated safety related equipment inoperable and take the ACTION required by Specification 3.8.D.

4.8 - SURVEILLANCE REQUIREMENTS

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* When handling irradiated fuel in the secondary containment, during CORE ALTERATION(s), and operations with a potential for draining the reactor vessel.

3.8 - LIMITING CONDITIONS FOR OPERATION

D. Control Room Emergency Ventilation System

The control room emergency ventilation system shall be OPERABLE, with the system comprised of an OPERABLE control room emergency filtration system and an OPERABLE refrigeration control unit (RCU).

APPLICABILITY:

OPERATIONAL MODE(s) 1, 2, 3, and *.

ACTION:

1. In OPERATIONAL MODE(s) 1, 2 or 3:

    a. With the control room emergency filtration system inoperable, restore the inoperable system to OPERABLE status within 7 days or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

    b. With the refrigeration control unit (RCU) inoperable, restore the inoperable RCU to OPERABLE status within 30 days or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

4.8 - SURVEILLANCE REQUIREMENTS

D. Control Room Emergency Ventilation System

The control room emergency ventilation system shall be demonstrated OPERABLE:

1. At least once per 18 months by verifying that the RCU has the capability to remove the required heat load.

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

3. 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:

    a. Verifying that the system satisfies the in-place penetration and bypass leakage testing acceptance criteria of $<0.05\%$ 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 scfm $\pm 10\%$.

* When handling irradiated fuel in the secondary containment, during CORE ALTERATION(s), and operations with a potential for draining the reactor vessel.

3.8 - LIMITING CONDITIONS FOR OPERATION

2. In OPERATIONAL MODE *, with the control room emergency filtration system or the RCU inoperable, immediately suspend CORE ALTERATION(s), handling of irradiated fuel in the secondary containment and operations with a potential for draining the reactor vessel.

3. The provisions of Specification 3.0.C are not applicable in OPERATIONAL MODE *.

4.8 - SURVEILLANCE REQUIREMENTS

b. 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 ASTM-D-3803-89, for a methyl iodide penetration of <0.50%, when tested at 30°C and 70% relative humidity; and

c. Verifying a system flow rate of 2000 scfm ±10% during system operation when tested in accordance with ANSI N510-1980.

4. After every 1440 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 ASTM-D-3803-89, for a methyl iodide penetration of <0.50%, when tested at 30°C and 70% relative humidity.

5. At least once per 18 months by:

   a. Verifying that the pressure drop across the combined HEPA filters and charcoal adsorber banks is <6 inches water gauge while operating the filter train at a flow rate of 2000 scfm ±10%.

   b. Verifying that the filter train starts and isolation dampers close on manual initiation from the control room.

3.8 - LIMITING CONDITIONS FOR OPERATION

4.8 - SURVEILLANCE REQUIREMENTS

c. Verifying that during the pressurization mode of operation, control room positive pressure is maintained at ≥1/8 inch water gauge relative to adjacent areas during system operation at a flow rate ≤2000 scfm.

d. Verifying that the heaters dissipate 12 ±1.2 kw when tested in accordance with ANSI N510-1980. This reading shall include the appropriate correction for variations from 480 volts at the bus.

6. After each complete or partial replacement of an HEPA filter bank by verifying that the HEPA filter bank satisfies the in-place penetration and leakage testing acceptance criteria of <0.05% in accordance with ANSI N510-1980 while operating the system at a flow rate of 2000 scfm ±10%.

7. After each complete or partial replacement of an charcoal adsorber bank by verifying that the charcoal adsorber bank satisfies the in-place penetration and leakage testing acceptance criteria of <0.05% in accordance with ANSI N510-1980 for a halogenated hydrocarbon refrigerant test gas while operating the system at flow rate of 2000 scfm ±10%.

3.8 - LIMITING CONDITIONS FOR OPERATION

4.8 - SURVEILLANCE REQUIREMENTS

"Accept" line, testing of that type of snubber may be terminated. If the point plotted falls above the "Accept" line, testing must continue until the point falls on or below the "Accept" line or all the snubbers of that type have been tested.

The representative sample selected for the functional test sample plans shall be randomly selected from the snubbers of each type and reviewed before beginning the testing. The review shall ensure as far as practical that they are representative of the various configurations, operating environments, range of size, and capacity of snubbers of each type.

Snubbers placed in the same location as snubbers which failed the previous functional test shall be retested at the time of the next functional test but shall not be included in the sample plan, and failure of this functional test shall not be the sole cause for increasing the sample size under the sample plan. If during testing, additional sampling is required due to failure of only one type of snubber, the functional testing results shall be reviewed at the time to determine if additional samples should be limited to the type of snubber which has failed the functional testing.

6. Functional Test Acceptance Criteria

The snubber functional test shall verify that:

a. Activation (restraining action) is achieved within the specified range in both tension and compression;

3.8 - LIMITING CONDITIONS FOR OPERATION

4.8 - SURVEILLANCE REQUIREMENTS

the snubber is required to be OPERABLE. The parts replacements shall be documented and the documentation shall be retained.

3.8 - LIMITING CONDITIONS FOR OPERATION

I. Main Condenser Offgas Activity

The release rate of the sum of the activities of the noble gases measured prior to the offgas holdup line shall be limited to ≤100 μCi/sec/MWt, after 30 minutes decay.

APPLICABILITY:

OPERATIONAL MODE(s) 1, 2[a] and 3[a].

ACTION:

With the release rate of the sum of the activities of the noble gases at the main condenser air ejector effluent (as measured prior to the offgas holdup line) >100 μCi/sec/MWt, after 30 minutes decay, restore the release rate to within its limit within 72 hours or be in at least STARTUP with the main steam isolation valves closed within the next 8 hours.

4.8 - SURVEILLANCE REQUIREMENTS

I. Main Condenser Offgas Activity

1. The release rate of noble gases from the main condenser air ejector shall be continuously monitored in accordance with the ODCM.

2. The release rate of the sum of the activities from noble gases from the main condenser air ejector shall be determined to be within the limits of Specification 3.8.I at the following frequencies[b] by performing an isotopic analysis of a representative sample of gases taken at the recombiner outlet, or the air ejector outlet, if the recombiner is bypassed:

   a. At least once per 31 days; and

   b. Within 4 hours following the determination of an increase of >50%.

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a When the main condenser air ejector is in operation.

b The provisions of Specification 4.0.D are not applicable.

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3/4.8.A Containment Cooling Service Water System

The containment cooling service water system, with the ultimate heat sink, provides sufficient cooling capacity for continued operation of the containment cooling system and of other safety-related equipment (e.g., CCSW keep-fill, the control room emergency ventilation system refrigeration units), during normal and accident conditions. The redundant cooling capacity of the system, assuming a single failure, is consistent with the assumptions used in the safety analysis to keep the accident conditions within acceptable limits. Since only two of the four pumps is required to provide the necessary cooling capacity, a thirty day repair period is allowed for one pump out of service. OPERABILITY of this system is also dependent upon special measures for protection from flooding in the condenser pit area.

3/4.8.B Diesel Generator Cooling Water System

The diesel generator cooling water system, with the ultimate heat sink, provides sufficient cooling capacity for continued operation of the diesel generators during normal and accident conditions. The cooling capacity of the system is consistent with the assumptions used in the safety analysis to keep the accident conditions within acceptable limits. OPERABILITY of this system is also dependent upon special measures for protection from flooding in the condenser pit area.

3/4.8.C Ultimate Heat Sink

The canals provide an ultimate heat sink with sufficient cooling capacity to either provide normal cooldown of the units, or to mitigate the effects of accident conditions within acceptable limits for one unit while conducting a normal cooldown on the other unit.

3/4.8.D Control Room Emergency Ventilation System

The control room emergency filtration system maintains habitable conditions for operations personnel during and following all design basis accident conditions. This system, in conjunction with control room design, is based on limiting the radiation exposure to personnel occupying the room to five rem or less whole body, or its equivalent.

The frequency of tests and sample analysis is necessary to show that the HEPA filters and charcoal adsorbers can perform as evaluated. The control room emergency filtration system in-place testing procedures are established utilizing applicable sections of ANSI N510-1980 standard. Operation of the system with the heaters OPERABLE for ten hours a month is sufficient to reduce the buildup of moisture on the adsorbers and HEPA filters. The charcoal adsorber efficiency test procedures allow for the removal of one representative sample cartridge and testing in accordance with the guidelines of ASTM-D-3803-89. The sample is at least two inches in diameter and has a length equivalent to the thickness of the bed. If the iodine removal efficiency

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test results are unacceptable, all adsorbent in the system is replaced. HEPA filter particulate removal efficiency is verified to be at least 99% by in-place testing with a DOP testing medium.

The control room refrigeration control unit (RCU) provides conditioned air for personnel comfort, safety and equipment reliability. The testing of the control room RCU system verifies that the heat-removal capability of the system is sufficient to remove sufficient heat load from the control room such that the control room air temperature is ≤ 95 °F. The test frequency is appropriate since significant degradation of the control room RCU system is not expected over this time period.

3/4.8.E Flood Protection

Flood protection measures are provided to protect the systems and equipment necessary for safe shutdown during high water conditions. The equipment necessary to implement the appropriate measures, as detailed in plant procedures, is required to be available, but not necessarily onsite, to implement the procedures in a timely manner. The selected water levels are based on providing timely protection from the design basis flood of the river.

3/4.8.F Snubbers

Mechanical snubbers are provided to ensure that the structural integrity of the reactor coolant system and all other safety-related systems is maintained during and following a seismic event or other event initiating dynamic loads. Snubbers are classified and grouped by design, manufacturer and accessibility. A list of individual snubbers with information of snubber location, classification or group, and system affected is maintained at the plant. The accessibility of each snubber is determined and documented for each snubber. The determination is based upon the existing radiation levels and the expected time to perform a visual inspection in each snubber location as well as other factors associated with accessibility during plant operation (e.g., temperature, atmosphere, location, etc.), and the recommendations of Regulatory Guides 8.8 and 8.10.

The visual inspection frequency is based upon maintaining a constant level of snubber protection to the systems. Therefore, the required inspection interval varies with the number of unacceptable snubbers found during the previous inspection, the total population or category size for each snubber type, and the previous inspection interval. A snubber is considered unacceptable if it fails to satisfy the acceptance criteria of the visual inspection. Snubbers may be categorized, based upon their accessibility during power operation, as accessible or inaccessible. These categories may be examined separately or jointly as determined and documented prior to the inspections. The categorization is used as the basis for determining the next inspection interval for that category.

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If a review and evaluation can not justify continued operation with an unacceptable snubber, the snubber is declared inoperable and the applicable action taken. To determine the next surveillance interval, the unacceptable snubber may be reclassified as acceptable if it can be demonstrated that the snubber is OPERABLE in its as-found condition by the performance of a functional test. The next visual inspection interval may be twice, the same, or reduced by as much as two-thirds of the previous inspection interval, depending on the number of unacceptable snubbers found in proportion to the size of the population or category for each type of snubber included in the previous inspection. The inspection interval may be as long as 48 months and the provisions of Specification 4.0.B may be applied.

When a snubber is found to be inoperable, an engineering evaluation is performed, in addition to the determination of the snubber mode of failure, in order to determine if any safety-related component or system has been adversely affected by the inoperability of the snubber. The engineering evaluation shall determine whether or not the snubber mode of failure has imparted a significant effect or degradation on the supported component or system.

To provide additional assurance of snubber functional reliability , a representative sample of the installed snubbers will be functionally tested at 18 month intervals. This sample is identified using one of three methods:

1. Functionally test 10% of a type of snubber with an additional 10% tested for each functional testing failure, or

2. Functionally test a sample size and determine sample acceptance or rejection using Figure 4.8.F-1, or

3. Functionally test a representative sample size and determine sample acceptance or rejection using the stated equation.

Figure 4.8.F-1 was developed using "Wald's Sequential Probability Ratio Plan" as described in "Quality Control and Industrial Statistics" by Acheson J. Duncan.

Permanent or other exemptions from the surveillance program for individual snubbers may be granted by the NRC if a justifiable basis for exemption is presented and, if applicable, snubber life destructive testing was performed to qualify the snubber for the applicable design conditions at either the completion of their fabrication or at a subsequent date. Snubbers so exempted are listed in the list of individual snubbers indicating the extent of the exemptions.

The service life of a snubber is established via manufacturer input and information through consideration of the snubber service conditions and associated installation and maintenance records (newly installed snubbers, seal replace, spring replaced, in high radiation area, in high temperature area, etc.). The requirement to monitor the snubber service life is included to ensure that the snubbers periodically undergo a performance evaluation in view of their age and operating conditions. These records provide statistical bases for future consideration of snubber service life.

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3/4.8.G Sealed Source Contamination

The limitations on removable contamination for sources requiring leak testing, including alpha emitters, is based on 10 CFR 70.39(c) limits for plutonium. This limitation will ensure that leakage from byproduct, source, and special nuclear material sources will not exceed allowable intake values. Sealed sources, including startup sources and fission detectors, are classified into three groups according to their use, with surveillance requirements commensurate with the probability of damage to a source in that group. Those sources which are frequently handled are required to be tested more often than those which are not. Sealed sources which are continuously enclosed within a shielded mechanism, i.e., sealed sources within radiation monitoring or boron measuring devices, are considered to be stored and need not be tested unless they are removed from the shielded mechanism.

3/4.8.H Explosive Gas Mixture

This specification is provided to ensure that the concentration of potentially explosive gas mixtures contained in the offgas holdup system is maintained below the flammability limits of hydrogen and oxygen. Maintaining the concentration of hydrogen and oxygen below their flammability limits provides assurance that the releases of radioactive materials will be controlled in conformance with the requirements of General Design Criterion 60 of Appendix A to 10CFR Part 50.

3/4.8.I Main Condenser Offgas Activity

Restricting the gross radioactivity rate of noble gases from the main condenser provides reasonable assurance that the total body exposure to an individual at the exclusion area boundary will not exceed a small fraction of the limits of 10CFR Part 100 in the event this effluent is inadvertently discharged directly to the environment without treatment. This specification implements the requirements of General Design Criteria 60 and 64 of Appendix A to 10CFR Part 50. The release rates are determined at a more expeditious frequency following the determination of an increase of greater than 50%, as indicated by the air ejector noble gas monitor, after factoring out increases due to changes in THERMAL POWER level and off-gas flow in the nominal steady-state fission gas release from the primary coolant.

3/4.8.J Liquid Holdup Tanks

Restricting the quantity of radioactive material contained in the specified tanks provides assurance that in the event of an uncontrolled release of the tanks' contents, the resulting concentrations would be less than the limits of 10CFR Part 20, Appendix B, Table II, Column 2, at the nearest potable water supply and the nearest surface water supply in an unrestricted area. Recirculation of the tank contents for the purpose of reducing the radioactive content is not considered to be an addition of radioactive material to the tank.

3.9 - LIMITING CONDITIONS FOR OPERATION

A. A.C. Sources - Operating

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

1. Two physically independent circuits between the offsite transmission network and the onsite Class 1E distribution system, and
2. Two separate and independent diesel generators, each with:
   a. A separate fuel oil day tank containing ≥205 gallons of available fuel,
   b. A separate bulk fuel storage system containing ≥10,000 gallons of available fuel, and
   c. A separate fuel oil transfer pump.

APPLICABILITY:

OPERATIONAL MODE(s) 1, 2, and 3.

ACTION:

1. With one of the above required offsite circuit power sources inoperable:
   a. Demonstrate the OPERABILITY of the remaining offsite circuit by performing Surveillance Requirement 4.9.A.1.a within 1 hour and at least once per 8 hours thereafter.

4.9 - SURVEILLANCE REQUIREMENTS

A. A.C Sources - Operating

1. Each of the required independent circuits between the offsite transmission network and the onsite Class 1E distribution system shall be determined OPERABLE:
   a. At least once per 7 days by verifying correct breaker alignments and indicated power availability, and
   b. At least once per 18 months by manually transferring the power supply from the normal circuit to the alternate circuit.
2. Each of the required diesel generators shall be demonstrated OPERABLE[a] at least once per 31 days by:
   a. Verifying the fuel levels in both the fuel oil day tank and the bulk fuel storage tank.
   b. Verifying the fuel transfer pump starts and transfers fuel from the bulk fuel storage system to the fuel oil day tank.

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a All diesel generator starts may be preceded by an engine prelube period. All diesel generator starts that require loading may be preceded by an engine prelube period and followed by a warmup period prior to loading. Diesel generator loadings may include gradual loading as recommended by the manufacturer/vendor.

3.9 - LIMITING CONDITIONS FOR OPERATION

b. Restore the inoperable offsite circuit to OPERABLE status within 7 days or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

2. With one of the above required diesel generator power sources inoperable:

   a. Demonstrate the OPERABILITY of the offsite circuit power sources by performing Surveillance Requirement 4.9.A.1.a within 1 hour and at least once per 8 hours thereafter.

   b. If the diesel generator is inoperable due to any cause other than an inoperable support system, an independently testable component, or preplanned maintenance or testing, demonstrate the OPERABILITY of the remaining OPERABLE diesel generator by performing Surveillance Requirement 4.9.A.2.c[b] within 24 hours unless the absence of any potential common mode failure for the remaining diesel generator is demonstrated (if it has not been successfully tested within the past 24 hours) and within the subsequent 72 hours, and

4.9 - SURVEILLANCE REQUIREMENTS

c. Verifying[c] the diesel starts and accelerates to synchronous speed with generator voltage and frequency at 4160 ±420 volts and 60 ±1.2 Hz, respectively.

d. Verifying the diesel generator is synchronized, loaded to between 2470 and 2600 kW[d] in accordance with the manufacturer's/vendor's recommendations, and operates with this load for ≥60 minutes.

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

f. Verifying the pressure in required starting air receiver tanks to be ≥220 psig.

3. Each of the required diesel generators shall be demonstrated OPERABLE at least once per 31 days and after each operation of the diesel where the period of operation was ≥1 hour by removing any accumulated water from the day tank.

4. Each of the required diesel generators shall be demonstrated OPERABLE at least once per 92 days by checking for and removing accumulated water from the fuel oil bulk storage tanks.

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b Contrary to the provisions of Specification 3.0.B, this test is required to be completed regardless of when the inoperable diesel generator is restored to OPERABILITY for failures that are potentially generic to the remaining diesel generator and for which appropriate alternative testing cannot be designed.

c Surveillance Requirement 4.9.A.7 may be substituted for Surveillance Requirement 4.9.A.2.c.

d Momentary transients outside of the load range do not invalidate this test. Diesel generator loadings may include gradual loading as recommended by the manufacturer/vendor. This surveillance shall be conducted on only one diesel generator at a time.

3.9 - LIMITING CONDITIONS FOR OPERATION

c. Restore the diesel generator to OPERABLE status within 7 days or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

3. With one of the above offsite circuit power sources and one of the above required diesel generator power sources inoperable:

   a. Demonstrate the OPERABILITY of the remaining offsite circuit power source by performing Surveillance Requirement 4.9.A.1.a within 1 hour and at least once per 8 hours thereafter.

   b. If the diesel generator is inoperable due to any cause other than preplanned maintenance or testing, demonstrate the OPERABILITY[e] of the remaining OPERABLE diesel generator by performing Surveillance Requirement 4.9.A.2.c[b] within 8 hours unless the absence of any potential common mode failure for the remaining diesel generator is demonstrated (if it has not been successfully tested within the past 24 hours) and within the subsequent 72 hours for each OPERABLE diesel generator.

4.9 - SURVEILLANCE REQUIREMENTS

5. Each of the required diesel generators shall be demonstrated OPERABLE by:

   a. Sampling new fuel oil prior to addition to the storage tanks in accordance with applicable ASTM standards, and

   b. Verifying prior to addition to the storage tanks that the sample meets the applicable ASTM standards for API gravity, water and sediment, and the visual test for free water and particulate contamination, and

   c. Verifying within 31 days of obtaining the sample that the kinematic viscosity is within applicable ASTM limits.

6. Each of the required diesel generators shall be demonstrated OPERABLE by:

   a. Sampling and analyzing the bulk fuel storage tanks at least once per 31 days in accordance with applicable ASTM standards, and

   b. Verifying that the sample meets the applicable ASTM standards for water and sediment, kinematic viscosity, and ASTM particulate contaminant is <10 mg/liter.

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e A successful test of OPERABILITY per Surveillance Requirement 4.9.A.2.c under this ACTION statement satisfies the diesel generator test requirements of ACTION(s) 1 or 2 above.

b Contrary to the provisions of Specification 3.0.B, this test is required to be completed regardless of when the inoperable diesel generator is restored to OPERABILITY for failures that are potentially generic to the remaining diesel generator and for which appropriate alternative testing cannot be designed.

3.9 - LIMITING CONDITIONS FOR OPERATION

c. Restore at least one of the inoperable A.C. power sources to OPERABLE status within 12 hours or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours, and

d. Restore both offsite circuits and both diesel generators to OPERABLE status within 7 days from the time of the initial loss or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

4. With one of the above required diesel generator power sources inoperable, in addition to ACTION 2 or 3, as applicable:

   a. Verify within 2 hours that at least one of the required two systems, subsystems, trains, components and devices in two train systems is OPERABLE including its emergency power supply.

   b. Otherwise, take the applicable ACTIONs for both systems, subsystems, trains, components or devices inoperable, or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

4.9 - SURVEILLANCE REQUIREMENTS

7. Each of the required diesel generators shall be demonstrated OPERABLE[a] at least once per 184 days by verifying[c] the diesel starts and accelerates to synchronous speed in ≤13 seconds. The generator voltage and frequency shall be verified to reach 4160 ±420 volts and 60 ±1.2 Hz, respectively, in ≤13 seconds after the start signal.

8. Each of the required diesel generators shall be demonstrated OPERABLE[a] at least once per 18 months by:

   a. Deleted.

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a All diesel generator starts may be preceded by an engine prelube period. All diesel generator starts that require loading may be preceded by an engine prelube period and followed by a warmup period prior to loading. Diesel generator loadings may include gradual loading as recommended by the manufacturer/vendor.

c Surveillance Requirement 4.9.A.7 may be substituted for Surveillance Requirement 4.9.A.2.c.

3.9 - LIMITING CONDITIONS FOR OPERATION

5. With two of the above required offsite circuit power sources inoperable:

   a. Restore at least one of the inoperable offsite circuits to OPERABLE status within 24 hours or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours, and

   b. Restore at least two offsite circuits to OPERABLE status within 7 days from the time of initial loss or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

6. With both of the above required diesel generator power sources inoperable:

   a. Demonstrate the OPERABILITY of the offsite circuit power sources by performing Surveillance Requirement 4.9.A.1.a within 1 hour and at least once per 8 hours thereafter.

4.9 - SURVEILLANCE REQUIREMENTS

b. Verifying the diesel generator capability to reject its largest single emergency load (≥642 kW) while maintaining speed ≤1001 rpm and voltage at 4160 ± 420 volts.

c. Verifying the diesel generator capability to reject a load between 2470 and 2600 kw[d], without tripping on overspeed. The generator voltage shall not exceed 5000 volts during or following the load rejection.

d. Simulating a loss of offsite power by itself, and:

   1) Verifying de-energization of the emergency buses, and load shedding from the emergency buses.

   2) Verifying the diesel starts on the auto-start signal, energizes the emergency buses with permanently connected loads in ≤13 seconds, energizes the auto-connected shutdown loads, and operates with this load for ≥5 minutes. After energization, the steady-state voltage and frequency of the emergency busses shall be maintained at 4160 ±420 volts and 60 ±1.2 Hz, respectively, during this test.

d Momentary transients outside of the load range do not invalidate this test. Diesel generator loadings may include gradual loading as recommended by the manufacturer/vendor. This surveillance shall be conducted on only one diesel generator at a time.

3.9 - LIMITING CONDITIONS FOR OPERATION

4.9 - SURVEILLANCE REQUIREMENTS

g. Verifying that all automatic diesel generator trips, except engine overspeed and generator differential current are automatically bypassed upon an emergency actuation signal.

h. Verifying the diesel generator operates for ≥24 hours. During the first 2 hours of this test, the diesel generator shall be loaded to between 2730 and 2860 kW[d] and during the remaining 22 hours of this test, the diesel generator shall be loaded to between 2470 and 2600 kW[d]. The generator voltage and frequency shall be 4160 ±420 volts and 60 ±1.2 Hz, respectively, in ≤13 seconds after the start signal; the steady state generator voltage and frequency shall be maintained within these limits during this test. Within 5 minutes after completing this 24 hour test, perform Surveillance Requirement 4.9.A.2.c[f].

i. Verifying that the auto-connected loads to each diesel generator do not exceed the 2000 hour rating of 2860 kW.

d Momentary transients outside of the load range do not invalidate this test. Diesel generator loadings may include gradual loading as recommended by the manufacturer/vendor. This surveillance shall be conducted on only one diesel generator at a time.

f If Surveillance Requirement 4.9.A.2.c is not satisfactorily completed, it is not necessary to repeat the preceding 24 hour test. Instead, the diesel generator may be operated at approximately full load for 2 hours or until the operating temperature has stabilized.

3.9 - LIMITING CONDITIONS FOR OPERATION

4.9 - SURVEILLANCE REQUIREMENTS

j. Verifying the diesel generator's capability to:

1) synchronize with the offsite power source while the generator is loaded with its emergency loads upon a simulated restoration of offsite power,

2) transfer its loads to the offsite power source, and

3) be restored to its standby status.

k. Verifying that the automatic load sequence logic is OPERABLE with the interval between each load block within ±10% of its design interval.

9. Each of the required diesel generators shall be demonstrated OPERABLE[a] at least once per 10 years or after any modifications which could affect diesel generator interdependence by starting both diesel generators simultaneously, and verifying that both diesel generators accelerate to ≥900 rpm in ≤13 seconds.

10. Each of the required diesel generators shall be demonstrated OPERABLE at least once per 10 years by draining each fuel oil storage tank, removing the accumulated sediment and cleaning the tank.

a All diesel generator starts may be preceded by an engine prelube period. All diesel generator starts that require loading may be preceded by an engine prelube period and followed by a warmup period prior to loading. Diesel generator loadings may include gradual loading as recommended by the manufacturer/vendor.

TABLE 4.9.A-1

DIESEL GENERATOR TEST SCHEDULE

(NOT USED)

3.9 - LIMITING CONDITIONS FOR OPERATION

C. D.C. Sources - Operating

As a minimum, the following D.C. electrical power sources shall be OPERABLE with the identified parameters within the limits specified in Table 4.9.C-1:

1. Two station 250 volt batteries, each with a full capacity charger.
2. Two station 125 volt batteries, each with a full capacity charger.
3. Two unit 24/48 volt batteries, each with a full capacity charger.

APPLICABILITY:

OPERATIONAL MODE(s) 1, 2, and 3.

ACTION:

1. With one of the above required 24/48 volt or 250 volt station batteries and/or chargers inoperable, restore the inoperable equipment to OPERABLE status within 2 hours[b].

4.9 - SURVEILLANCE REQUIREMENTS

C. D.C. Sources - Operating

Each of the required 24/48 volt, 125 volt and 250 volt batteries and chargers shall be demonstrated OPERABLE[a]:

1. At least once per 7 days by verifying that:
   a. The parameters in Table 4.9.C-1 meet Category A limits, and
   b. There is correct breaker alignment to the battery chargers and total battery terminal voltage is ≥26.0, ≥125.9, or ≥260.4 volts, as applicable, on float charge.
2. At least once per 92 days and within 7 days after a battery discharge with a battery terminal voltage below 21.7, 105 or 210 volts, as applicable, or battery overcharge with battery terminal voltage above 30, 150 or 300 volts, as applicable, by verifying that:
   a. The parameters in Table 4.9.C-1 meet the Category B limits,
   b. There is no visible corrosion at either terminals or connectors, or the connection resistance of these items is ≤150 x10^{-6} ohms or ≤20% above baseline connection resistance, whichever is higher, and

a An alternate 125 volt battery shall adhere to these same Surveillance Requirements to be considered OPERABLE, except the Unit 2 total battery terminal voltage on float charge shall be verified weekly as ≥130.2 volts.

b Each 250 volt battery may be inoperable for a maximum of seven days per operating cycle for maintenance or testing. If it is determined that a 250 volt battery need be replaced as a result of maintenance or testing, a specific battery may be inoperable for an additional seven days per operating cycle.

3.9 - LIMITING CONDITIONS FOR OPERATION

2. With one of the above required 125 volt station batteries and/or chargers inoperable, within 2 hours[c], either restore the inoperable equipment to OPERABLE status, or place an OPERABLE corresponding alternate 125 volt battery (with an OPERABLE full capacity charger) in service.

3. With the provisions of either ACTION 1 or 2 above not met, be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

4. With any Category A parameter(s) outside the limit(s) shown in Table 4.9.C-1, the battery may be considered OPERABLE provided that its associated charger is OPERABLE, and 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.

5. With any Category B parameter(s) outside the limit(s) shown in Table 4.9.C-1, 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 the limit(s) within 7 days.

4.9 - SURVEILLANCE REQUIREMENTS

c. The average electrolyte temperature of all connected cells is above 60°F.

3. At least every 18 months by verifying that:

   a. The cells, cell plates and battery racks show no visual indication of physical damage or abnormal deterioration.

   b. The cell-to-cell and terminal connections are clean, tight, free of corrosion and coated with anti-corrosion material.

   c. The resistance of each cell-to-cell and terminal connection is $\leq 150 \times 10^{-6}$ ohms or ≤20% above baseline connection resistance, whichever is higher.

   d. The battery chargers will supply a load equal to the manufacturer's rating for at least 4 hours.

4. At least every 18 months, by verifying that the battery capacity is adequate to supply and maintain in OPERABLE status all of the actual or simulated emergency loads for design duty cycle when the battery is subjected to a battery service test.

c With Unit 2 and 3 in OPERATIONAL MODE(s) 1, 2 or 3, each 125 volt battery may be inoperable for up to a maximum of seven days per operating cycle for maintenance or testing provided the alternate 125 volt battery is placed into service and is OPERABLE. If it is determined that a 125 volt battery need be replaced as a result of maintenance or testing, a specific battery may be inoperable for an additional seven days provided the alternate 125 volt battery is placed into service and is OPERABLE. With the other Unit in MODE(s) 4 or 5, operations may continue with one of the two 125 volt battery systems inoperable provided the alternate 125 volt battery is placed into service and is OPERABLE.

3.9 - LIMITING CONDITIONS FOR OPERATION

6. With any Category B parameter not within its allowable value(s), immediately declare the battery inoperable.

4.9 - SURVEILLANCE REQUIREMENTS

5. At least once per 60 months, verify that the battery capacity is 80% of the manufacturer's rating when subjected to either a performance discharge test or a modified performance discharge test. The modified performance discharge test satisfies the requirements of both the service test and performance test and therefore, may be performed in lieu of a service test.

6. For any battery that shows signs of degradation or has reached 85% of the service life for the expected application and delivers a capacity of less than 100% of the manufacturer's rated capacity, a performance discharge test or a modified performance test of battery capacity shall be performed at least once every 12 months or the battery shall be replaced or restored to 100% or greater of the manufacturer's rated capacity during the next refuel outage. Degradation is indicated when the battery capacity drops more than 10% from its capacity on the previous performance test, or is below 90% of the manufacturer's rating. If the battery has reached 85% of service life, delivers a capacity of 100% or greater of the manufacturer's rated capacity and has shown no signs of degradation, a performance test or a modified performance test of battery capacity shall be performed at least once every two years.

3.9 - LIMITING CONDITIONS FOR OPERATION

E. Distribution - Operating

The following power distribution systems shall be energized:

1. A.C. power distribution, consisting of:
   a. Both Unit engineered safety features 4160 volt buses:
      1) For Unit 2, Nos. 23-1 and 24-1,
      2) For Unit 3, Nos. 33-1 and 34-1.
   b. Both Unit engineered safety features 480 volt buses:
      1) For Unit 2, Nos. 28 and 29,
      2) For Unit 3, Nos. 38 and 39.
   c. The Unit 120 volt Essential Service Bus and Instrument Bus.
2. 250 volt D.C. power distribution, consisting of:
   a. RB MCC Nos. 2 and 3, and
   b. TB MCC Nos. 2 and 3.
3. For Unit 2, 125 volt D.C. power distribution, consisting of:
   a. TB Main Bus Nos. 2A-1and 3A,
   b. TB Res. Bus Nos. 2B and 2B-1,
   c. Reserve Bus No. 2, and
   d. RB Distribution Panel No. 2.

4.9 - SURVEILLANCE REQUIREMENTS

E. Distribution - Operating

Each of the required power distribution system divisions shall be determined energized at least once per 7 days by verifying correct breaker alignment and voltage on the busses/MCCs/panels.

3.9 - LIMITING CONDITIONS FOR OPERATION

4. For Unit 3, 125 volt D.C. power distribution, consisting of:

   a. TB Main Bus Nos. 2A-1, 3A and 3A-1,

   b. TB Res. Bus Nos. 3B and 3B-1, and

   c. RB Distribution Panel No. 3.

5. 24/48 volt D.C. power distribution, consisting of:

   a. For Unit 2, Bus Nos. 2A and 2B.

   b. For Unit 3, Bus Nos. 3A and 3B.

APPLICABILITY:

OPERATIONAL MODE(s) 1, 2, and 3.

ACTIONS:

1. With one of the above required A.C. distribution systems not energized, re-energize the system within 8 hours or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.
2. With one of the above required D.C. distribution systems not energized, re-energize the system within 2 hours or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

4.9 - SURVEILLANCE REQUIREMENTS

BASES

The initial conditions of design basis transient and accident analyses assume Engineering Safety Features (ESF) systems are OPERABLE. The A.C. and D.C. electrical power sources are designed to provide sufficient capacity, capability, redundancy, and reliability to ensure the availability of necessary power to ESF systems so that the fuel, reactor coolant system and containment design limits are not exceeded.

The A.C. and D.C. sources are designed to permit inspection and testing of all important areas and features, especially those that have a standby function. Periodic component tests are supplemented by extensive functional testing during refueling outages under simulated accident conditions.

3/4.9.A A.C. Sources - Operating

The OPERABILITY of the A.C. electrical power sources is consistent with the initial assumptions of the accident analyses and is based upon meeting the design basis of the plant. This includes maintaining at least one of the onsite or offsite A.C. sources, D.C. power sources and associated distribution systems OPERABLE during accident conditions concurrent with an assumed loss of all offsite power and a worst-case single failure.

There are two sources of electrical energy available, i.e., the offsite transmission system and the onsite diesel generators. Two unit reserve auxiliary transformers are available to supply the Station class 1E distribution system. The reserve auxiliary transformer is sized to carry 100% of the auxiliary load. If this reserve auxiliary transformer (the normal circuit) is lost, auxiliary power from the other unit can be obtained for one division through a 4160 volt bus tie (the alternate circuit). Additionally, two diesel generators are available to handle an accident. The allowable outage time takes into account the capacity and capability of the remaining A.C. sources, reasonable time for repairs, and the low probability of a design basis accident occurring during this period. Surveillance is required to ensure a highly reliable power source and no common cause failure mode for the remaining required offsite A.C. source.

Upon failure of one diesel generator, performance of appropriate surveillance requirements ensures a highly reliable power supply by checking the availability of the required offsite circuits, and the remaining required diesel generator. The initial surveillance is required to be completed regardless of how long the diesel inoperability persists, since the intent is that all diesel generator inoperabilities must be investigated for common cause failures. After the initial surveillance, an additional start test is required approximately mid-way through the allowed outage time to demonstrate continued OPERABILITY of the available onsite power sources. The diesel generator surveillance is limited to the normal start testing, since for cases in which less than a full complement of A.C. sources may be available, paralleling of two of the remaining A.C. sources may compromise the A.C. source independence. Additionally, the action provisions ensure that continued plant operation is not allowed when a complete loss of a required safety function (i.e., certain required components) would occur upon a loss of offsite power. These certain components which are critical to accomplishment of the required safety functions may be identified in advance and administratively controlled and/or evaluated on a case-by-case basis. With suitable

BASES

Surveillance Requirements are also provided for demonstrating the OPERABILITY of the diesel generators. The specified testing is based on the guidance provided in Regulatory Guide 1.9, Revision 3 (7/93), Regulatory Guide 1.108, Revision 1, and Regulatory Guide 1.137, Revision 1, as modified by plant specific analysis, diesel generator manufacturer/vendor recommendations and responses to Generic Letter 84-15.

The diesel generators are equipped with a prelubrication system which maintains a continuous flow of oil to the diesel engine moving parts while the engine is shutdown. The purpose of this system is to increase long term diesel generator reliability by reducing the stress and wear caused by frequent dry starting of the diesel generator. The diesel generator prelube may be accomplished either through normal operation of the installed prelubrication system or by manual prelubrication of the diesel generator in accordance with the manufacturer's/vendor's instructions. Performance of an idle start of the diesel generator is not considered to be a means of prelubrication.

A periodic "start test" of the diesel generators demonstrates proper startup from standby conditions, and verifies that the required generator voltage and frequency is attained. For this test, the diesel generator may be slow started and reach rated speed on a prescribed schedule that is selected to minimize stress and wear. In cases where this Surveillance Requirement is being used to identify a possible common mode failure in accordance with the action provisions, this test eliminates the risk of paralleling two of the remaining A.C. sources, which may compromise the A.C. source independence.

A "load-run test" normally follows the periodic "start test" of the diesel generator to demonstrate operation at or near the continuous rating. This surveillance should only be conducted on one diesel generator at a time in order to avoid common mode failures that might result from offsite circuit or grid perturbations. A minimum run time of 60 minutes is required to stabilize engine temperatures. Actual run time should be in accordance with vendor recommendations with regard to good operating practice and should be sufficient to ensure that cooling and lubrication are adequate for extended periods of operation, while minimizing the time that the diesel generator is connected to the offsite source. This Surveillance Requirement may include gradual loading, as recommended by the manufacturer, so that mechanical stress and wear on the diesel engine are minimized. A load band is provided to avoid routine overloading of the diesel generators. Momentary transients outside the load band because of changing bus loads do not impact the validity of this test.

A periodic surveillance requirement is provided to assure the diesel generator is aligned to provide standby power on demand. Periodic surveillance requirements also verify that, without the aid of the refill compressor, sufficient air start capacity for each diesel generator is available. With either pair of air receiver tanks at the minimum specified pressure, there is sufficient air in the tanks to start the associated diesel generator.

The periodicity of surveillance requirements for the shared diesel generators shall be equivalent to those required for the unit diesel generators. For example, it is not the intention to perform surveillances for the shared diesel generators twice during the specified surveillance interval in order to satisfy each unit's diesel generator surveillance requirements. By appropriately staggering

BASES

the surveillance intervals between all three (3) diesel generators further ensures that for any loaded diesel generator surveillances, not more than one diesel generator is rendered inoperable at any given time in order to perform such testing.

Surveillance requirements provide verification that there is an adequate inventory of fuel oil in the storage tanks that is sufficient to provide time to place the facility in a safe shutdown condition and to bring in replenishment fuel from an offsite location. Additional diesel fuel can normally be obtained and delivered to the site within an eight hour period; thus a two day supply provides for adequate margin. The operation of each required fuel oil transfer pump is demonstrated by transferring fuel oil from its associated storage tank to its associated day tank. This surveillance provides assurance that the fuel oil transfer pump is OPERABLE, the fuel oil piping system is intact, the fuel delivery piping is not obstructed, and the necessary fuel oil day tank instrumentation is OPERABLE.

A comprehensive surveillance program is provided to ensure the availability of high quality fuel oil for the diesel generators which is necessary to ensure proper operation. Water content should be minimized, because water in the fuel would contribute to excessive corrosion of the system, causing decreased reliability. The growth of micro-organisms results in slime formations, which are one of the chief causes of jellying in hydrocarbon fuels. Therefore, minimizing such slimes is also essential to assuring high reliability.

Sampling of both new diesel fuel oil and the bulk fuel oil storage tanks is in accordance with the American Society for Testing Materials (ASTM) standard D4057. Testing for API gravity is in accordance with ASTM D1298, water and sediment is in accordance with ASTM D1796, and the visual test for free water and particulate contamination (clear and bright) is in accordance with ASTM D4176. Testing for kinematic viscosity is in accordance with ASTM D445 and particulate contaminant testing is in accordance with ASTM D2276. Parameter limits are in accordance with ASTM D396 for API gravity, ASTM D975 for water and sediment and for kinematic viscosity, and ASTM D4176 for "clear and bright." The specific revision in use for each of these standards is controlled by procedure.

The diesel fuel oil day tanks are not equipped with the capability to obtain samples. Any accumulated water is removed by partially draining the day tank to the bulk fuel oil storage tank on a routine basis. Monthly sampling of the bulk fuel oil storage tank is then used to detect the presence of any water.

Fuel oil testing may indicate that such fuel oil is not within the required parameters. However, continued operation is acceptable while measures are taken to restore the properties of the fuel oil to within its limits since the properties of interest, even if they were not within the required limits, would not have an immediate effect on diesel generator operation. If the fuel oil properties cannot be returned to within their limits in the allowed time, the associated diesel generator(s) is (are) declared inoperable and the appropriate ACTION(s) taken.

A semi-annual surveillance is provided to verify the diesel generator can "fast start" from standby conditions and achieve the required voltage and frequency within the timing assumptions of the design basis loss of coolant accident safety analysis. Conducting this test on a semi-annual frequency is consistent with the intent of the reduction of cold testing identified in Generic Letter 84-15.

Additional surveillance requirements provide for periodic inspections and demonstration of the diesel generator capabilities, some are conducted in conjunction with a simulated loss of offsite power and/or a simulated ESF actuation signal. These tests of the diesel generator are expected to be conducted during an outage to functionally test the system. This testing is consistent with the intent of the diesel generator reliability programs recommended by Regulatory Guide 1.155.

3/4.9.B A.C. Sources - Shutdown

The A.C. sources required during Cold Shutdown, Refueling, when handling irradiated fuel and during operations with a potential for draining the reactor vessel provide assurance that:

1. Systems to provide adequate coolant inventory makeup are available for the irradiated fuel in the core in case of an inadvertent draindown of the reactor vessel;

2. Systems needed to mitigate a fuel handling accident are available;

3. Systems necessary to mitigate the effects of events that can lead to core damage during shutdown are OPERABLE; and

4. Instrumentation and control capability is available for monitoring and maintaining the unit in a cold shutdown condition and refueling condition.

With one or more of the required A.C. electrical power sources inoperable, the action provisions require a suspension of activities that will preclude the occurrence of actions that could potentially initiate the postulated events. However, timely suspension of these activities is not intended to preclude completion of actions necessary to establish a safe, conservative condition.

The Surveillance Requirements for A.C. Source Shutdown are the same as those for operation, with the exception of the periodic "load-run test" which is not required due to the limited redundancy of A.C. power sources.

3/4.9.C D.C. Sources - Operating

The station D.C. electrical power system provides the A.C. emergency power system with control power. It also provides both motive and control power to selected safety-related equipment. During normal operation, the D.C. electrical loads are powered from the battery chargers with batteries floating on the system. In case of loss of normal power to the battery charger, the D.C. load is automatically powered from the station batteries.

BASES

Each battery of the D.C. electrical power systems is sized to start and carry the normal D.C. loads plus all D.C. loads required for safe shutdown on one unit and operations required to limit the consequences of a design basis event on the other unit for a period of 4 hours following loss of all A.C. sources. The battery chargers are sized to restore the battery to full charge under normal (non-emergency) load conditions. A normally disconnected alternate 125 volt battery is also provided as a backup for each normal battery. If both units are operating , the normal 125 volt battery must be returned to service within the specified time frame since the design configuration of the alternate battery circuit is susceptible to single failure and, hence, is not as reliable as the normal station circuit. During times when the other unit is in a Cold Shutdown or Refuel condition, an alternate 125 volt battery is available to replace a normal station 125 volt battery on a continuous basis to provide a second available power source. With the alternate 125 volt battery in service, the normally open breaker on the DC Reserve Bus is placed in the open position and posted, i.e., "tagged out."

With one of the required D.C. electrical power subsystems inoperable the remaining system has the capacity to support a safe shutdown and to mitigate an accident condition. However, a subsequent worst-case single failure would result in complete loss of ESF functions. Therefore, an allowed outage time is provided based on a reasonable time to assess plant status as a function of the inoperable D.C. electrical power subsystem and, if the D.C. electrical power subsystem is not restored to OPERABLE status, prepare to effect an orderly and safe plant shutdown.

Inoperable chargers do not necessarily indicate that the D.C. systems are not capable of performing their post-accident functions as long as the batteries are within their specified parameter limits. With both the required charger inoperable and the battery degraded, prompt action is required to assure an adequate D.C. power supply.

ACTION(s) are provided to delineate the measurements and time frames needed to continue to assure OPERABILITY of the Station batteries when battery parameters are outside their identified limits.

Battery surveillance requirements are based on the defined battery cell parameter values. Category A defines the normal parameter limit for each designated pilot cell in each battery. The pilot cells are the average cells in the battery based on previous test results. These cells are monitored closely as an indication of battery performance. Category B defines the normal parameter limits for each connected cell. The term "connected cell" excludes any battery cell that may be jumpered out because of a degraded condition or for any other reason. Category B also defines allowable values for each connected cell. These values, although reduced, provide assurance that sufficient capacity exists to perform the intended function and maintain a margin of safety. When any battery parameter is outside the Category B allowable value, the assurance of sufficient capacity as described above no longer exists and the battery must be declared inoperable.

Verifying battery terminal voltage while on float charge for the batteries helps to ensure the effectiveness of the charging system and the ability of the batteries to perform their intended function. The voltage requirements are based on the nominal design voltage of the battery and are consistent with the initial voltages assumed in the battery sizing calculations.

BASES

Visual inspection to detect corrosion of the battery cells and connections, or measurement of the resistance of each connection provides an indication of physical damage or abnormal deterioration that could potentially degrade battery performance. The limits established for this Surveillance Requirement shall be no more than 20% above the resistance as measured during installation or not above the ceiling value established by the manufacturer.

Verifying an acceptable average temperature of battery cells is consistent with the recommendations of IEEE-450 and ensures that lower than normal temperatures do not act to inhibit or reduce battery capacity.

Verifying that the chargers will provide the manufacturer's rated current and voltage for four hours ensures that charger deterioration has not occurred and that the charger will provide the necessary capacity to restore the battery to a fully charged state.

A battery service test is a special test of the battery's capability "as found" to satisfy the design requirements of the D.C. electrical power system. The discharge rate and test length should correspond to the design duty cycle requirements.

A battery modified performance test is a test of the battery capacity and the battery's ability to meet the loads that exceed the constant current discharge rate of the battery (high rate short duration loads) of the battery's duty cycle. This test satisfies the requirements of both a service test and a performance test and is intended to detect any change in capacity and to determine overall battery degradation due to age and usage. The batteries have a rated capacity of 125% of the load expected at the end of their service life allowing for a minimum battery capacity of at least 80% of the manufacturer's rating. A battery capacity of 80% indicates that the battery rate of deterioration is increasing, even if there is ample capacity to meet the load requirements.

3/4.9.D D.C. Sources - Shutdown

The D.C. sources required to be OPERABLE during Cold Shutdown, Refueling, when handling irradiated fuel and during operations with a potential for draining the reactor vessel provide assurance that:

1. Systems to provide adequate coolant inventory makeup are available for the irradiated fuel in the core in case of an inadvertent drain down of the reactor vessel;

2. Systems needed to mitigate a fuel-handling accident are available;

3. Systems necessary to mitigate the effects of events that can lead to core damage during shutdown are OPERABLE;

4. Instrumentation and control capability is available for monitoring and maintaining the unit in a cold shutdown condition and refueling condition.

BASES

With one or more of the required D.C. electrical power sources inoperable, the action provisions require a suspension of activities that will preclude the occurrence of actions that could potentially initiate the postulated events. However, timely suspension of these activities is not intended to preclude completion of actions necessary to establish a safe, conservative condition.

3/4.9.E Distribution - Operating

The OPERABILITY of the A.C. and D.C. onsite power distribution systems ensures that sufficient power will be available to the safety related equipment required for (1) the safe shutdown of the facility and (2) the mitigation and control of accident conditions within the facility.

The surveillance requirements verify that the A.C. and D.C. electrical power distribution systems are functioning properly, with all the required circuit breakers closed and the buses energized from normal power. The verification of proper voltage availability on the buses ensures that the required power is readily available for motive as well as control functions for critical system loads connected to these buses. The frequency takes into account the redundant capability of the A.C. and D.C. electrical power distribution subsystems, and other indications available in the control room that will alert the operator to subsystem malfunctions.

3/4.9.F Distribution - Shutdown

The OPERABILITY of the minimum specified A.C. and D.C. onsite power distribution systems, during Cold Shutdown and Refueling and when handling irradiated fuel in the secondary containment, ensures that the facility can be maintained in these conditions for extended time periods and sufficient instrumentation and control capability is available for monitoring and maintaining the unit status. Requiring OPERABILITY of the minimum specified onsite power distribution systems when handling irradiated fuel in the secondary containment helps to ensure that systems needed to mitigate a fuel handling accident are available.

3/4.9.G RPS Power Monitoring

Specifications are provided to ensure the OPERABILITY of the reactor protection system (RPS) bus electrical protection assemblies (EPAs). Each RPS motor generator (MG) set and the alternate power source has 2 EPA CHANNEL(s) wired in series. A trip of either CHANNEL from either overvoltage, undervoltage, or underfrequency will disconnect the associated MG set or alternate power source.

The associated surveillance requirements provide for demonstration of the OPERABILITY of the RPS EPA's. The setpoints for overvoltage, undervoltage, and underfrequency have been chosen based on analysis (ref. February 4, 1983 letter to H. Denton from T. Rausch).

3.10 - LIMITING CONDITIONS FOR OPERATION

H. Water Level - Spent Fuel Storage Pool

The pool water level shall be maintained at a level of 33 feet.

APPLICABILITY:

Whenever irradiated fuel assemblies are in the spent fuel storage pool.

ACTION:

With the requirements of the above specification not satisfied, suspend all movement of fuel assemblies and crane operations with loads in the spent fuel storage pool area after placing the fuel assemblies and crane load in a safe condition. The provisions of Specification 3.0.C are not applicable.

4.10 - SURVEILLANCE REQUIREMENTS

H. Water Level - Spent Fuel Storage Pool

The water level in the spent fuel storage pool shall be determined to be at least at its minimum required depth at least once per 7 days.

3.11 - LIMITING CONDITIONS FOR OPERATION

B. TRANSIENT LINEAR HEAT GENERATION RATE

The TRANSIENT LINEAR HEAT GENERATION RATE (TLHGR) shall be maintained such that the FUEL DESIGN LIMITING RATIO for CENTERLINE MELT (FDLRC) is less than or equal to 1.0. Where FDLRC is equal to:

$$\frac{(LHGR)\ (1.2)}{(TLHGR)\ (FRTP)}$$

APPLICABILITY:

OPERATIONAL MODE 1, when THERMAL POWER is greater than or equal to 25% of RATED THERMAL POWER.

ACTION:

With FDLRC greater than 1.0, initiate corrective ACTION within 15 minutes and within 6 hours either:

1. Restore FDLRC to less than or equal to 1.0, or
2. Adjust the flow biased APRM setpoints specified in Specifications 2.2.A and 3.2.E by 1/FDLRC, or
3. Adjust[a] each APRM gain such that the APRM readings are ≥100% times the FRACTION OF RATED THERMAL POWER (FRTP) times FDLRC.

With the provisions of the ACTION above not met, reduce THERMAL POWER to less than 25% of RATED THERMAL POWER within the next 4 hours.

4.11 - SURVEILLANCE REQUIREMENTS

B. TRANSIENT LINEAR HEAT GENERATION RATE

The value of FDLRC shall be verified:

1. At least once per 24 hours,
2. Within 12 hours after completion of a THERMAL POWER increase of at least 15% of RATED THERMAL POWER, and
3. Initially and at least once per 12 hours when the reactor is operating with FDLRC greater than or equal to 1.0.
4. The provisions of Specification 4.0.D are not applicable.

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a Provided that the adjusted APRM reading does not exceed 100% of RATED THERMAL POWER and a notice of adjustment is posted on the reactor control panel.

3.11 - LIMITING CONDITIONS FOR OPERATION

C. MINIMUM CRITICAL POWER RATIO

The MINIMUM CRITICAL POWER RATIO (MCPR) shall be equal to or greater than the MCPR operating limit specified in the CORE OPERATING LIMITS REPORT.

APPLICABILITY:

OPERATIONAL MODE 1, when THERMAL POWER is greater than or equal to 25% of RATED THERMAL POWER.

ACTION:

With MCPR less than the applicable MCPR limit as determined for one of the conditions specified in the CORE OPERATING LIMITS REPORT:

1. Initiate corrective ACTION within 15 minutes, and
2. Restore MCPR to within the required limit within 2 hours.

With the provisions of the ACTION above not met, reduce THERMAL POWER to less than 25% of RATED THERMAL POWER within the next 4 hours.

4.11 - SURVEILLANCE REQUIREMENTS

C. MINIMUM CRITICAL POWER RATIO

MCPR shall be determined to be equal to or greater than the applicable MCPR operating limit specified in the CORE OPERATING LIMITS REPORT.

1. At least once per 24 hours,
2. Within 12 hours after completion of a THERMAL POWER increase of at least 15% of RATED THERMAL POWER, and
3. Initially and at least once per 12 hours when the reactor is operating with a LIMITING CONTROL ROD PATTERN for MCPR.
4. The provisions of Specification 4.0.D are not applicable.

3.11 - LIMITING CONDITIONS FOR OPERATION

D. STEADY STATE LINEAR HEAT GENERATION RATE

The LINEAR HEAT GENERATION RATE (LHGR) for each type of fuel as a function of AVERAGE PLANAR EXPOSURE shall not exceed the STEADY STATE LINEAR HEAT GENERATION RATE (SLHGR) limits specified in the CORE OPERATING LIMITS REPORT.

APPLICABILITY:

OPERATIONAL MODE 1, when THERMAL POWER is greater than or equal to 25% of RATED THERMAL POWER.

ACTION:

With an LHGR exceeding the SLHGR limits specified in the CORE OPERATING LIMITS REPORT:

1. Initiate corrective ACTION within 15 minutes, and
2. Restore the LHGR to within the SLHGR limit within 2 hours.

With the provisions of the ACTION above not met, reduce THERMAL POWER to less than 25% of RATED THERMAL POWER within the next 4 hours.

4.11 - SURVEILLANCE REQUIREMENTS

D. STEADY STATE LINEAR HEAT GENERATION RATE

The SLHGR shall be determined to be equal to or less than the limit:

1. At least once per 24 hours,
2. Within 12 hours after completion of a THERMAL POWER increase of at least 15% of RATED THERMAL POWER, and
3. Initially and at least once per 12 hours when the reactor is operating with a LIMITING CONTROL ROD PATTERN for SLHGR.
4. The provisions of Specification 4.0.D are not applicable.

BASES

3/4.11.A AVERAGE PLANAR LINEAR HEAT GENERATION RATE

This specification assures that the peak cladding temperature following the postulated design basis loss-of-coolant accident will not exceed the limit specified in 10 CFR 50.46. The specification also assures that fuel rod mechanical integrity is maintained during normal and transient operations.

The peak cladding temperature (PCT) following a postulated loss-of-coolant accident is primarily a function of the average heat generation rate of all the rods of a fuel assembly at any axial location and is dependent only secondarily on the rod-to-rod power distribution within an assembly. The peak clad temperature is calculated assuming a LINEAR HEAT GENERATION RATE (LHGR) for the highest powered rod which is equal to or less than the design LHGR corrected for densification. The APLHGR limits specified are equivalent to the LHGR of the highest powered fuel rod assumed in the LOCA analysis divided by its local peaking factor. A conservative multiplier is applied to the LHGR assumed in the LOCA analysis to account for the uncertainty associated with the measurement of the APLHGR.

The calculational procedure used to establish the maximum APLHGR values uses NRC approved calculational models which are consistent with the requirements of Appendix K of 10 CFR Part 50. The approved calculational models are listed in Specification 6.9.

The daily requirement for calculating APLHGR when THERMAL POWER is greater than or equal to 25% of RATED THERMAL POWER is sufficient since power distribution shifts are very slow when there have not been significant power or control rod changes. The requirement to calculate APLHGR within 12 hours after the completion of a THERMAL POWER increase of at least 15% of RATED THERMAL POWER ensures thermal limits are met after power distribution shifts while still allotting time for the power distribution to stabilize. The requirement for calculating APLHGR after initially determining a LIMITING CONTROL ROD PATTERN exists ensures that APLHGR will be known following a change in THERMAL POWER or power shape, that could place operation above a thermal limit.

3/4.11.B TRANSIENT LINEAR HEAT GENERATION RATE

The flow biased neutron flux - high scram setting and control rod block functions of the APRM instruments for both two recirculation loop operation and single recirculation loop operation must be adjusted to ensure that ≥1% plastic strain does not occur; and, the fuel does not experience centerline melt during anticipated operational occurrences beginning at any power level and terminating at 120% of RATED THERMAL POWER. The scram settings and rod block settings are adjusted in accordance with the formula in this specification when the value of FDLRC indicates a higher peaked power distribution to ensure that an LHGR transient would not be increased in the degraded condition.

The daily requirement for calculating FDLRC when THERMAL POWER is greater than or equal to 25% of RATED THERMAL POWER is sufficient since power distribution shifts are very slow when

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there have not been significant power or control rod changes. The requirement to calculate FDLRC within 12 hours after the completion of a THERMAL POWER increase of at least 15% of RATED THERMAL POWER ensures thermal limits are met after power distribution shifts while still allotting time for the power distribution to stabilize. The requirement for calculating FDLRC after initially determining FDLRC is greater than 1.0 exists to ensure that FDLRC will be known following a change in THERMAL POWER or power shape that could place operation above a thermal limit.

The FUEL DESIGN LIMIT RATIO FOR CENTERLINE MELT (FDLRC) is defined as:

$$FDLRC = \frac{(LHGR)(1.2)}{(TLHGR)(FRTP)} ;$$

where LHGR is the LINEAR HEAT GENERATION RATE, and TLHGR is the TRANSIENT LINEAR HEAT GENERATION RATE. The TLHGR is specified in the CORE OPERATING LIMITS REPORT.

3/4.11.C MINIMUM CRITICAL POWER RATIO

The required operating limit MCPR at steady state operating conditions as specified in Specification 3.11.C are derived from the established fuel cladding integrity Safety Limit MCPR, and an analysis of abnormal operational transients. For any abnormal operating transient analysis evaluation with the initial condition of the reactor being at the steady state operating limit, it is required that the resulting MCPR does not decrease below the Safety Limit MCPR at any time during the transient assuming instrument trip setting given in Specification 2.2.

To assure that the fuel cladding integrity Safety Limit is not exceeded during any anticipated abnormal operational transient, the most limiting transients have been analyzed to determine which result in the largest reduction in the CRITICAL POWER RATIO (CPR). The type of transients evaluated were change of flow, increase in pressure and power, positive reactivity insertion, and coolant temperature decrease. The limiting transient yields the largest delta MCPR. When added to the Safety Limit MCPR, the required minimum operating limit MCPR of Specification 3.11.C is obtained and presented in the CORE OPERATING LIMITS REPORT.

The steady state values for MCPR specified were determined using NRC-approved methodology listed in Specification 6.9.

The purpose of the reduced flow MCPR curves specified in the CORE OPERATING LIMITS REPORT are to define MCPR operating limits at other than rated core flow conditions. The reduced flow MCPR curves assure that the Safety Limit MCPR will not be violated.

At THERMAL POWER levels less than or equal to 25% of RATED THERMAL POWER, the reactor will be operating at minimum recirculation pump speed and the moderator void content will be very small. For all designated control rod patterns which may be employed at this point, operating plant experience indicates that the resulting MCPR value has considerable margin. Thus, the demonstration of MCPR below this power level is unnecessary. The daily requirement for

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calculating MCPR when THERMAL POWER is greater than or equal to 25% of RATED THERMAL POWER is sufficient since power distribution shifts are very slow when there have not been significant power or control rod changes. The requirement for calculating MCPR after initially determining that a LIMITING CONTROL ROD PATTERN exists ensures that MCPR will be known following a change in THERMAL POWER or power shape, regardless of magnitude, that could place operation above a thermal limit.

3/4.11.D STEADY STATE LINEAR HEAT GENERATION RATE

This specification assures that the maximum LINEAR HEAT GENERATION RATE in any fuel rod is less than the design STEADY STATE LINEAR HEAT GENERATION RATE even if fuel pellet densification is postulated. This provides assurance that the fuel end-of-life steady state criteria are met. The daily requirement for calculating LHGR when THERMAL POWER is greater than or equal to 25% of RATED THERMAL POWER is sufficient since power distributions shifts are very slow when there have not been significant power or control rod changes. The requirement to calculate LHGR within 12 hours after the completion of a THERMAL POWER increase of at least 15% of RATED THERMAL POWER ensures thermal limits are met after power distribution shifts while still allotting time for the power distribution to stabilize. The requirement for calculating SLHGR after initially determining a LIMITING CONTROL ROD PATTERN exists ensures that SLHGR will be known following a change in THERMAL POWER or power shape that could place operation above a thermal limit.

6.2.B Unit Staff

The unit staff shall include the following:

1. Three non-licensed operators shall be on site at all times.

2. At least one licensed Reactor Operator shall be present in the control room when fuel is in the reactor. In addition, while the unit is in MODE(s) 1, 2, 3 or 4 at least one licensed Senior Reactor Operator shall be present in the control room.

3. Shift crew composition may be less than the minimum requirement of 10 CFR 50.54(m)(2)(i) and 6.2.B.1 and 6.2.C for a period of time not to exceed two 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.

4. A Radiation Protection Technician shall be on site when fuel is in the reactor. The position may be vacant for not more than two hours, in order to provide for unexpected absence, provided immediate action is taken to fill the required position.

5. Administrative procedures shall be developed and implemented to limit the working hours of unit staff who perform safety-related functions; e.g, senior reactor operators, reactor operators, health physicists, auxiliary operators, and key maintenance personnel.

   The amount of overtime worked by unit staff members performing safety-related functions shall be limited in accordance with the NRC Policy Statement on working hours (Generic Letter 82-12).

6. The Operations Manager or Shift Operations Supervisor shall hold a Senior Reactor Operator License.

6.2.C Shift Technical Advisor

The Shift Technical Advisor (STA) shall provide technical advisory support to the Unit Supervisor in the areas of thermal hydraulics, reactor engineering and plant analysis with regard to the safe operation of the facility. In addition, the STA shall meet the qualifications specified by the Commission Policy Statement on Engineering Expertise on Shift. A single STA may fulfill this function for both units.

6.8 PROCEDURES AND PROGRAMS

6.8.A Written procedures shall be established, implemented, and maintained covering the activities referenced below:

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

2. The Emergency Operating Procedures required to implement the requirements of NUREG-0737 and Supplement 1 to NUREG-0737 as stated in Section 7.1 of Generic Letter No. 82-33,

3. Station Security Plan implementation,

4. Generating Station Emergency Response Plan implementation,

5. PROCESS CONTROL PROGRAM (PCP) implementation,

6. OFFSITE DOSE CALCULATION MANUAL (ODCM) implementation, and

7. Fire Protection Program implementation.

6.8.B Deleted.

6.8.C Deleted

6.8.D The following programs shall be established, implemented, and maintained:

1. Reactor Coolant Sources Outside Primary Containment

   This program provides controls to minimize leakage from those portions of systems outside primary containment that could contain highly radioactive fluids during a serious transient or accident to as low as practical levels. The systems include CS, HPCI, LPCI, IC, process sampling, containment monitoring, and standby gas treatment systems. The program shall include the following:

   a. Preventive maintenance and periodic visual inspection requirements, and

   b. Leak test requirements for each system at a frequency of at least once per operating cycle.

6.12.B In addition to the requirements of 6.12.A, areas accessible to personnel with radiation levels greater than 1000 mrem/hr at 30 cm (12 in.) from the radiation source or from any surface which the radiation penetrates shall require the following:

1. Doors shall be locked to prevent unauthorized entry and shall not prevent individuals from leaving the area. In place of locking the door, direct or electronic surveillance that is capable of preventing unauthorized entry may be used. The keys shall be maintained under the administrative control of the Shift Manager on duty and/or health physics supervision.

2. Personnel access and exposure control requirements of activities being performed within these areas shall be specified by an approved RWP(or equivalent document).

3. Each person entering the area shall be provided with an alarming radiation monitoring device that continuously integrates the radiation dose rate (such as an electronic dosimeter.) Surveillance and radiation monitoring by a Radiation Protection Technician may be substituted for an alarming dosimeter.

4. During emergency situations which involve personnel injury or actions taken to prevent major equipment damage, surveillance and radiation monitoring of the work area by a qualified individual may be substituted for the routine RWP (or equivalent document).

5. For individual HIGH RADIATION AREAS accessible to personnel with radiation levels of greater than 1000 mrem/h at 30 cm (12 in.) that are located within large areas where no enclosure exists for purposes of locking, and where no enclosure can be reasonably constructed around the individual areas, then such individual areas shall be barricaded, conspicuously posted, and a flashing light shall be activated as a warning device.

QUAD CITIES STATION

UNITS 1 & 2

(DPR-29 & DPR-30)

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1.0 DEFINITIONS

CHANNEL FUNCTIONAL TEST

A CHANNEL FUNCTIONAL TEST shall be:

a. Analog CHANNEL(s) - the injection of a simulated signal into the CHANNEL as close to the sensor as practicable to verify OPERABILITY including required alarm and/or trip functions and CHANNEL failure trips.

b. Bistable CHANNEL(s) - the injection of a simulated signal into the sensor to verify OPERABILITY including required alarm and/or trip functions.

The CHANNEL FUNCTIONAL TEST may be performed by any series of sequential, overlapping or total CHANNEL steps such that the entire CHANNEL is tested.

CORE ALTERATION

CORE ALTERATION shall be the movement of any fuel, sources, or reactivity control components, within the reactor vessel with the vessel head removed and fuel in the vessel. The following exceptions are not considered to be CORE ALTERATIONS:

a. Movement of source range monitors, local power range monitors, intermediate range monitors, traversing incore probes, or special movable detectors (including undervessel replacement); and

b. Control rod movement, provided there are no fuel assemblies in the associated control cell.

Suspension of CORE ALTERATIONS shall not preclude completion of movement of a component to a safe position.

CORE OPERATING LIMITS REPORT (COLR)

The CORE OPERATING LIMITS REPORT (COLR) shall be the unit specific document that provides core operating limits for the current operating cycle. These cycle specific core operating limits shall be determined for each operating cycle in accordance with Specification 6.9. Plant operation within these operating limits is addressed in individual specifications.

CRITICAL POWER RATIO (CPR)

The CRITICAL POWER RATIO (CPR) shall be the ratio of that power in the assembly which is calculated by application of the applicable NRC approved critical power correlation to cause some point in the assembly to experience transition boiling, divided by the actual assembly power.

DOSE EQUIVALENT I-131

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

1.0 DEFINITIONS

FRACTION OF LIMITING POWER DENSITY (FLPD)
The FRACTION OF LIMITING POWER DENSITY (FLPD) shall be the LHGR existing at a given location divided by the specified LHGR limit for that bundle.

FRACTION OF RATED THERMAL POWER (FRTP)
The FRACTION OF RATED THERMAL POWER (FRTP) shall be the measured THERMAL POWER divided by the RATED THERMAL POWER.

FREQUENCY NOTATION
The FREQUENCY NOTATION specified for the performance of Surveillance Requirements shall correspond to the intervals defined in Table 1-1.

IDENTIFIED LEAKAGE
IDENTIFIED LEAKAGE shall be: a) leakage into primary containment collection systems, such as pump seal or valve packing leaks, that is captured and conducted to a sump or collecting tank, or b) leakage into the primary containment atmosphere from sources that are both specifically located and known either not to interfere with the operation of the leakage detection systems or not to be PRESSURE BOUNDARY LEAKAGE.

LIMITING CONTROL ROD PATTERN (LCRP)
A LIMITING CONTROL ROD PATTERN (LCRP) shall be a pattern which results in the core being on a thermal hydraulic limit, i.e., operating on a limiting value for APLHGR, LHGR, or MCPR.

LINEAR HEAT GENERATION RATE (LHGR)
LINEAR HEAT GENERATION RATE (LHGR) shall be the heat generation per unit length of fuel rod. It is the integral of the heat flux over the heat transfer area associated with the unit length.

LOGIC SYSTEM FUNCTIONAL TEST (LSFT)
A LOGIC SYSTEM FUNCTIONAL TEST (LSFT) shall be a test of all required logic components, i.e., all required relays and contacts, trip units, solid state logic elements, etc, of a logic circuit, from as close to the sensor as practicable up to, but not including the actuated device, to verify OPERABILITY. The LOGIC SYSTEM FUNCTIONAL TEST may be performed by means of any series of sequential, overlapping or total system steps so that the entire logic system is tested.

MAXIMUM FRACTION OF LIMITING POWER DENSITY (MFLPD)
The MAXIMUM FRACTION OF LIMITING POWER DENSITY (MFLPD) shall be the highest value of the FLPD which exists in the core.

MINIMUM CRITICAL POWER RATIO (MCPR)
The MINIMUM CRITICAL POWER RATIO (MCPR) shall be the smallest CPR which exists in the core for each class of fuel.

1.0 DEFINITIONS

OFFSITE DOSE CALCULATION MANUAL (ODCM)

The OFFSITE DOSE CALCULATION MANUAL (ODCM) shall contain the methodology and parameters used in the calculation of offsite doses resulting from radioactive gaseous and liquid effluents, in the calculation of gaseous and liquid effluent monitoring Alarm/Trip Setpoints, and in the conduct of the Environmental Radiological Monitoring Program. The ODCM shall also contain (1) the Radioactive Effluent Controls and Radiological Environmental Monitoring Programs required by Specification 6.8 and (2) descriptions of the information that should be included in the Annual Radiological Environmental Operating and Semi-annual Radioactive Effluent Release Reports required by Specification 6.9.

OPERABLE - OPERABILITY

A system, subsystem, train, component, or device shall be OPERABLE or have OPERABILITY when it is capable of performing its specified safety function(s) and when all necessary attendant instrumentation, controls, normal or emergency electrical power, cooling or seal water, lubrication or other auxiliary equipment that are required for the system, subsystem, train, component or device to perform its specified safety function(s) are also capable of performing their related support function(s).

OPERATIONAL MODE

An OPERATIONAL MODE, i.e., MODE, shall be any one inclusive combination of mode switch position and average reactor coolant temperature as specified in Table 1-2.

PHYSICS TESTS

PHYSICS TESTS shall be those tests performed to measure the fundamental nuclear characteristics of the reactor core and related instrumentation and 1) described in Chapter 14 of the UFSAR, 2) authorized under the provisions of 10 CFR 50.59, or 3) otherwise approved by the Commission.

PRESSURE BOUNDARY LEAKAGE

PRESSURE BOUNDARY LEAKAGE shall be leakage through a non-isolable fault in a reactor coolant system component body, pipe wall or vessel wall.

TABLE 1-1

SURVEILLANCE FREQUENCY NOTATION

	NOTATION	FREQUENCY
1. Shift	S	At least once per 12 hours
2. Day	D	At least once per 24 hours
3. Week	W	At least once per 7 days
4. Month	M	At least once per 31 days
5. Quarter	Q	At least once per 92 days
6. Semiannual	SA	At least once per 184 days
7. Annual	A	At least once per 366 days
8. Sesquiannual	E	At least once per 18 months (550 days)
9. Startup	S/U	Prior to each reactor startup
10. Not Applicable	NA	Not applicable

2.0 SAFETY LIMITS AND LIMITING SAFETY SYSTEM SETTINGS

2.1 SAFETY LIMITS

THERMAL POWER, Low Pressure or Low Flow

2.1.A THERMAL POWER shall not exceed 25% of RATED THERMAL POWER with the reactor vessel steam dome pressure less than 785 psig or core flow less than 10% of rated flow.

APPLICABILITY: OPERATIONAL MODE(s) 1 and 2.

ACTION:

With THERMAL POWER exceeding 25% of RATED THERMAL POWER and the reactor vessel steam dome pressure less than 785 psig or core flow less than 10% of rated flow, be in at least HOT SHUTDOWN within 2 hours and comply with the requirements of Specification 6.7.

THERMAL POWER, High Pressure and High Flow

2.1.B The MINIMUM CRITICAL POWER RATIO (MCPR) shall not be less than 1.07 with the reactor vessel steam dome pressure greater than or equal to 785 psig and core flow greater than or equal to 10% of rated flow. During single recirculation loop operation, this MCPR limit shall be increased by 0.01.

APPLICABILITY: OPERATIONAL MODE(s) 1 and 2.

ACTION:

With MCPR less than the above applicable limit and the reactor vessel steam dome pressure greater than or equal to 785 psig and core flow greater than or equal to 10% of rated flow, be in at least HOT SHUTDOWN within 2 hours and comply with the requirements of Specification 6.7.

2.0 SAFETY LIMITS AND LIMITING SAFETY SYSTEM SETTINGS

Reactor Coolant System Pressure

2.1.C The reactor coolant system pressure, as measured in the reactor vessel steam dome, shall not exceed 1345 psig.

APPLICABILITY: OPERATIONAL MODE(s) 1, 2, 3 and 4.

ACTION:

With the reactor coolant system pressure, as measured in the reactor vessel steam dome, above 1345 psig, be in at least HOT SHUTDOWN with reactor coolant system pressure less than or equal to 1345 psig within 2 hours and comply with the requirements of Specification 6.7.

Reactor Vessel Water Level

2.1.D The reactor vessel water level shall be greater than twelve inches above the top of the active irradiated fuel.

APPLICABILITY: OPERATIONAL MODE(s) 3, 4 and 5.

ACTION:

With the reactor vessel water level at or below twelve inches above the top of the active irradiated fuel, manually initiate the ECCS to restore the water level, after depressurizing the reactor vessel, if required, and comply with the requirements of Specification 6.7.

3.0 - LIMITING CONDITIONS FOR OPERATION

A. Compliance with the Limiting Conditions for Operation contained in the succeeding Specifications is required during the OPERATIONAL MODE(s) or other conditions specified therein; except that upon failure to meet the Limiting Conditions for Operation, the associated ACTION requirements shall be met.

B. Noncompliance with a Specification shall exist when the requirements of the Limiting Condition for Operation and associated ACTION requirements are not met within the specified time intervals. If the Limiting Condition for Operation is restored prior to expiration of the specified time intervals, completion of the ACTION requirements is not required.

C. When a Limiting Condition for Operation is not met, except as provided in the associated ACTION requirements, within one hour ACTION shall be initiated to place the unit in an OPERATIONAL MODE in which the Specification does not apply by placing it, as applicable, in:

1. At least HOT SHUTDOWN within the next 12 hours, and

2. At least COLD SHUTDOWN within the subsequent 24 hours.

Where corrective measures are completed that permit operation under the ACTION requirements, the ACTION may be taken in accordance with the specified time limits as measured from the time of failure to meet the Limiting Condition for Operation. Exceptions to these requirements are stated in the individual Specifications.

This Specification is not applicable in OPERATIONAL MODE 4 or 5.

D. When an LCO is not met, entry into a MODE or other specified condition in the Applicability shall not be made except when the associated ACTIONS to be entered permit continued operation in the MODE or other specified condition in the Applicability for an unlimited period of time. This Specification shall not prevent changes in MODES or other specified conditions in the Applicability that are required to comply with ACTIONS. Exceptions to these requirements are stated in the individual Specifications.

4.0 - SURVEILLANCE REQUIREMENTS

A. Surveillance Requirements shall be met during the reactor OPERATIONAL MODE(s) or other conditions specified for individual Limiting Conditions for Operation unless otherwise stated in an individual Surveillance Requirement.

B. Each Surveillance Requirement shall be performed within the specified surveillance interval with a maximum allowable extension not to exceed 25 percent of the surveillance interval.

C. Failure to perform a Surveillance Requirement within the allowed surveillance interval, defined by Specification 4.0.B, shall constitute noncompliance with the OPERABILITY requirements for a Limiting Condition for Operation. The time limits of the ACTION requirements are applicable at the time it is identified that a Surveillance Requirement has not been performed. The ACTION requirements may be delayed for up to 24 hours to permit the completion of the surveillance when the allowable outage time limits of the ACTION requirements are less than 24 hours. Surveillance requirements do not have to be performed on inoperable equipment.

D. Entry into an OPERATIONAL MODE or other specified applicable condition shall not be made unless the Surveillance Requirement(s) associated with the Limiting Condition for Operation have been performed within the applicable surveillance interval or as otherwise specified. This provision shall not prevent passage through or to OPERATIONAL MODE(s) as required to comply with ACTION requirements.

E. Surveillance Requirements for inservice inspection and testing of ASME Code Class 1, 2, and 3 components shall be applicable as follows:

   1. Inservice Inspection of ASME Code Class 1, 2, and 3 components and inservice testing of ASME Code Class 1, 2, and 3 pumps and valves shall be performed in accordance with Section XI of the ASME Boiler and Pressure Vessel Code and applicable Addenda as required by 10 CFR Part 50, Section 50.55a(g) and 50.55a(f), respectively, except where specific written relief has been granted by the Commission pursuant to 10 CFR Part 50, Section 50.55a(g)(6)(i) or 50.55a(f)(6)(i), respectively.

4.0 - SURVEILLANCE REQUIREMENTS

2. Surveillance intervals specified in Section XI of the ASME Boiler and Pressure Vessel Code and applicable Addenda for the inservice inspection and testing activities required by the ASME Boiler and Pressure Vessel Code and applicable Addenda shall be applicable as follows in these Technical Specifications:

ASME Boiler and Pressure Vessel Code and applicable Addenda terminology for inservice inspection and testing activities	Required Frequencies for performing inservice inspection and testing activities
Weekly	At least once per 7 days
Monthly	At least once per 31 days
Quarterly or every 3 months	At least once per 92 days
Semiannually or every 6 months	At least once per 184 days
Every 9 months	At least once per 276 days
Yearly or annually	At least once per 366 days
Biennially or every 2 years	At least once per 731 days

3. The provisions of Specification 4.0.B are applicable to the above required frequencies for performing inservice inspection and testing activities.

4. Performance of the above inservice inspection and testing activities shall be in addition to other specified Surveillance Requirements.

5. Nothing in the ASME Boiler and Pressure Vessel Code shall be construed to supersede the requirements of any Technical Specification.

6. The Inservice Inspection Program for piping identified in NRC Generic Letter 88-01 shall be performed in accordance with the staff positions on schedule, methods, and personnel and sample expansion included in Generic Letter 88-01 or in accordance with alternate measures approved by the NRC staff.

3.1 - LIMITING CONDITIONS FOR OPERATION

A. Reactor Protection System (RPS)

The reactor protection system (RPS) instrumentation CHANNEL(s) shown in Table 3.1.A-1 shall be OPERABLE.

APPLICABILITY:

As shown in Table 3.1.A-1.

ACTION:

1. With the number of OPERABLE CHANNEL(s) less than required by the Minimum CHANNEL(s) per TRIP SYSTEM requirement for one TRIP SYSTEM, place the inoperable CHANNEL(s) and/or that TRIP SYSTEM in the tripped condition[a] within 1 hour.

2. With the number of OPERABLE CHANNEL(s) less than required by the Minimum CHANNEL(s) per TRIP SYSTEM requirement for both TRIP SYSTEM(s), place at least one TRIP SYSTEM in the tripped condition[b] within 1 hour and take the ACTION required by Table 3.1.A-1.

4.1 - SURVEILLANCE REQUIREMENTS

A. Reactor Protection System

1. Each reactor protection system instrumentation CHANNEL shall be demonstrated OPERABLE by the performance of the CHANNEL CHECK, CHANNEL FUNCTIONAL TEST and CHANNEL CALIBRATION operations for the OPERATIONAL MODE(s) and at the frequencies shown in Table 4.1.A-1.

2. LOGIC SYSTEM FUNCTIONAL TEST(s) of all CHANNEL(s) shall be performed at least once per 18 months.

3. The response time of each reactor trip functional unit shown in Table 3.1.A-1 shall be demonstrated at least once per 18 months. Each test shall include at least one CHANNEL per TRIP SYSTEM such that all CHANNEL(s) are tested at least once every N times 18 months where N is the total number of redundant CHANNEL(s) in a specific reactor TRIP SYSTEM.

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a An inoperable CHANNEL need not be placed in the tripped condition when this would cause the trip function to occur. In these cases, the inoperable CHANNEL shall be restored to OPERABLE status within 2 hours or the ACTION required by Table 3.1.A-1 for that trip function shall be taken.

b The TRIP SYSTEM need not be placed in the tripped condition if this would cause the trip function to occur. When a TRIP SYSTEM can be placed in the tripped condition without causing the trip function to occur, place the TRIP SYSTEM with the most inoperable CHANNEL(s) in the tripped condition; if both systems have the same number of inoperable CHANNEL(s), place either TRIP SYSTEM in the tripped condition.

TABLE 4.1.A-1

REACTOR PROTECTION SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

Functional Unit	Applicable OPERATIONAL MODES	CHANNEL CHECK	CHANNEL FUNCTIONAL TEST	CHANNEL[a] CALIBRATION
1. Intermediate Range Monitor:				
a. Neutron Flux - High	2	S[b]	S/U[c], W[o]	E[o,q]
	3, 4, 5	S	W[o]	E[o,q]
b. Inoperative	2, 3, 4, 5	NA	W[o]	NA
2. Average Power Range Monitor[f]:				
a. Setdown Neutron Flux - High	2	S[b]	S/U[c], W[o]	SA[o]
	3, 5[m]	S	W	SA
b. Flow Biased Neutron Flux - High	1	S, D[g]	W	W[d,e], SA
c. Fixed Neutron Flux - High	1	S	W	W[d], SA
d. Inoperative	1, 2, 3, 5[m]	NA	W	NA
3. Reactor Vessel Steam Dome Pressure - High	1, 2[i]	NA	M	Q
4. Reactor Vessel Water Level - Low	1, 2	D	M	E[h]
5. Main Steam Line Isolation Valve - Closure	1	NA	M	E
6. Main Steam Line Radiation - High	1, 2[l]	S	M	E[p]
7. Drywell Pressure - High	1, 2[n]	NA	M	Q

TABLE 4.1.A-1 (Continued)

REACTOR PROTECTION SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

TABLE NOTATION

(a) Neutron detectors may be excluded from the CHANNEL CALIBRATION.

(b) The IRM and SRM channels shall be determined to overlap for at least (½) decades during each startup after entering OPERATIONAL MODE 2 and the IRM and APRM channels shall be determined to overlap for at least (½) decades during each controlled shutdown, if not performed within the previous 7 days.

(c) Within 24 hours prior to startup, if not performed within the previous 7 days. The weekly CHANNEL FUNCTIONAL TEST may be used to fulfill this requirement.

(d) This calibration shall consist of the adjustment of the APRM CHANNEL to conform, within 2% of RATED THERMAL POWER, to the power values calculated by a heat balance during OPERATIONAL MODE 1 when THERMAL POWER is ≥25% of RATED THERMAL POWER. This adjustment must be accomplished: a) within 2 hours if the APRM CHANNEL is indicating lower power values than the heat balance, or b) within 12 hours if the APRM CHANNEL is indicating higher power values than the heat balance. Until any required APRM adjustment has been accomplished, notification shall be posted on the reactor control panel.

Any APRM CHANNEL gain adjustment made in compliance with Specification 3.11.B shall not be included in determining the above difference. This calibration is not required when THERMAL POWER is <25% of RATED THERMAL POWER. The provisions of Specification 4.0.D are not applicable.

(e) This calibration shall consist of the adjustment of the APRM flow biased channel to conform to a calibrated flow signal.

(f) The LPRMs shall be calibrated at least once per 2000 effective full power hours (EFPH).

(g) Verify measured recirculation loop flow to be greater than or equal to established recirculation loop flow at the existing pump speed.

(h) Trip units are calibrated at least once per 31 days and transmitters are calibrated at the frequency identified in the table.

(i) This function is not required to be OPERABLE when the reactor pressure vessel head is unbolted or removed per Specification 3.12.A.

(j) With any control rod withdrawn. Not applicable to control rods removed per Specification 3.10.I or 3.10.J.

(k) This function may be bypassed, provided a control rod block is actuated, for reactor protection system reset in Refuel and Shutdown positions of the reactor mode switch.

TABLE 4.1.A-1 (Continued)

REACTOR PROTECTION SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

(l) This function not required to be OPERABLE when THERMAL POWER is less than 45% of RATED THERMAL POWER.

(m) Required to be OPERABLE only prior to and during required SHUTDOWN MARGIN demonstrations performed per Specification 3.12.B.

(n) This function is not required to be OPERABLE when PRIMARY CONTAINMENT INTEGRITY is not required.

(o) The provisions of Specification 4.0.D are not applicable to the CHANNEL FUNCTIONAL TEST and CHANNEL CALIBRATION surveillances for a period of 24 hours after entering OPERATIONAL MODE 2 or 3 when shutting down from OPERATIONAL MODE 1.

(p) A current source provides an instrument channel alignment every 3 months.

(q) The CHANNEL CALIBRATION surveillance requirements shall be performed if not performed within the previous seven days.

3.2 - LIMITING CONDITIONS FOR OPERATION

A. Isolation Actuation

The isolation actuation instrumentation CHANNEL(s) shown in Table 3.2.A-1 shall be OPERABLE with their trip setpoints set consistent with the values shown in the Trip Setpoint column.

APPLICABILITY:

As shown in Table 3.2.A-1.

ACTION:

1. With an isolation actuation instrumentation CHANNEL trip setpoint less conservative than the value shown in the Trip Setpoint column of Table 3.2.A-1, declare the CHANNEL inoperable until the CHANNEL is restored to OPERABLE status with its trip setpoint adjusted consistent with the Trip Setpoint value.
2. With the number of OPERABLE CHANNEL(s) less than required by the Minimum CHANNEL(s) per TRIP SYSTEM requirement for one TRIP SYSTEM, place the inoperable CHANNEL(s) and/or TRIP SYSTEM in the tripped condition[a] within one hour.

4.2 - SURVEILLANCE REQUIREMENTS

A. Isolation Actuation

1. Each isolation actuation instrumentation CHANNEL shall be demonstrated OPERABLE by the performance of the CHANNEL CHECK, CHANNEL FUNCTIONAL TEST and CHANNEL CALIBRATION operations for the OPERATIONAL MODE(s) and at the frequencies shown in Table 4.2.A-1.
2. LOGIC SYSTEM FUNCTIONAL TEST(s) of all CHANNEL(s) shall be performed at least once per 18 months.

---

a An inoperable CHANNEL need not be placed in the tripped condition where this would cause the trip function to occur. In these cases, the inoperable CHANNEL shall be restored to OPERABLE status within 2 hours or the ACTION required by Table 3.2.A-1 for that trip function shall be taken.

TABLE 3.2.A-1

ISOLATION ACTUATION INSTRUMENTATION

Functional Unit	Trip Setpoint[j]	Minimum CHANNEL(s) per TRIP SYSTEM[a]	Applicable OPERATIONAL MODE(s)	ACTION
1. PRIMARY CONTAINMENT ISOLATION				
a. Reactor Vessel Water Level - Low	≥144 inches	2	1, 2, 3	20
b. Drywell Pressure - High[d]	≤2.5 psig	2	1, 2, 3	20
c. Drywell Radiation - High	≤100 R/hr	1	1, 2, 3	23
2. SECONDARY CONTAINMENT ISOLATION				
a. Reactor Vessel Water Level - Low[c,k]	≥144 inches	2	1, 2, 3 & *	24
b. Drywell Pressure - High[c,d,k]	≤2.5 psig	2	1, 2, 3	24
c. Reactor Building Ventilation Exhaust Radiation - High[c,k]	≤3 mR/hr	2	1, 2, 3 & **	24
d. Refueling Floor Radiation - High[c,k]	≤100 mR/hr	2	1, 2, 3 & **	24
3. MAIN STEAM LINE (MSL) ISOLATION				
a. Reactor Vessel Water Level - Low Low	≥84 inches	2	1, 2, 3	21
b. MSL Tunnel Radiation - High[b]	≤15[h] x normal background	2	1, 2, 3	21
c. MSL Pressure - Low	≥825 psig	2	1	22
d. MSL Flow - High[k]	≤140% of rated	2/line	1, 2, 3	21
e. MSL Tunnel Temperature - High	≤200°F	2 of 4 in each of 2 sets	1, 2, 3	21

TABLE 4.2.A-1

ISOLATION ACTUATION INSTRUMENTATION SURVEILLANCE REQUIREMENTS

Functional Unit	CHANNEL CHECK	CHANNEL FUNCTIONAL TEST	CHANNEL CALIBRATION	Applicable OPERATIONAL MODE(s)
1. PRIMARY CONTAINMENT ISOLATION				
a. Reactor Vessel Water Level - Low	S	M	E[a]	1, 2, 3
b. Drywell Pressure - High[b]	NA	M	Q	1, 2, 3
c. Drywell Radiation - High	S	M	E	1, 2, 3
2. SECONDARY CONTAINMENT ISOLATION				
a. Reactor Vessel Water Level - Low[c,d]	S	M	E[a]	1, 2, 3 & *
b. Drywell Pressure - High[b,c,d]	NA	M	Q	1, 2, 3
c. Reactor Building Ventilation Exhaust Radiation - High[c,d]	S	M	E	1, 2, 3 & **
d. Refueling Floor Radiation - High[c,d]	S	M	E	1, 2, 3 & **
3. MAIN STEAM LINE (MSL) ISOLATION				
a. Reactor Vessel Water Level - Low Low	S	M	E[a]	1, 2, 3
b. MSL Tunnel Radiation - High	S	M	E[e]	1, 2, 3
c. MSL Pressure - Low	NA	M	Q	1
d. MSL Flow - High[d]	S	M	E	1, 2, 3
e. MSL Tunnel Temperature - High	NA	E	E	1, 2, 3

TABLE 4.2.A-1 (Continued)

ISOLATION ACTUATION INSTRUMENTATION SURVEILLANCE REQUIREMENTS

Functional Unit	CHANNEL CHECK	CHANNEL FUNCTIONAL TEST	CHANNEL CALIBRATION	Applicable OPERATIONAL MODE(s)
4. REACTOR WATER CLEANUP SYSTEM ISOLATION				
a. Standby Liquid Control System Initiation	NA	E	NA	1, 2, 3
b. Reactor Vessel Water Level - Low	S	M	E[a]	1, 2, 3
5. REACTOR CORE ISOLATION COOLING ISOLATION				
a. Steam Flow - High	NA	M	Q	1, 2, 3
b. Reactor Vessel Pressure - Low	NA	M	Q	1, 2, 3
c. Area Temperature - High	NA	E	E	1, 2, 3
6. HIGH PRESSURE COOLANT INJECTION ISOLATION				
a. Steam Flow - High	NA	M	E[a]	1, 2, 3
b. Reactor Vessel Pressure - Low	NA	M	E[a]	1, 2, 3
c. Area Temperature - High	NA	E	E	1, 2, 3
7. RHR SHUTDOWN COOLING MODE ISOLATION				
a. Reactor Vessel Water Level - Low	S	M	E[a]	3, 4, 5
b. Reactor Vessel Pressure - High (Cut-in Permissive)	NA	M	Q	1, 2, 3

TABLE 4.2.A-1 (Continued)

ISOLATION ACTUATION INSTRUMENTATION
SURVEILLANCE REQUIREMENTS

TABLE NOTATION

* During CORE ALTERATIONS or operations with a potential for draining the reactor vessel.

** When handling irradiated fuel in the secondary containment.

(a) Trip units are calibrated at least once per 31 days and transmitters are calibrated at the frequency identified in the table.

(b) This function is not required to be OPERABLE when PRIMARY CONTAINMENT INTEGRITY is not required.

(c) Isolates the reactor building ventilation system and actuates the standby gas treatment system.

(d) Also isolates the control room ventilation system.

(e) These instrument channels will be calibrated using simulated electrical signals once every three months. In addition, calibration including the sensors will be performed every 18 months.

3.2 - LIMITING CONDITIONS FOR OPERATION

B. Emergency Core Cooling Systems (ECCS) Actuation

The ECCS actuation instrumentation CHANNEL(s) shown in Table 3.2.B-1 shall be OPERABLE with their trip setpoints set consistent with the values shown in the Trip Setpoint column.

APPLICABILITY:

As shown in Table 3.2.B-1.

ACTION:

1. With an ECCS actuation instrumentation CHANNEL trip setpoint less conservative than the value shown in the Trip Setpoint column of Table 3.2.B-1, declare the CHANNEL inoperable until the CHANNEL is restored to OPERABLE status with its trip setpoint adjusted consistent with the Trip Setpoint value.
2. With one or more ECCS actuation instrumentation CHANNEL(s) inoperable, take the ACTION required by Table 3.2.B-1.
3. With either ADS TRIP SYSTEM inoperable, restore the inoperable TRIP SYSTEM to OPERABLE status within:
   a. 7 days provided that both the HPCI and RCIC systems are OPERABLE, or
   b. 72 hours.

   With the above provisions of this ACTION not met, be in at least HOT

4.2 - SURVEILLANCE REQUIREMENTS

B. ECCS Actuation

1. Each ECCS actuation instrumentation CHANNEL shall be demonstrated OPERABLE by the performance of the CHANNEL CHECK, CHANNEL FUNCTIONAL TEST and CHANNEL CALIBRATION operations for the OPERATIONAL MODE(s) and at the frequencies shown in Table 4.2.B-1.
2. LOGIC SYSTEM FUNCTIONAL TEST(s) of all CHANNEL(s) shall be performed at least once per 18 months.

3.2 - LIMITING CONDITIONS FOR OPERATION

C. ATWS - RPT

The anticipated transient without scram recirculation pump trip (ATWS - RPT) instrumentation CHANNEL(s) shown in Table 3.2.C-1 shall be OPERABLE with their trip setpoints set consistent with the values shown in the Trip Setpoint column.

APPLICABILITY:

OPERATIONAL MODE 1.

ACTION:

1. With an ATWS - RPT instrumentation CHANNEL trip setpoint less conservative than the value shown in the Trip Setpoint column of Table 3.2.C-1, declare the CHANNEL inoperable until the CHANNEL is restored to OPERABLE status with the CHANNEL trip setpoint adjusted consistent with the Trip Setpoint value.
2. With one level CHANNEL or one pressure CHANNEL inoperable in one or both TRIP SYSTEM(s), within 14 days, either restore the inoperable CHANNEL to OPERABLE status or place the inoperable CHANNEL in the tripped condition[a]. Otherwise, be in STARTUP within the next 6 hours.
3. With two level CHANNELS or two pressure CHANNELS inoperable in one or both TRIP SYSTEM(s), declare the TRIP SYSTEM(s) inoperable.

4.2 - SURVEILLANCE REQUIREMENTS

C. ATWS - RPT

1. Each ATWS - RPT instrumentation CHANNEL shall be demonstrated OPERABLE by the performance of the CHANNEL CHECK, CHANNEL FUNCTIONAL TEST and CHANNEL CALIBRATION operations at the frequencies shown in Table 4.2.C-1.
2. LOGIC SYSTEM FUNCTIONAL TEST(s) of all CHANNEL(s) shall be performed at least once per 18 months.

---

a The inoperable CHANNEL(s) need not be placed in the tripped condition where this would cause the Trip Function to occur.

3.2 - LIMITING CONDITIONS FOR OPERATION

4. With one level CHANNEL and one pressure CHANNEL inoperable in one or both TRIP SYSTEM(s), restore at least one inoperable CHANNEL to OPERABLE status within 14 days or be in STARTUP within the next 6 hours.

5. With one TRIP SYSTEM inoperable, restore the inoperable TRIP SYSTEM to OPERABLE status within 72 hours or be in at least STARTUP within the next 6 hours.

6. With both TRIP SYSTEM(s) inoperable, restore at least one TRIP SYSTEM to OPERABLE status within one hour or be in at least STARTUP within the next 6 hours.

4.2 - SURVEILLANCE REQUIREMENTS

3.2 - LIMITING CONDITIONS FOR OPERATION

D. Reactor Core Isolation Cooling Actuation

The reactor core isolation cooling (RCIC) system actuation instrumentation CHANNEL(s) shown in Table 3.2.D-1 shall be OPERABLE with their trip setpoints set consistent with the values shown in the Trip Setpoint column.

APPLICABILITY:

OPERATIONAL MODE(s) 1, 2 and 3 with the reactor steam dome pressure >150 psig.

ACTION:

1. With a RCIC system actuation instrumentation CHANNEL trip setpoint less conservative than the value shown in the Trip Setpoint column of Table 3.2.D-1, declare the CHANNEL inoperable until the CHANNEL is restored to OPERABLE status with its trip setpoint adjusted consistent with the Trip Setpoint value.
2. With one or more RCIC system actuation instrumentation CHANNEL(s) inoperable, take the ACTION required by Table 3.2.D-1.

4.2 - SURVEILLANCE REQUIREMENTS

D. Reactor Core Isolation Cooling Actuation

1. Each RCIC system actuation instrumentation CHANNEL shall be demonstrated OPERABLE by the performance of the CHANNEL CHECK, CHANNEL FUNCTIONAL TEST and CHANNEL CALIBRATION operations at the frequencies shown in Table 4.2.D-1.
2. LOGIC SYSTEM FUNCTIONAL TEST(s) of all CHANNEL(s) shall be performed at least once per 18 months.

TABLE 3.2.D-1 (Continued)

REACTOR CORE ISOLATION COOLING ACTUATION INSTRUMENTATION

ACTION

ACTION 40 - With the number of OPERABLE CHANNEL(s) less than required by the Minimum CHANNEL(s) per TRIP SYSTEM requirement:

a. With one CHANNEL inoperable, place the inoperable CHANNEL in the tripped condition within one hour or declare the RCIC system inoperable.

b. With more than one CHANNEL inoperable, declare the RCIC system inoperable.

ACTION 41 - With the number of OPERABLE CHANNEL(s) less than required by the Minimum CHANNEL(s) per TRIP SYSTEM requirement, declare the RCIC system inoperable.

ACTION 42 - With the number of OPERABLE CHANNEL(s) less than required by the Minimum CHANNEL(s) per TRIP SYSTEM requirement, place at least one inoperable CHANNEL in the tripped condition within one hour or declare the RCIC system inoperable.

ACTION 43 - With the number of OPERABLE CHANNEL(s) less than required by the Minimum OPERABLE CHANNEL(s) per TRIP SYSTEM requirement, restore the inoperable CHANNEL to OPERABLE status within 8 hours or declare the RCIC system inoperable.

TABLE 3.2.E-1

CONTROL ROD BLOCK INSTRUMENTATION

Functional Unit	Trip Setpoint	Minimum CHANNEL(s) per Trip Function[i]	Applicable OPERATIONAL MODE(s)	ACTION
1. ROD BLOCK MONITORS[a]				
a. Upscale	As specified in COLR	2	1[e]	50
b. Inoperative	NA	2	1[e]	50
c. Downscale	≥3/125 of full scale	2	1[e]	50
2. AVERAGE POWER RANGE MONITORS				
a. Flow Biased Neutron Flux - High				
1. Dual Recirculation Loop Operation	$\leq(0.58W+50)$[g]	4	1	51
2. Single Recirculation Loop Operation	$\leq(0.58W+46.5)$[g]	4	1	51
b. Inoperative	NA	4	1, 2, 5[h]	51
c. Downscale	≥3/125 of full scale	4	1	51
d. Startup Neutron Flux - High	≤12/125 of full scale	4	2, 5[h]	51

TABLE 3.2.E-1 (Continued)

CONTROL ROD BLOCK INSTRUMENTATION

Functional Unit	Trip Setpoint	Minimum CHANNEL(s) per Trip Function[i]	Applicable OPERATIONAL MODE(s)	ACTION
3. SOURCE RANGE MONITORS				
a. Detector not full in[b]	NA	3 2	2 5	51 51
b. Upscale[c]	$\leq 1 \times 10^5$ cps	3 2	2 5	51 51
c. Inoperative[c]	NA	3 2	2 5	51 51
4. INTERMEDIATE RANGE MONITORS				
a. Detector not full in[d]	NA	6	2, 5	51
b. Upscale	≤108/125 of full scale	6	2, 5	51
c. Inoperative	NA	6	2, 5	51
d. Downscale[d]	≥3/125 of full scale	6	2, 5	51

TABLE 3.2.E-1 (Continued)

CONTROL ROD BLOCK INSTRUMENTATION

Functional Unit	Trip Setpoint	Minimum CHANNEL(s) per Trip Function[i]	Applicable OPERATIONAL MODE(s)	ACTION
5. SCRAM DISCHARGE VOLUME (SDV)				
a. Water Level - High	≤25 gal	1 per bank	1, 2, 5[f]	52
b. SDV Switch in Bypass	NA	1	5[f]	52

TABLE 3.2.E-1 (Continued)

CONTROL ROD BLOCK INSTRUMENTATION

TABLE NOTATION

(a) The RBM shall be automatically bypassed when a peripheral control rod is selected or the reference APRM channel indicates less than 30% of RATED THERMAL POWER.

(b) This function shall be automatically bypassed if detector count rate is >100 cps or the IRM channels are on range 3 or higher.

(c) This function shall be automatically bypassed when the associated IRM channels are on range 8 or higher.

(d) This function shall be automatically bypassed when the IRM channels are on range 1.

(e) With THERMAL POWER ≥30% of RATED THERMAL POWER.

(f) With more than one control rod withdrawn. Not applicable to control rods removed per Specification 3.10.I or 3.10.J.

(g) The Average Power Range Monitor rod block function is varied as a function of recirculation loop flow (W). The trip setting of this function must be maintained in accordance with Specification 3.11.B. W is equal to the percentage of the drive flow required to produce a rated core flow of 98 x 10^6 lbs/hr.

(h) Required to be OPERABLE only during SHUTDOWN MARGIN demonstrations performed per Specification 3.12.B.

(i) A CHANNEL may be placed in an inoperable status for up to 2 hours for required surveillance without placing the CHANNEL in the tripped condition provided the Functional Unit maintains control rod block capability.

TABLE 4.2.E-1

CONTROL ROD BLOCK INSTRUMENTATION SURVEILLANCE REQUIREMENTS

Functional Unit	CHANNEL CHECK	CHANNEL FUNCTIONAL TEST	CHANNEL CALIBRATION[a]	Applicable OPERATIONAL MODE(s)
1. ROD BLOCK MONITORS				
a. Upscale	NA	S/U[b,c], M[c]	Q	1[d]
b. Inoperative	NA	S/U[b,c], M[c]	NA	1[d]
c. Downscale	NA	S/U[b,c], M[c]	Q	1[d]
2. AVERAGE POWER RANGE MONITORS				
a. Flow Biased Neutron Flux - High				
1. Dual Recirculation Loop Operation	NA	S/U[b], M	SA	1
2. Single Recirculation Loop Operation	NA	S/U[b], M	SA	1
b. Inoperative	NA	S/U[b], M	NA	1, 2, 5[j]
c. Downscale	NA	S/U[b], M	SA	1
d. Startup Neutron Flux - High	NA	S/U[b], M	SA[k]	2, 5[j]
3. SOURCE RANGE MONITORS				
a. Detector not full in[f]	NA	S/U[b], W	E	2[i], 5
b. Upscale[g]	NA	S/U[b], W	E	2[i], 5
c. Inoperative[g]	NA	S/U[b], W	NA	2[i], 5

TABLE 4.2.E-1 (Continued)

CONTROL ROD BLOCK INSTRUMENTATION
SURVEILLANCE REQUIREMENTS

Functional Unit	CHANNEL CHECK	CHANNEL FUNCTIONAL TEST	CHANNEL CALIBRATION[a]	Applicable OPERATIONAL MODE(s)
4. INTERMEDIATE RANGE MONITORS				
a. Detector not full in[h]	NA	S/U[b], W	E	2[i], 5
b. Upscale	NA	S/U[b], W	E[k]	2[i], 5
c. Inoperative	NA	S/U[b], W	NA	2[i], 5
d. Downscale[h]	NA	S/U[b], W	E[k]	2[i], 5
5. SCRAM DISCHARGE VOLUME (SDV)				
a. Water Level - High	NA	Q	NA	1, 2, 5[e]
b. SDV Switch in Bypass	NA	E	NA	5[e]

TABLE 4.2.E-1 (Continued)

CONTROL ROD BLOCK INSTRUMENTATION
SURVEILLANCE REQUIREMENTS

TABLE NOTATION

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

(b) Within 7 days prior to startup.

(c) Includes reactor manual control "relay select matrix" system input.

(d) With THERMAL POWER ≥30% of RATED THERMAL POWER.

(e) With more than one control rod withdrawn. Not applicable to control rods removed per Specification 3.10.I or 3.10.J.

(f) This function shall be automatically bypassed if detector count rate is >100 cps or the IRM channels are on range 3 or higher.

(g) This function shall be automatically bypassed when the associated IRM channels are on range 8 or higher.

(h) This function shall be automatically bypassed when the IRM channels are on range 1.

(i) The provisions of Specification 4.0.D are not applicable to the CHANNEL FUNCTIONAL TEST and CHANNEL CALIBRATION surveillances for entry into the applicable OPERATIONAL MODE(s) from OPERATIONAL MODE 1 provided the surveillances are performed within 12 hours after such entry.

(j) Required to be OPERABLE only during SHUTDOWN MARGIN demonstrations performed per Specification 3.12.B.

(k) The CHANNEL CALIBRATION surveillance requirements shall be performed within 12 hours upon each entry into any OPERATIONAL MODE(s) from OPERATIONAL MODE 1 if not performed within the previous seven days.

TABLE 3.2.F-1

ACCIDENT MONITORING INSTRUMENTATION

INSTRUMENTATION	Required CHANNEL(s)	Minimum CHANNEL(s)	Applicable OPERATIONAL MODE(s)	ACTION
1. Reactor Vessel Pressure	2	1	1, 2	60
2. Reactor Vessel Water Level	2	1	1, 2	60
3 Torus Water Level	2	1	1, 2	60
4. Torus Water Temperature	2	1	1, 2	60
5. Drywell Pressure - Wide Range	2	1	1, 2	60
6. Drywell Pressure - Narrow Range	2	1	1, 2	60
7. Drywell Air Temperature	2	1	1, 2	60
8. Drywell Oxygen Concentration - Analyzer and Monitor	2	1	1, 2	62
9. Drywell Hydrogen Concentration - Analyzer and Monitor	2	1	1, 2	62
10. Safety & Relief Valve Position Indicators - Acoustic & Temperature	2/valve (1 each)	1/valve	1, 2	63
11. (Source Range) Neutron Monitors	2	2	1, 2	60
12. Drywell Radiation Monitors	2	2	1, 2, 3	61
13. Torus Air Temperature	2	1	1, 2	60
14. Torus Pressure	2[a]	1	1, 2	60

a This function is shared with Drywell Pressure-Wide Range and Drywell Pressure-Narrow Range.

TABLE 3.2.F-1 (Continued)

ACCIDENT MONITORING INSTRUMENTATION

ACTION

ACTION 60 -

a. With the number of OPERABLE accident monitoring instrumentation CHANNEL(s) less than the Required CHANNEL(s) shown in Table 3.2.F-1, restore the inoperable CHANNEL(s) to OPERABLE status within 30 days or be in at least HOT SHUTDOWN within the next 12 hours.

b. With the number of OPERABLE accident monitoring instrumentation CHANNEL(s) less than the Minimum CHANNEL(s) shown in Table 3.2.F-1, restore the inoperable CHANNEL(s) to OPERABLE status within 48 hours or be in at least HOT SHUTDOWN within the next 12 hours.

ACTION 61- With the number of OPERABLE accident monitoring instrumentation CHANNEL(s) less than the Minimum CHANNEL(s) shown in Table 3.2.F-1, initiate the preplanned alternate method of monitoring the appropriate parameter(s) within 72 hours, and:

a. Either restore the inoperable CHANNEL(s) to OPERABLE status within 7 days of the event, or

b. Prepare and submit a Special Report to the Commission pursuant to Specification 6.9.B within 30 days following the event outlining the action taken, the cause of the inoperability and the plans and schedule for restoring the system to OPERABLE status.

ACTION 62-

a. With the number of OPERABLE accident monitoring instrumentation CHANNEL(s) one less than the Required CHANNEL(s) shown in Table 3.2.F-1, restore the inoperable CHANNEL(s) to OPERABLE status within 30 days or be in at least HOT SHUTDOWN within the next 12 hours.

b. With the number of OPERABLE accident monitoring instrumentation CHANNEL(s) less than the Minimum CHANNEL(s) shown in Table 3.2.F-1; and provided the high radiation sampling system (HRSS) combustible gas monitoring capability for the drywell is OPERABLE; restore the inoperable CHANNEL(s) to OPERABLE status within 30 days or be in at least HOT SHUTDOWN within the next 12 hours.

c. With the number of OPERABLE accident monitoring instrumentation CHANNEL(s) less than the Minimum CHANNEL(s) shown in Table 3.2.F-1; and the HRSS combustible gas monitoring capability for the drywell inoperable; restore at least one inoperable CHANNEL to OPERABLE status within 7 days or be in at least HOT SHUTDOWN within the next 12 hours.

TABLE 3.2.F-1 (Continued)

ACCIDENT MONITORING INSTRUMENTATION

ACTION 63 -

a. With the number of OPERABLE accident monitoring instrumentation CHANNEL(s) less than the Required CHANNEL(s) shown in Table 3.2.F-1, restore the inoperable CHANNEL(s) to OPERABLE status prior to startup from a COLD SHUTDOWN of longer than 72 hours.

b. With the number of OPERABLE accident monitoring instrumentation CHANNEL(s) less than the Minimum CHANNEL(s) shown in Table 3.2.F-1, restore at least one of the inoperable CHANNEL(s) to OPERABLE status within 30 days or be in at least HOT SHUTDOWN within the next 12 hours.

TABLE 4.2.F-1

ACCIDENT MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

INSTRUMENTATION	CHANNEL CHECK	CHANNEL CALIBRATION	Applicable OPERATIONAL MODE(s)
1. Reactor Vessel Pressure	M	E	1, 2
2. Reactor Vessel Water Level	M	E	1, 2
3 Torus Water Level	M	E	1, 2
4. Torus Water Temperature	M	E	1, 2
5. Drywell Pressure - Wide Range	M	E	1, 2
6. Drywell Pressure - Narrow Range	M	E	1, 2
7. Drywell Air Temperature	M	E	1, 2
8. Drywell Hydrogen/Oxygen Concentration - Analyzer and Monitor	M	Q	1, 2
9. Safety & Relief Valve Position Indicators - Acoustic & Temperature	M	E	1, 2
10. (Source Range) Neutron Monitors	M	E[b]	1, 2
11. Drywell Radiation Monitors	M	E[a]	1, 2, 3
12. Torus Air Temperature	M	E	1, 2
13. Torus Pressure	M	E	1, 2

TABLE 4.2.F-1 (Continued)

ACCIDENT MONITORING INSTRUMENTATION
SURVEILLANCE REQUIREMENTS

TABLE NOTATION

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

(b) Neutron detectors may be excluded from the CHANNEL CALIBRATION.

3.2 - LIMITING CONDITIONS FOR OPERATION

G. Source Range Monitoring

At least the following source range monitor (SRM) channels shall be OPERABLE:

a. In OPERATIONAL MODE 2[a], three.

b. In OPERATIONAL MODE 3 and 4, two.

APPLICABILITY:

OPERATIONAL MODE(s) 2[a], 3, and 4.

ACTION:

1. In OPERATIONAL MODE 2[a] with one of the above required source range monitor CHANNEL(s) inoperable, at least 3 source range monitor CHANNEL(s) shall be restored to OPERABLE status within 4 hours or the reactor shall be in at least HOT SHUTDOWN within the next 12 hours.

2. In OPERATIONAL MODE(s) 3 or 4 with one or more of the above required source range monitor CHANNEL(s) inoperable, verify all insertable control rods to be fully inserted in the core and lock the reactor mode switch in the Shutdown position within one hour.

4.2 - SURVEILLANCE REQUIREMENTS

G. Source Range Monitoring

Each of the required source range monitor CHANNEL(s) shall be demonstrated OPERABLE by:

1. Verifying, prior to withdrawal of the control rods, that the SRM count rate is ≥3 cps with the detector fully inserted.

2. Performance of a CHANNEL CHECK at least once per:

   a. 12 hours in OPERATIONAL MODE 2[a], and

   b. 24 hours in OPERATIONAL MODE(s) 3 or 4.

3. Performance of a CHANNEL FUNCTIONAL TEST:

   a. Within 7 days prior to startup, and

   b. At least once per 31 days[b].

4. Performance of a CHANNEL CALIBRATION[c] at least once per 18 months[b].

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a With IRM's on range 2 or below.

b The provisions of Specification 4.0.D are not applicable for entry into the applicable OPERATIONAL MODE(s) from OPERATIONAL MODE 1, provided the surveillance is performed within 12 hours after such entry.

c Neutron detectors may be excluded from the CHANNEL CALIBRATION.

3.2 - LIMITING CONDITIONS FOR OPERATION

H. Explosive Gas Monitoring

The explosive gas monitoring instrumentation CHANNEL(s) shown in Table 3.2.H-1 shall be OPERABLE with their alarm/trip setpoints set to ensure that the limits of Specification 3.8.H are not exceeded.

APPLICABILITY:

During offgas holdup system operation.

ACTION:

1. With an explosive gas monitoring instrumentation CHANNEL alarm/trip setpoint less conservative than required by the above specification, declare the CHANNEL inoperable and take the ACTION shown in Table 3.2.H-1.
2. With less than the minimum number of explosive gas monitoring instrumentation CHANNEL(s) OPERABLE, take the ACTION shown in Table 3.2.H-1. Restore the inoperable instrumentation to OPERABLE status within 30 days and, if unsuccessful, prepare and submit a Special Report to the Commision pursuant to Specification 6.9.B to explain why this inoperability was not corrected in a timely manner.
3. The provisions of Specification 3.0.C are not applicable.

4.2 - SURVEILLANCE REQUIREMENTS

H. Explosive Gas Monitoring

Each explosive gas monitoring instrumentation CHANNEL shall be demonstrated OPERABLE by the performance of the CHANNEL CHECK, CHANNEL FUNCTIONAL TEST and CHANNEL CALIBRATION operations at the frequencies shown in Table 4.2.H-1.

TABLE 4.2.H-1

EXPLOSIVE GAS MONITORING INSTRUMENTATION
SURVEILLANCE REQUIREMENTS

Functional Unit	CHANNEL CHECK	CHANNEL FUNCTIONAL TEST	CHANNEL CALIBRATION
MAIN CONDENSER OFFGAS TREATMENT SYSTEM EXPLOSIVE GAS MONITORING SYSTEM			
Hydrogen Monitor	D	M	Q

3.2 - LIMITING CONDITIONS FOR OPERATION

I. Suppression Chamber and Drywell Spray Actuation

The Suppression Chamber and Drywell Spray Actuation instumentation CHANNEL(s) shown in Table 3.2.I-1 shall be OPERABLE with their trip setpoints set consistent with the values shown in the Trip Setpoint column of Table 3.2.I-1.

APPLICABILITY:

OPERATIONAL MODE(s) 1, 2 & 3.

ACTION:

With a Suppression Chamber and Drywell Spray Actuation instrumentation CHANNEL trip setpoint less conservative than the value shown in the Trip Setpoint column of Table 3.2.I-1, declare the CHANNEL inoperable and take the ACTION shown in Table 3.2.I-1.

4.2 - SURVEILLANCE REQUIREMENTS

I. Suppression Chamber and Drywell Spray Actuation

1. Each Suppression Chamber and Drywell Spray Actuation instrumentation CHANNEL shall be demonstrated OPERABLE by the performance of the CHANNEL CHECK, CHANNEL FUNCTIONAL TEST and CHANNEL CALIBRATION operations at the frequencies shown in Table 4.2.I-1.

2. LOGIC SYSTEM FUNCTIONAL TEST(s) of all CHANNEL(s) shall be performed at least once per 18 months.

TABLE 3.2.I-1

SUPPRESSION CHAMBER AND DRYWELL SPRAY ACTUATION INSTRUMENTATION

Functional Unit	Trip Setpoint[a]	Minimum CHANNEL(s) per TRIP SYSTEM[c]	ACTION
1. Drywell Pressure - High (Permissive)	0.5≤ ρ ≤1.5 psig	2	80
2. Reactor Vessel Water Level - Low (Permissive)	≥ -48 inches	1	80

ACTION

ACTION 80 -

a. With the number of OPERABLE CHANNEL(s) less than required by the Minimum OPERABLE CHANNEL(s) per TRIP SYSTEM requirement for one TRIP SYSTEM, place at least one inoperable CHANNEL in the tripped condition[b] within one hour or declare the Suppression Chamber and Drywell Spray Actuation mode of the Residual Heat Removal system inoperable.

b. With the number of OPERABLE CHANNEL(s) less than required by the Minimum OPERABLE CHANNEL(s) per TRIP SYSTEM requirement for both TRIP SYSTEM(s), declare the Suppression Chamber and Drywell Spray Actuation mode of the Residual Heat Removal system inoperable.

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a Reactor vessel water level settings are expressed in inches above the top of active fuel (which is 360 inches above vessel zero).

b If an instrument is inoperable, it shall be placed (or simulated) in a tripped condition so that it will not prevent a containment spray.

c A CHANNEL may be placed in an inoperable status for up to 2 hours for required surveillance without placing the CHANNEL in the tripped condition provided the Functional Unit maintains Suppression Chamber and Drywell Spray Actuation capability.

3.2 - LIMITING CONDITIONS FOR OPERATION

J. Feedwater Pump Trip

The feedwater pump trip instrumentation CHANNEL(s) shown in Table 3.2.J-1 shall be OPERABLE with their trip setpoints set consistent with the values shown in the Trip Setpoint column of Table 3.2.J-1.

APPLICABILITY:

OPERATIONAL MODE 1.

ACTION:

With a feedwater pump trip instrumentation CHANNEL trip setpoint less conservative than the value shown in the Trip Setpoint column of Table 3.2.J-1, declare the CHANNEL inoperable and take the ACTION shown in Table 3.2.J-1

4.2 - SURVEILLANCE REQUIREMENTS

J. Feedwater Pump Trip

1. Each feedwater pump trip instrumentation CHANNEL shall be demonstrated OPERABLE by the performance of the CHANNEL CHECK, CHANNEL FUNCTIONAL TEST and CHANNEL CALIBRATION operations at the frequencies shown in Table 4.2.J-1.

2. LOGIC SYSTEM FUNCTIONAL TEST(s) of all CHANNEL(s) shall be performed at least once per 18 months.

TABLE 3.2.J-1

FEEDWATER PUMP TRIP INSTRUMENTATION

Functional Unit	Trip Setpoint[a]	Minimum CHANNEL(s)[b]	ACTION
Reactor Vessel Water Level -High	≤201 inches	2	90

ACTION

ACTION 90 -
a. With the number of OPERABLE CHANNEL(s) one less than required by the Minimum CHANNEL(s) requirement, restore the inoperable CHANNEL to OPERABLE status within 7 days or place the inoperable CHANNEL in the tripped condition within the next 8 hours.

b. With the number of OPERABLE CHANNEL(s) two less than required by the Minimum CHANNEL(s) requirement, restore at least one of the inoperable CHANNEL(s) to OPERABLE status within 72 hours or be in at least STARTUP within the next 8 hours.

a Reactor vessel water level settings are expressed in inches above the top of active fuel (which is 360 inches above vessel zero).

b A CHANNEL may be placed in an inoperable status for up to 2 hours for required surveillance without placing the CHANNEL in the tripped condition.

BASES

3/4.2.E Control Rod Block Actuation Instrumentation

The control rod block functions are provided to prevent excessive control rod withdrawal so that the MINIMUM CRITICAL POWER RATIO (MCPR) does not go below the MCPR fuel cladding integrity Safety Limit. During shutdown conditions, control rod block instrumentation initiates withdrawal blocks to ensure that all control rods remain inserted to prevent inadvertent criticality.

The trip logic for this function is one-out-of-n; e.g., any trip on one of the six average power range monitors (APRMs), eight intermediate range monitors (IRMs), or four source range monitors (SRMs), will result in a rod block. The minimum instrument CHANNEL requirements assure sufficient instrumentation to assure that the single failure criterion is met. The minimum instrument CHANNEL requirements for the rod block monitor may be reduced by one for a short period of time to allow for maintenance, testing, or calibration.

The APRM rod block function is flow-biased and prevents a significant reduction in MCPR, especially during operation at reduced flow. The APRM provides gross core protection, i.e., limits the gross withdrawal of control rods in the normal withdrawal sequence.

In the REFUEL MODE during SHUTDOWN MARGIN demonstrations and the STARTUP/HOT STANDBY OPERATIONAL MODE, the APRM rod block function setpoint is significantly reduced to provide the same type of protection in the REFUEL and STARTUP/HOT STANDBY OPERATIONAL MODE(s) as the APRM flow-biased rod block does in the RUN OPERATIONAL MODE, i.e., prevents control rod withdrawal before a scram is reached.

The rod block monitor (RBM) function provides local protection of the core, i.e., the prevention of transition boiling in a local region of the core for a single rod withdrawal error. The trip setting is flow-biased. At low power, the worst-case withdrawal of a single control rod without rod block action will not violate the fuel cladding integrity Safety Limit. Thus the RBM rod block function is not required below the specified power level. The worst-case single control rod withdrawal error is analyzed for each reload to assure that, with the specific trip settings, rod withdrawal is blocked before the MCPR reaches the fuel cladding integrity Safety Limit.

The IRM rod block function provides local as well as gross core protection. The scaling arrangement is such that the trip setting is less than a factor of ten above the indicated level. Analysis of the worst-case accident results in rod block action before MCPR approaches the MCPR fuel cladding integrity Safety Limit.

A downscale indication on an APRM is an indication that the instrument has failed or is not sensitive enough. In either case, the instrument will not respond to changes in control rod motion, and the control rod motion is thus prevented.

The SRM rod blocks of low count rate and the detector not fully inserted assure that the SRMs are not withdrawn from the core prior to commencing rod withdrawal for startup. The scram

BASES

discharge volume, high water level rod block provides annunciation for operator action. The alarm setpoint has been selected to provide adequate time to allow for the determination of the cause for the level increase and corrective action prior to automatic scram initiation.

3/4.2.F Accident Monitoring Instrumentation

Instrumentation is provided to monitor sufficient accident conditions to adequately assess important variables and provide operators with necessary information to complete the appropriate mitigation actions. OPERABILITY of the instrumentation listed provides adequate monitoring of the containment following a loss-of-coolant accident. Information from this instrumentation will provide the operator with a detailed knowledge of the conditions resulting from the accident; based on this information, the operator can make logical decisions regarding post accident recovery. Allowable outage times are based on diverse instrumentation availability for guiding the operator should an accident occur, and on the low probability of an instrument being out-of-service concurrent with an accident. As noted in the surveillance requirements, the instrumentation CHANNEL(s) associated with Torus Pressure provides a dual function and is shared in common with the Drywell Pressure (Narrow and Wide Ranges) instrumentation. This instrumentation is identified in response to Generic Letter 82-33 and the associated NRC Safety Evaluation Report, and some instrumentation is included in accordance with the response to Generic Letter 83-36.

3/4.2.G Source Range Monitoring Instrumentation

The source range monitors (SRM) provide the operator with the status of the neutron flux in the core at very low power levels during startup and shutdown. The consequences of reactivity accidents are functions of the initial neutron flux. Therefore, the requirements for a minimum count rate assures that any transient, should it occur, begins at or above the initial value used in the analyses of transients from cold conditions. Two OPERABLE SRM CHANNEL(s) are adequate to monitor the approach to criticality using homogeneous patterns of scattered control rod withdrawal. Three OPERABLE SRMs provide an added conservatism. When the intermediate range monitors are on scale, adequate information is available without the SRMs and they can be retracted.

3/4.2.H Explosive Gas Monitoring Instrumentation

Instrumentation is provided to monitor the concentrations of potentially explosive mixtures in the off-gas holdup system to prevent a possible uncontrolled release via this pathway. This instrumentation is included in accordance with Generic Letter 89-01.

BASES

3/4.2.I Suppression Chamber and Drywell Spray Actuation Instrumentation

Instrumentation is provided to monitor the parameters which are necessary to permit initiation of the containment cooling mode of the residual heat removal system to condense steam in the containment atmosphere. The spray mode does not significantly affect the rise of drywell pressure following a loss of coolant accident, but does result in quicker depressurization following completion of the blowdown.

3/4.2.J Feedwater Trip System Actuation

The feedwater trip system actuation instrumentation is designed to detect a potential failure of the feedwater control system which causes excessive feedwater flow. If undetected, this would lead to reactor vessel water carryover into the main steam lines and to the main turbine. This instrumentation is included in response to Generic Letter 89-19.

3/4.2.K Toxic Gas Monitoring

Toxic gas monitoring instrumentation is provided in or near the control room ventilation system intakes to allow prompt detection and the necessary protective actions to be initiated. Isolation from high toxic chemical concentration has been added to the station design as a result of the "Control Room Habitability Study" submitted to the NRC in December 1981 in response to NUREG-0737 Item III D.3.4. As explained in Section 3 of this study, ammonia, chlorine, and sulphur dioxide detection capability has been provided. In a report generated by Sargent and Lundy in April 1991, justification was provided to delete the chlorine and sulphur dioxide detectors from the plant. The setpoints chosen for the control room ventilation isolation are based on early detection in the outside air supply at the odor threshold, so that the toxic chemical will not achieve toxicity limit concentrations in the Control Room.

3.3 - LIMITING CONDITIONS FOR OPERATION

A. SHUTDOWN MARGIN (SDM)

The SHUTDOWN MARGIN (SDM) shall be equal to or greater than:

1. 0.38% Δk/k with the highest worth control rod analytically determined, or

2. 0.28% Δk/k with the highest worth control rod determined by test.

APPLICABILITY:

OPERATIONAL MODE(s) 1, 2, 3, 4, and 5.

ACTION:

With the SHUTDOWN MARGIN less than specified:

1. In OPERATIONAL MODE 1 or 2, restore the required SHUTDOWN MARGIN within 6 hours or be in at least HOT SHUTDOWN within the next 12 hours.

2. In OPERATIONAL MODE 3 or 4, immediately verify all insertable control rods to be fully inserted and suspend all activities that could reduce the SHUTDOWN MARGIN. In OPERATIONAL MODE 4, establish SECONDARY CONTAINMENT INTEGRITY within 8 hours.

3. In OPERATIONAL MODE 5, suspend CORE ALTERATION(s) and other activities that could reduce the SHUTDOWN MARGIN and fully insert all insertable control rods within 1 hour. Establish SECONDARY CONTAINMENT INTEGRITY within 8 hours.

4.3 - SURVEILLANCE REQUIREMENTS

A. SHUTDOWN MARGIN

The SHUTDOWN MARGIN shall be determined to be equal to or greater than that specified at any time during the operating cycle:

1. By demonstration, prior to or during the first startup after each refueling outage.

2. Within 24 hours after detection of a withdrawn control rod that is immovable, as a result of excessive friction or mechanical interference, or known to be unscrammable. The required SHUTDOWN MARGIN shall be verified acceptable with an increased allowance for the withdrawn worth of the immovable or unscrammable control rod.

3. By calculation, prior to each fuel movement during the fuel loading sequence.

3.3 - LIMITING CONDITIONS FOR OPERATION

C. Control Rod OPERABILITY

All control rods shall be OPERABLE.

APPLICABILITY:

OPERATIONAL MODE(s) 1 and 2.

ACTION:

1. With one control rod inoperable due to being immovable as a result of excessive friction or mechanical interference, or known to be unscrammable:

   a. Within one hour:

      1) Verify that the inoperable control rod, if withdrawn, is separated from all other inoperable withdrawn control rods by at least two control cells in all directions.

      2) Disarm the associated directional control valves[a] either:

         a) Electrically, or

         b) Hydraulically by closing the drive water and exhaust water isolation valves.

   b. With the provisions of ACTION 1.a above not met, be in at least HOT SHUTDOWN within the next 12 hours.

4.3 - SURVEILLANCE REQUIREMENTS

C. Control Rod OPERABILITY

1. When above the low power setpoint of the RWM, all withdrawn control rods not required to have their directional control valves disarmed electrically or hydraulically shall be demonstrated OPERABLE by moving each control rod at least one notch:

   a. At least once per 7 days, and

   b. Within 24 hours when any control rod is immovable as a result of excessive friction or mechanical interference, or known to be unscrammable.

2. All control rods shall be demonstrated OPERABLE by performance of Surveillance Requirements 4.3.D, 4.3.F, 4.3.G, 4.3.H and 4.3.I.

---

a May be rearmed intermittently, under administrative control, to permit testing associated with restoring the control rod to OPERABLE status.

3.3 - LIMITING CONDITIONS FOR OPERATION

D. Maximum Scram Insertion Times

The maximum scram insertion time of each control rod from the fully withdrawn position to 90% insertion, based on de-energization of the scram pilot valve solenoids as time zero, shall not exceed 7 seconds.

APPLICABILITY:

OPERATIONAL MODE(s) 1 and 2.

ACTION:

With the maximum scram insertion time of one or more control rods exceeding 7 seconds:

1. Declare the control rod(s) exceeding the above maximum scram insertion time inoperable, and
2. When operation is continued with three or more control rods with maximum scram insertion times in excess of 7 seconds, perform Surveillance Requirement 4.3.D.3 at least once per 60 days of power operation.

With the provisions of the ACTION above not met, be in at least HOT SHUTDOWN within 12 hours.

4.3 - SURVEILLANCE REQUIREMENTS

D. Maximum Scram Insertion Times

The maximum scram insertion time of the control rods shall be demonstrated through measurement with reactor coolant pressure greater than 800 psig and, during single control rod scram time tests, with the control rod drive pumps isolated from the accumulators:

1. For all control rods prior to THERMAL POWER exceeding 40% of RATED THERMAL POWER:
   a. following CORE ALTERATION(s), or
   b. after a reactor shutdown that is greater than 120 days,
2. For specifically affected individual control rods[a] following maintenance on or modification to the control rod or control rod drive system which could affect the scram insertion time of those specific control rods, and
3. For at least 10% of the control rods, on a rotating basis, at least once per 120 days of power operation.

a The provisions of Specification 4.0.D are not applicable provided this surveillance is conducted prior to exceeding 40% of RATED THERMAL POWER.

3.3 - LIMITING CONDITIONS FOR OPERATION

H. Control Rod Drive Coupling

All control rods shall be coupled to their drive mechanisms.

APPLICABILITY:

OPERATIONAL MODE(s) 1, 2, and 5[a].

ACTION:

1. In OPERATIONAL MODE 1 or 2 with one control rod not coupled to its associated drive mechanism, within 2 hours:

   a. If permitted by the RWM, insert the control rod drive mechanism to accomplish recoupling and verify recoupling by withdrawing the control rod, and:

      1) Observing any indicated response of the nuclear instrumentation, and

      2) Demonstrating that the control rod will not go to the overtravel position.

   b. If not permitted by the RWM or, if recoupling is not accomplished in accordance with ACTION 1.a above, then declare the control rod inoperable, fully insert the control rod and disarm the associated directional control valves[b] either:

      1) Electrically, or

4.3 - SURVEILLANCE REQUIREMENTS

H. Control Rod Drive Coupling

Each affected control rod shall be demonstrated to be coupled to its drive mechanism by verifying that the control rod drive does not go to the overtravel position:

1. Deleted.

2. Anytime the control rod is withdrawn to the "Full out" position, and

3. Following maintenance on or modification to the control rod or control rod drive system which could have affected the control rod drive coupling integrity.

---

a In OPERATIONAL MODE 5, this Specification is applicable for withdrawn control rods and is not applicable to control rods removed per Specification 3.10.I or 3.10.J.

b May be rearmed intermittently, under administrative control, to permit testing associated with restoring the control rod to OPERABLE status.

3.3 - LIMITING CONDITIONS FOR OPERATION

I. Control Rod Position Indication System

All control rod position indicators shall be OPERABLE.

APPLICABILITY:

OPERATIONAL MODE(s) 1, 2, and 5[a].

ACTION:

1. In OPERATIONAL MODE 1 or 2 with one or more control rod position indicators inoperable, within one hour either:

    a. Determine the position of the control rod by an alternate method, or

    b. Move the control rod to a position with an OPERABLE position indicator, or

    c. Declare the control rod inoperable, fully insert the inoperable withdrawn control rod(s), and disarm the associated directional control valves[b] either:

        1) Electrically, or

        2) Hydraulically by closing the drive water and exhaust water isolation valves.

4.3 - SURVEILLANCE REQUIREMENTS

I. Control Rod Position Indication System

The control rod position indication system shall be determined OPERABLE by verifying:

1. At least once per 24 hours that the position of each control rod is indicated.

2. That the indicated control rod position changes during the movement of the control rod drive when performing Surveillance Requirement 4.3.C.1.

3. Deleted.

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a In OPERATIONAL MODE 5, this Specification is applicable for withdrawn control rods and is not applicable to control rods removed per Specification 3.10.I or 3.10.J.

b May be reamed intermittently, under administrative control, to permit testing associated with restoring the control rod(s) to OPERABLE status.

3.3 - LIMITING CONDITIONS FOR OPERATION

L. Rod Worth Minimizer (RWM)

The rod worth minimizer (RWM) shall be OPERABLE.

APPLICABILITY:

OPERATIONAL MODE(s) 1 and 2[a], when thermal power is less than or equal to 10% of RATED THERMAL POWER.

ACTION:

With the RWM inoperable, verify control rod movement and compliance with the prescribed control rod pattern by a second licensed operator or technically qualified individual who is present at the reactor control console. Otherwise, control rod movement may be made only by actuating the manual scram or placing the reactor mode switch in the Shutdown position.

4.3 - SURVEILLANCE REQUIREMENTS

L. Rod Worth Minimizer (RWM)

The RWM shall be demonstrated OPERABLE:

1. By verifying that the control rod patterns and sequence input to the RWM computer are correctly loaded following any loading of the program into the computer.
2. In OPERATIONAL MODE 2 within 8 hours prior to withdrawal of control rods for the purpose of making the reactor critical:
   a. by verifying proper indication of the selection error of at least one out-of-sequence control rod.
   b. by verifying the rod block function.
3. In OPERATIONAL MODE 1 prior to reducing thermal power below 10% of RATED THERMAL POWER:
   a. by verifying proper indication of the selection error of at least one out-of-sequence control rod.
   b. by verifying the rod block function.

---

a Entry into OPERATIONAL MODE 2 and withdrawal of selected control rods is permitted for the purpose of determining the OPERABILITY of the RWM prior to withdrawal of control rods for the purpose of bringing the reactor to criticality.

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3/4.3.A SHUTDOWN MARGIN

A sufficient SHUTDOWN MARGIN ensures that 1) the reactor can be made subcritical from all operating conditions, 2) the reactivity transients associated with postulated accident conditions are controllable within acceptable limits, and 3) the reactor will be maintained sufficiently subcritical to preclude inadvertent criticality in the shutdown condition.

The SHUTDOWN MARGIN limitation is a restriction to be applied principally to a new refueling pattern. Satisfaction of the limitation must be determined at the time of loading and must be such that it will apply to the entire subsequent fuel cycle. This determination is provided by core design calculations and administrative control of fuel loading patterns. These procedures include restrictions to allow only those intermediate fuel assembly configurations that have been shown to provide the required SHUTDOWN MARGIN.

Since core reactivity values will vary through core life as a function of fuel depletion and poison burnup, the demonstration of SHUTDOWN MARGIN will be performed during xenon free conditions and adjusted to 68°F to accommodate the current moderator temperature. The generalized form is that the reactivity of the core loading will be limited so the core can be made subcritical by at least R + 0.38% $\Delta k/k$ or R + 0.28% $\Delta k/k$, as appropriate, with the strongest control rod fully withdrawn and all others fully inserted. Two different values are supplied in the Limiting Condition for Operation to provide for the different methods of determination of the highest control rod worth, either analytically or by test. This is due to the reduced uncertainty in the SHUTDOWN MARGIN test when the highest worth control rod is determined by demonstration. When SHUTDOWN MARGIN is determined by calculations not associated with a test, additional margin must be added to the specified SHUTDOWN MARGIN limit to account for uncertainties in the calculation.

The value of R in units of % $\Delta k/k$ is the difference between the calculated beginning-of-life core reactivity, at the beginning of the operating cycle, and the calculated value of maximum core reactivity at any time later in the operating cycle, where it would be greater than at the beginning. The value of R shall include the potential SHUTDOWN MARGIN loss assuming full B_4C settling in all inverted poison tubes present in the core. R must be a positive quantity or zero and a new value of R must be determined for each new fuel cycle.

The value of % $\Delta k/k$ in the above expression is provided as a finite, demonstrable, subcriticality margin. This margin is verified using an in-sequence control rod withdrawal at the beginning-of-life fuel cycle conditions. This assures subcriticality with not only the strongest fully withdrawn but at least an R + 0.28% Δk (or 0.38% Δk) margin beyond this condition. This reactivity characteristic has been a basic assumption in the analysis of plant performance and can be best demonstrated at the time of fuel loading, but the margin must also be determined anytime a control rod is incapable of insertion following a scram signal. Any control rod that is immovable as a result of excessive friction or mechanical interference, or is known to be unscrammable, per Specification 3.3.C, is considered to be incapable of insertion following a scram signal. It is important to note that a control rod can be electrically immovable, but scrammable, and no increase in SHUTDOWN MARGIN is required for these control rods.

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During MODE 5, adequate SDM is required to ensure that the reactor does not reach criticality during control rod withdrawals. An evaluation of each in-vessel fuel movement during fuel loading (including shuffling fuel within the core) is required to ensure adequate SDM is maintained during refueling. This evaluation ensures that the intermediate loading patterns are bounded by the safety analyses for the final core loading pattern. For example, bounding analyses that demonstrate adequate SDM for the most reactive configurations during the refueling may be performed to demonstrate acceptability of the entire fuel movement sequence. These bounding analyses include additional margins to the associated uncertainties. Spiral offload/reload sequences inherently satisfy the SR, provided the fuel assemblies are reloaded in the same configuration analyzed for the new cycle. Removing fuel from the core will always result in an increase in SDM.

3/4.3.B Reactivity Anomalies

During each fuel cycle, excess operating reactivity varies as fuel depletes and as any burnable poison in supplementary control is burned. The magnitude of this excess reactivity may be inferred from the critical rod configuration. As fuel burnup progresses, anomalous behavior in the excess reactivity may be detected by comparison of the critical rod pattern selected base states to the predicted rod inventory at that state. Power operating base conditions provide the most sensitive and directly interpretable data relative to core reactivity. Furthermore, using power operating base conditions permits frequent reactivity comparisons. Requiring a reactivity comparison at the specified frequency assures that a comparison will be made before the core reactivity change exceeds 1% $\Delta k/k$. Deviations in core reactivity greater than 1% $\Delta k/k$ are not expected and require thorough evaluation. A 1% $\Delta k/k$ reactivity limit is considered safe since an insertion of the reactivity into the core would not lead to transients exceeding design conditions of the reactor system.

3/4.3.C Control Rod OPERABILITY

Control rods are the primary reactivity control system for the reactor. In conjunction with the Reactor Protection System, the control rods provide the means for reliable control of reactivity changes to ensure the specified acceptable fuel design limits are not exceeded. This specification, along with others, assures that the performance of the control rods in the event of an accident or transient, meets the assumptions used in the safety analysis. Of primary concern is the trippability of the control rods. Other causes for inoperability are addressed in other Specifications following this one. However, the inability to move a control rod which remains trippable does not prevent the performance of the control rod's safety function.

The specification requires that a rod be taken out-of-service if it cannot be moved with drive pressure. Damage within the control rod drive mechanism could be a generic problem, therefore with a control rod immovable because of excessive friction or mechanical interference, operation of the reactor is limited to a time period which is reasonable to determine the cause of the inoperability and at the same time prevent operation with a large number of inoperable control rods. Control rods that are inoperable due to exceeding allowed scram times, but are movable by

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control rod drive pressure, need not be disarmed electrically if the shutdown margin provisions are met for each position of the affected rod(s).

If the rod is fully inserted and then disarmed electrically or hydraulically, it is in a safe position of maximum contribution to shutdown reactivity. (Note: To disarm the drive electrically, four amphenol-type plug connectors are removed from the drive insert and withdrawal solenoids, rendering the drive immovable. This procedure is equivalent to valving out the drive and is preferred, as drive water cools and minimizes crud accumulation in the drive.). If it is disarmed electrically in a non-fully inserted position, that position shall be consistent with the SHUTDOWN MARGIN limitation stated in Specification 3.3.A. This assures that the core can be shut down at all times with the remaining control rods, assuming the strongest OPERABLE control rod does not insert. The occurrence of more than eight inoperable control rods could be indicative of a generic control rod drive problem which requires prompt investigation and resolution.

In order to reduce the potential for Control Rod Drive (CRD) damage and more specifically, collet housing failure, a program of disassembly and inspection of CRDs is conducted during or after each refueling outage. This program follows the recommendations of General Electric SIL-139 with nondestructive examination results compiled and reported to General Electric on collet housing cracking problems.

The required surveillance intervals are adequate to determine that the rods are OPERABLE and not so frequent as to cause excessive wear on the system components.

<u>3/4.3.D</u> <u>Control Rod Maximum Scram Insertion Times;</u>

<u>3/4.3.E</u> <u>Control Rod Average Scram Insertion Times; and</u>

<u>3/4.3.F</u> <u>Four Control Rod Group Scram Insertion Times</u>

These specifications ensure that the control rod insertion times are consistent with those used in the safety analyses. The control rod system is analyzed to bring the reactor subcritical at a rate fast enough to prevent fuel damage, i.e., to prevent the MCPR from becoming less than the fuel cladding integrity Safety Limit. The analyses demonstrate that if the reactor is operated within the limitation set in Specification 3.11.C, the negative reactivity insertion rates associated with the scram performance (as adjusted for statistical variation in the observed data) result in protection of the MCPR Safety Limit.

Analysis of the limiting power transient shows that the negative reactivity rates, resulting from the scram with the average response of all the drives, as given in the above specification, provide the required protection, and MCPR remains greater than the fuel cladding integrity SAFETY LIMIT. In the analytical treatment of most transients, 290 milliseconds are allowed between a neutron sensor reaching the scram point and the start of motion of the control rods. This is adequate and conservative when compared to the typically observed time delay of about 210 milliseconds. Approximately 90 milliseconds after neutron flux reaches the trip point, the pilot scram valve

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solenoid de-energizes and 120 milliseconds later the control rod motion is estimated to actually begin. However, 200 milliseconds rather than 120 milliseconds is conservatively assumed for this time interval in the transient analyses and is also included in the allowable scram insertion times specified in Specifications 3.3.D, 3.3.E, and 3.3.F. In the statistical treatment of the limiting transients, a statistical distribution of total scram delay is used rather than the bounding value described above.

The performance of the individual control rod drives is monitored to assure that scram performance is not degraded. Observed plant data or Technical Specification limits were used to determine the average scram performance used in the transient analyses, and the results of each set of control rod scram tests performed during the current cycle are compared against earlier results to verify that the performance of the control rod insertion system has not changed significantly.

If test results should be determined to fall outside of the statistical population defining the scram performance characteristics used in the transient analyses, a re-determination of thermal margin requirements is undertaken as required by Specification 3.11.C. A smaller test sample than that required by these specifications is not statistically significant and should not be used in the re-determination of thermal margins. Individual control rod drives with excessive scram times can be fully inserted into the core and de-energized in the manner of an inoperable rod drive provided the allowable number of inoperable control rod drives is not exceeded. In this case, the scram speed of the drive shall not be used as a basis in the re-determination of thermal margin requirements. For excessive average scram insertion times, only the individual control rods in the two-by-two array which exceed the allowed average scram insertion time are considered inoperable.

The scram times for all control rods are measured at the time of each refueling outage. Experience with the plant has shown that control drive insertion times vary little through the operating cycle; hence no re-assessment of thermal margin requirements is expected under normal conditions. The history of drive performance accumulated to date indicates that the 90% insertion times of new and overhauled drives approximate a normal distribution about the mean which tends to become skewed toward longer scram times as operating time is accumulated. The probability of a drive not exceeding the mean 90% insertion time by 0.75 seconds is greater than 0.999 for a normal distribution. The measurement of the scram performance of the drives surrounding a drive, which exceeds the expected range of scram performance, will detect local variations and also provide assurance that local scram time limits are not exceeded. Continued monitoring of other drives exceeding the expected range of scram times provides surveillance of possible anomalous performance.

The test schedule provides reasonable assurance of detection of slow drives before system deterioration beyond the limits of Specification 3.3.C. The program was developed on the basis of the statistical approach outlined above and judgement. The occurrence of scram times within the limits, but significantly longer than average, should be viewed as an indication of a systematic problem with control rod drives, especially if the number of drives exhibiting such scram times exceeds eight, which is the allowable number of inoperable rods.

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3/4.3.G Control Rod Scram Accumulators

The control rod scram accumulators are part of the control rod drive system and are provided to ensure that the control rods scram under varying reactor conditions. The control rod scram accumulators store sufficient energy to fully insert a control rod at any reactor vessel pressure. The accumulator is a hydraulic cylinder with a free floating piston. The piston separates the water used to scram the control rods from the nitrogen which provides the required energy. The scram accumulators are necessary to scram the control rods within the required insertion times.

Control rods with inoperable accumulators are declared inoperable and Specification 3.3.C then applies. This prevents a pattern of inoperable accumulators that would result in less reactivity insertion on a scram than has been analyzed even though control rods with inoperable accumulators may still be inserted with normal drive water pressure. OPERABILITY of the accumulator ensures that there is a means available to insert the control rods even under the most unfavorable depressurization of the reactor.

3/4.3.H Control Rod Drive Coupling

Control rod dropout accidents can lead to significant core damage. If coupling integrity is maintained, the possibility of a rod drop accident is eliminated. Neutron instrumentation response to rod movement may provide verification that a rod is following its drive. Absence of such response to drive movement may indicate an uncoupled condition, or may be due to the lack of proximity of the drive to the instrumentation. However, the overtravel position feature provides a positive check, as only uncoupled drives may reach this position.

3/4.3.I Control Rod Position Indication System (RPIS)

In order to ensure that the control rod patterns can be followed and therefore that other parameters are within their limits, the control rod position indication system must be OPERABLE. Normal control rod position is displayed by two-digit indication to the operator from position 00 to 48. Each even number is a latching position, whereas each odd number provides information while the rod is in-motion and inputs for rod drift annunciation. The ACTION statement provides for the condition where no positive information is displayed for a large portion or all of the rod's travel. Usually, only one digit of one or two of a rod's positions is unavailable with a faulty RPIS, and the control rod may be located in a known position. However, there are several alternate methods for determining control rod position including the full core display, the four rod display, the rod worth minimizer, and the process computer. Another method to determine position would be to move the control rod, by single notch movement, to a position with an OPERABLE position indicator. The original position would then be established and the control rod could be returned to its original position by single notch movement. As long as no control rod drift alarms are received, the position of the control rod would then be known.

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3/4.3.J Control Rod Drive Housing Support

The control rod housing support restricts the outward movement of a control rod to less than 3 inches in the extremely remote event of a housing failure. The amount of reactivity which could be added by this small amount of rod withdrawal, which is less than a normal single withdrawal increment, will not contribute to any damage to the primary coolant system. The design basis is given in Section 4.6.3.5 of the UFSAR. This support is not required if the reactor coolant system is at atmospheric pressure, since there would then be no driving force to rapidly eject a drive housing.

3/4.3.K Scram Discharge Volume Vent and Drain Valves

The scram discharge volume is required to be OPERABLE so that it will be available when needed to accept discharge water from the control rods during a reactor scram and will isolate the reactor coolant system from the containment when required. The operability of the scram discharge volume vent and drain valves assures the proper venting and draining of the volume, so that water accumulation in the volume does not occur. These specifications designate the minimum acceptable level of scram discharge volume vent and drain valve OPERABILITY, provide for the periodic verification that the valves are open, and for the testing of these valves under reactor scram conditions during each refueling outage.

3/4.3.L Rod Worth Minimizer

Control rod withdrawal and insertion sequences are established to assure that the maximum insequence individual control rod or control rod segments which are withdrawn at any time during the fuel cycle could not be worth enough to result in a peak fuel enthalpy greater than 280 cal/gm in the event of a control rod drop accident. These sequences are developed prior to initial operation of the unit following any refueling outage and the requirement that an operator follow these sequences is supervised by the RWM or a second technically qualified individual. These sequences are developed to limit reactivity worth of control rods and, together with the integral rod velocity limiters and the action of the control rod drive system, limit potential reactivity insertion such that the results of a control rod drop accident will not exceed a maximum fuel energy content of 280 cal/gm. The peak fuel enthalpy of 280 cal/gm is below the energy content at which rapid fuel dispersal and primary system damage have been found to occur based on experimental data. Therefore, the energy deposited during a postulated rod drop accident is significantly less than that required for rapid fuel dispersal.

The analysis of the control rod drop accident was originally presented in Sections 7.9.3, 14.2.1.2, and 14.2.1.4 of the original SAR. Improvements in analytical capability have allowed a more refined analysis of the control rod drop accident which is discussed below.

Every operating cycle the peak fuel rod enthalpy rise is determined by comparing cycle specific parameters with the results of parametric analyses. This peak fuel rod enthalpy is then compared

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to the analysis limit of 280 cal/gm to demonstrate compliance for that operating cycle. If the cycle specific parameters are outside the range used in the parametric study, an extension of the enthalpy may be required. Some of the cycle specific parameters used in the analysis are: maximum control rod worth, Doppler coefficient, effective delayed neutron fraction and maximum four bundle local peaking factor. The NRC approved methodology listed in Specification 6.9.A.6 provides a detailed description of the methodology used in performing the rod drop analyses.

The rod worth minimizer provides automatic supervision to assure that out-of-sequence control rods will not be withdrawn or inserted, i.e., it limits operator deviations from planned withdrawal sequences (reference UFSAR Section 7.7.2). It serves as a backup to procedural control of control rod worth. In the event that the rod worth minimizer is out-of-service when required, a second licensed operator or other technically qualified individual who is present at the reactor console can manually fulfill the control rod pattern conformance function of the rod worth minimizer. In this case, the normal procedural controls are backed up by independent procedural controls to assure conformance.

3/4.3.M Rod Block Monitor

The rod block monitor (RBM) is designed to automatically prevent fuel damage in the event of erroneous rod withdrawal from locations of high power density during high power operation. Two channels are provided, and one of these may be bypassed from the console for maintenance and/or testing. Tripping of one of the channels will block erroneous rod withdrawal soon enough to prevent fuel damage. This system backs up the operator, who withdraws control rods according to a written sequence. The specified restrictions with one channel out-of-service conservatively assure that fuel damage will not occur due to rod withdrawal errors when this condition exists.

3/4.3.N Economic Generation Control System

Operation of the facility with the economic generation control system (EGC) (automatic flow control) is limited to the range of 65% to 100% of rated core flow. In this flow range and above 20% of RATED THERMAL POWER, the reactor could safely tolerate a rate of change of load of 8 MWe/sec (reference UFSAR Section 7.7.3.2). Limits within the EGC and the flow control system prevent rates of change greater than approximately 4 MWe/sec. When EGC is in operation, this fact will be indicated on the main control room console.

3.5 - LIMITING CONDITIONS FOR OPERATION

A. Emergency Core Cooling System - Operating

The emergency core cooling systems (ECCS) shall be OPERABLE with:

1. The core spray (CS) system consisting of two subsystems with each subsystem comprised of:

   a. One OPERABLE CS pump, and

   b. An OPERABLE flow path capable of taking suction from the suppression chamber and transferring the water through the spray sparger to the reactor vessel.

2. The low pressure coolant injection (LPCI) subsystem comprised of[e]:

   a. Four OPERABLE LPCI pumps, and

   b. An OPERABLE flow path capable of taking suction from the suppression chamber and transferring the water to the reactor vessel.

3. The high pressure cooling injection (HPCI) system consisting of:

   a. One OPERABLE HPCI pump, and

   b. An OPERABLE flow path capable of taking suction from the suppression chamber and transferring the water to the reactor vessel.

4.5 - SURVEILLANCE REQUIREMENTS

A. Emergency Core Cooling System - Operating

The ECCS shall be demonstrated OPERABLE by:

1. At least once per 31 days:

   a. For the CS system, the LPCI subsystem and the HPCI system:

      1) Verifying that the system piping from the pump discharge valve to the system isolation valve is filled with water.

      2) 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[a] position.

   b. For the HPCI system, verifying that the HPCI pump flow controller is in the correct position.

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e The LPCI subsystem may be considered OPERABLE during alignment and operation for decay heat removal when below the actual RHR cut in permissive pressure in MODE 3, if capable of being manually realigned (remote or local) to the LPCI mode and not otherwise inoperable.

a Except that an automatic valve capable of automatic return to its ECCS position when an ECCS signal is present may be in position for another mode of operation.

3.5 - LIMITING CONDITIONS FOR OPERATION

4. The automatic depressurization system (ADS) with at least 5 OPERABLE ADS valves.

APPLICABILITY:

OPERATIONAL MODE(s) 1, 2[b] and 3[b].

ACTION:

1. For the core spray system:

   a. With one CS subsystem inoperable, provided that the LPCI subsystem is OPERABLE, restore the inoperable CS subsystem to OPERABLE status within 7 days, or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

   b. With both CS subsystems inoperable, be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

2. For the LPCI subsystem:

   a. With one LPCI pump inoperable[f], provided that both CS subsystems are OPERABLE, restore the inoperable LPCI pump to OPERABLE status within 30 days,

4.5 - SURVEILLANCE REQUIREMENTS

2. Verifying that, when tested pursuant to Specification 4.0.E:

   a. The CS pump in each subsystem develop a flow of at least 4500 gpm against a test line pressure corresponding to a reactor vessel pressure of ≥90 psig.

   b. Two LPCI pumps together develop a flow of at least 9,000 gpm against a test line pressure corresponding to a reactor vessel pressure of ≥20 psig.

   c. The HPCI pump develops a flow of at least 5000 gpm against a system head corresponding to reactor vessel pressure, when steam is being supplied to the turbine between 920 and 1005 psig[c].

3. At least once per 18 months:

   a. For the CS system, the LPCI subsystem, and the HPCI system, verify each system/subsystem actuates on an actual or simulated automatic initiation signal. Actual injection of coolant into the reactor vessel may be excluded from this test.

---

b The HPCI system and ADS are not required to be OPERABLE when reactor steam dome pressure is ≤150 psig.

f The provisions of Specification 3.9.A, Actions 4.a or 6.b are applicable to the LPCI subsystem such that with an inoperable diesel generator, for the remaining OPERABLE diesel generator, both LPCI pumps (and their associated flow path) associated with that OPERABLE diesel generator, shall be OPERABLE.

c The provisions of Specification 4.0.D are not applicable provided the surveillance is performed within 12 hours after reactor steam pressure is adequate to perform the test.

3.5 - LIMITING CONDITIONS FOR OPERATION

or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

b. With the LPCI subsystem otherwise inoperable[f], provided that both CS subsystems are OPERABLE, restore the LPCI subsystem to OPERABLE status within 7 days or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours[g].

c. With the LPCI subsystem and one or both CS subsystems inoperable, be in at least HOT SHUTDOWN within 12 hours and in COLD SHUTDOWN within the next 24 hours[g].

3. With the HPCI system inoperable, provided both CS subsystems, the LPCI subsystem, the ADS and the Reactor Core Isolation Cooling (RCIC) system are OPERABLE, restore the HPCI system to OPERABLE status within 14 days or be in at least HOT SHUTDOWN within the next 12 hours and reduce reactor steam dome pressure to ≤150 psig within the following 24 hours.

4.5 - SURVEILLANCE REQUIREMENTS

b. For the HPCI system, verifying that:

1) The system develops a flow of ≥5000 gpm against a system head corresponding to reactor vessel pressure, when steam is being supplied to the turbine between 150 and 180 psig[c].

2) The pump suction is automatically transferred from the condensate storage tank to the suppression chamber on a condensate storage tank water level - low signal and on a suppression chamber water level - high signal.

c. Performing a CHANNEL CALIBRATION of the ECCS discharge line "keep filled" alarm instrumentation.

d. Deleted.

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f The provisions of Specification 3.9.A, Actions 4.a or 6.b are applicable to the LPCI subsystem such that with an inoperable diesel generator, for the remaining OPERABLE diesel generator, both LPCI pumps (and their associated flow path) associated with that OPERABLE diesel generator, shall be OPERABLE.

g Whenever the two required RHR SDC mode subsystems are inoperable, if unable to attain COLD SHUTDOWN as required by this ACTION, maintain reactor coolant temperature as low as practical by use of alternate heat removal methods.

c The provisions of Specification 4.0.D are not applicable provided the surveillance is performed within 12 hours after reactor steam pressure is adequate to perform the test.

3.5 - LIMITING CONDITIONS FOR OPERATION

4. For the ADS:

   a. With one of the above required ADS valves inoperable, provided the HPCI system, both CS subsystems and three LPCI pumps are OPERABLE, restore the inoperable ADS valve to OPERABLE status within 14 days or be in at least HOT SHUTDOWN within the next 12 hours and reduce reactor steam dome pressure to ≤150 psig within the following 24 hours.

   b. With two or more of the above required ADS valves inoperable, be in at least HOT SHUTDOWN within 12 hours and reduce reactor steam dome pressure to ≤150 psig within the following 24 hours.

5. With an ECCS discharge line "keep filled" pressure alarm instrumentation CHANNEL inoperable, perform Surveillance Requirement 4.5.A.1.a.1) for CS and LPCI at least once per 24 hours.

4.5 - SURVEILLANCE REQUIREMENTS

4. At least once per 18 months for the ADS:

   a. Verifying the ADS actuates on an actual or simulated automatic initiation signal. Actual valve actuation may be excluded from this test.

   b. Manually opening each ADS valve when the reactor steam dome pressure is ≥150 psig[c] and observing that either:

      1) The turbine control valve or turbine bypass valve position responds accordingly, or

      2) There is a corresponding change in the measured steam flow.

c The provisions of Specification 4.0.D are not applicable provided the surveillance is performed within 12 hours after reactor steam pressure is adequate to perform the test.

3.5 - LIMITING CONDITIONS FOR OPERATION

6. Deleted.

7. In the event an ECCS system 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.B 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.

4.5 - SURVEILLANCE REQUIREMENTS

3.5 - LIMITING CONDITIONS FOR OPERATION

D. Reactor Core Isolation Cooling System

The reactor core isolation cooling (RCIC) system shall be OPERABLE with an OPERABLE flow path capable of automatically taking suction from the suppression chamber and transferring the water to the reactor pressure vessel.

APPLICABILITY:

OPERATIONAL MODE(s) 1, 2 and 3 with reactor steam dome pressure >150 psig.

ACTION:

With the RCIC system inoperable, operation may continue provided the HPCI system is OPERABLE; restore the RCIC system to OPERABLE status within 14 days or be in at least HOT SHUTDOWN within the next 12 hours and reduce reactor steam dome pressure to ≤150 psig within the following 24 hours.

4.5 - SURVEILLANCE REQUIREMENTS

D. Reactor Core Isolation Cooling System

The RCIC system shall be demonstrated OPERABLE:

1. At least once per 31 days by:
   a. Verifying that the system piping from the pump discharge valve to the system isolation valve is filled with water.
   b. 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.
   c. Verifying that the pump flow controller is in the correct position.
2. At least once per 92 days, when tested pursuant to 4.0.E, by verifying that the RCIC pump develops a flow of ≥400 gpm against a system head corresponding to reactor vessel pressure when steam is being supplied to the turbine between 920 and 1005 psig[a].
3. At least once per 18 months by:
   a. Verifying the RCIC system actuates on an actual or simulated automatic initiation signal. Actual injection of coolant into the reactor vessel may be excluded.

a The provisions of Specification 4.0.D are not applicable provided the surveillance is performed within 12 hours after reactor steam pressure is adequate to perform the test.

3.5 - LIMITING CONDITIONS FOR OPERATION

4.5 - SURVEILLANCE REQUIREMENTS

b. Verifying that the system will develop a flow of ≥400 gpm against a system head corresponding to reactor vessel pressure, when steam is supplied to the turbine at a pressure between 150 and 180 psig[a].

c. Verifying that the suction for the RCIC system is automatically transferred from the condensate storage tank to the suppression pool on a condensate storage tank water level - low signal and on a suppression pool water level - high signal.

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a The provisions of Specification 4.0.D are not applicable provided the surveillance is performed within 12 hours after reactor steam pressure is adequate to perform the test.

BASES

With the HPCI system inoperable, adequate core cooling is assured by the OPERABILITY of the redundant and diversified Automatic Depressurization System and both the CS system and LPCI subsystem. In addition, the Reactor Core Isolation Cooling (RCIC) system, a system for which no credit is taken in the safety analysis, will automatically initiate on a reactor low water level condition. The HPCI out-of-service period of 14 days is based on the demonstrated OPERABILITY of redundant and diversified low pressure core cooling systems and the RCIC system.

The surveillance requirements provide adequate assurance that the HPCI system will be OPERABLE when required. Although all active components are testable and full flow can be demonstrated by recirculation through a test loop during reactor operation, a complete system functional test requires a reactor shutdown. The pump discharge piping is maintained full to prevent water hammer damage and to provide cooling at the earliest moment.

Upon failure of the HPCI system to function properly after a small break loss-of-coolant, the Automatic Depressurization System (ADS) automatically causes all OPERABLE main steamline relief valves to open, depressurizing the reactor so that flow from the low pressure core cooling systems can enter the core in time to limit fuel cladding temperature to less than 2200°F. ADS is conservatively required to be OPERABLE whenever reactor vessel pressure exceeds 150 psig. This pressure is substantially below that for which the low pressure core cooling systems can provide adequate core cooling for events requiring ADS.

ADS automatically controls the five main steamline relief valves although the safety analyses support a minimum of 4 OPERABLE valves. It is therefore appropriate to permit one valve to be out-of-service for up to 14 days without materially reducing system reliability. A manual actuation of each ADS valve is performed to verify that the valve and solenoid are functioning properly and that no blockage exists in the ADS discharge lines. This is demonstrated by the response of the turbine control or bypass valve or by a change in the measured steam flow or by any other method suitable to verify steam flow. Adequate reactor steam dome pressure must be available to perform this test to avoid damaging the valve. Sufficient time is therefore allowed after the required pressure is achieved to perform this test once only. The pressure specified for this test is that pressure recommended by the valve manufacturer. Reactor startup is allowed prior to performing this test because valve OPERABILITY and the setpoints for overpressure protection are verified, per ASME requirements, prior to valve installation. Thus, a footnote is included in this SR to indicate that 4.0.D does not apply.

To preserve single failure criteria, a minimum of two independent OPERABLE low-pressure ECCS subsystems/loops are required in OPERATIONAL MODE(s) 4 and 5 to ensure adequate vessel inventory makeup in the event of an inadvertent vessel draindown. Only a single LPCI pump is required per loop because of the large injection capacity. All of the ECCS may be inoperable provided the reactor head is removed, the reactor cavity is flooded, the spent fuel gates are removed, and the water level is maintained within the limits required by the Refueling Operations specifications.

BASES

3/4.5.C Suppression Chamber

The suppression chamber is required to be OPERABLE as part of the ECCS to ensure that a sufficient supply of water is available to the HPCI and CS systems and the LPCI subsystem in the event of a LOCA. This limit on suppression chamber minimum water volume ensures that sufficient water is available to permit recirculation cooling flow to the core. The OPERABILITY of the suppression chamber in OPERATIONAL MODE(s) 1, 2 or 3 is also required by Specification 3.7.K.

Repair work might require making the suppression chamber inoperable. This specification will permit those repairs to be made and concurrently provide assurance that the irradiated fuel has an adequate cooling water supply when the suppression chamber must be made inoperable, including draining, in OPERATIONAL MODE(s) 4 or 5.

In OPERATIONAL MODE(s) 4 and 5 the suppression chamber minimum required water volume is reduced because the reactor coolant is maintained at or below 212°F. Since pressure suppression is not required below 212°F, the minimum water volume is based on net positive suction head (NPSH), recirculation volume and vortex prevention plus a safety margin for conservatism. With the suppression chamber water level less than the required limit, all ECCS subsystems are inoperable unless they are aligned to an OPERABLE condensate storage tank. When the suppression chamber level is less than 7 feet, the CS system or the LPCI subsystem is considered OPERABLE only if it can take suction from the condensate storage tank, and the condensate storage tank water level is sufficient to provide the required NPSH for the CS or LPCI pumps. Therefore, a verification that either the suppression chamber water level is greater than or equal to 7 feet or that CS or LPCI is aligned to take suction from the condensate storage tank and the condensate storage tank contains greater than or equal to 140,000 gallons of water, ensures CS or LPCI can supply at least 50,000 gallons of make-up water to the reactor pressure vessel. The CS suction is uncovered at the 90,000 gallon level.

3/4.5.D Reactor Core Isolation Cooling

The Reactor Core Isolation Cooling (RCIC) system is provided to supply continuous makeup water to the reactor core when the feedwater system is isolated from the turbine and when the feedwater system is not available. Under these conditions, the pumping capacity of the RCIC system is sufficient to maintain the water level above the core without any other water system in operation. If the water level in the reactor vessel decreases to the RCIC initiation level, the system automatically starts. The system may also be manually initiated at any time. The RCIC system is conservatively required to be OPERABLE whenever reactor pressure exceeds 150 psig even though the LPCI mode of the residual heat removal (RHR) system provides adequate core cooling up to 350 psig.

The RCIC system specifications are applicable during OPERATIONAL MODE(s) 1, 2 and 3 when reactor vessel pressure exceeds 150 psig because RCIC is the primary non-ECCS source of core cooling when the reactor is pressurized.

BASES

The HPCI subsystem provides an alternate method of supplying makeup water to the reactor should the normal feedwater become unavailable. Therefore, the specification calls for an OPERABILITY check of the HPCI subsystem should the RCIC system be found to be inoperable.

The surveillance requirements provide adequate assurance that RCIC will be OPERABLE when required. Although all active components are testable and full flow can be demonstrated by recirculation during reactor operation, a complete functional test requires a reactor shutdown. The pump discharge piping is maintained full to prevent water hammer damage and to start cooling at the earliest possible moment.

3.6 - LIMITING CONDITIONS FOR OPERATION

E. Safety Valves

The safety valve function of the 9 reactor coolant system safety valves shall be OPERABLE in accordance with the specified code safety valve function lift settings[a] established as:

1 safety valve[b] @1135 psig ±1%
2 safety valves @1240 psig ±1%
2 safety valves @1250 psig ±1%
4 safety valves @1260 psig ±1%

APPLICABILITY:

OPERATIONAL MODE(s) 1, 2 and 3.

ACTION:

1. With the safety valve function of one or more of the above required safety valves inoperable, be in at least HOT SHUTDOWN within 12 hours and in COLD SHUTDOWN within the next 24 hours.

2. Deleted.

4.6 - SURVEILLANCE REQUIREMENTS

E. Safety Valves

1. Deleted.

2. At least once per 18 months, 1/2 of the safety valves shall be removed, set pressure tested and reinstalled or replaced with spares that have been previously set pressure tested and stored in accordance with manufacturer's recommendations. At least once per 40 months, the safety valves shall be rotated such that all 9 safety valves are removed, set pressure tested and reinstalled or replaced with spares that have been previously set pressure tested and stored in accordance with manufacturer's recommendations.

a The lift setting pressure shall correspond to ambient conditions of the valves at nominal operating temperatures and pressures.

b Target Rock combination safety/relief valve.

3.6 - LIMITING CONDITIONS FOR OPERATION

F. Relief Valves

5 reactor coolant system relief valves and the reactuation time delay of two relief valves shall be OPERABLE with the following settings:

Relief Function Setpoint (psig)
Open
≤1115 psig
≤1115 psig
≤1135 psig
≤1135 psig
≤1135 psig[a]

APPLICABILITY:

OPERATIONAL MODE(s) 1, 2 and 3.

ACTION:

1. With one or more relief valves open, provided that suppression pool average water temperature is <110°F, take action to close the open relief valve(s); if suppression pool average water temperature is ≥110°F place the reactor mode switch in the Shutdown position.

4.6 - SURVEILLANCE REQUIREMENTS

F. Relief Valves

1. The relief valve function and the reactuation time delay function instrumentation shall be demonstrated OPERABLE by performance of a:

   a. CHANNEL FUNCTIONAL TEST of the relief valve function at least once per 18 months, and a

   b. CHANNEL CALIBRATION and LOGIC SYSTEM FUNCTIONAL TEST of the entire system at least once per 18 months.

2. Deleted.

a Target Rock combination safety/relief valve.

3.6 - LIMITING CONDITIONS FOR OPERATION

2. With the relief valve function and/or the reactuation time delay of one of the above required reactor coolant system relief valves inoperable, restore the inoperable relief valve function and the reactuation time delay function to OPERABLE status within 14 days or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

3. With the relief valve function and/or the reactuation time delay of more than one of the above required reactor coolant system relief valves inoperable, be in at least HOT SHUTDOWN within 12 hours and in COLD SHUTDOWN within the next 24 hours.

4. Deleted.

4.6 - SURVEILLANCE REQUIREMENTS

TABLE 3.6.I-1

REACTOR COOLANT SYSTEM CHEMISTRY LIMITS

OPERATIONAL MODE(s)	Chlorides	Conductivity (μmhos/cm @25°C)	pH
1	≤0.2 ppm	≤1.0	5.6≤ pH ≤8.6
2 and 3	≤0.1 ppm	≤2.0	5.6≤ pH ≤8.6

3.6 - LIMITING CONDITIONS FOR OPERATION

O. Residual Heat Removal - HOT SHUTDOWN

Two shutdown cooling mode subsystems of the residual heat removal (RHR) system shall be OPERABLE[a] and, unless at least one recirculation pump is in operation, at least one shutdown cooling mode subsystem shall be capable of circulating reactor coolant[b] with each subsystem consisting of at least:

1. One OPERABLE RHR pump, and
2. One OPERABLE RHR heat exchanger.

APPLICABILITY:

OPERATIONAL MODE 3, with reactor vessel pressure less than the RHR cut-in permissive setpoint.

ACTION:

1. With less than the above required RHR shutdown cooling mode subsystems OPERABLE, immediately initiate corrective action to return the required subsystems to OPERABLE status as soon as possible. Within 1 hour and at least once per 24 hours thereafter, demonstrate the operability of at least one alternate method capable of decay heat removal for each inoperable RHR shutdown cooling mode subsystem. Be in at least COLD SHUTDOWN within 24 hours.[c]

4.6 - SURVEILLANCE REQUIREMENTS

O. Residual Heat Removal - HOT SHUTDOWN

At least one shutdown cooling mode subsystem of the residual heat removal system, recirculation pump or alternate method shall be verified to be capable of circulating reactor coolant at least once per 12 hours.

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a Each shutdown cooling subsystem is considered OPERABLE if it can be manually aligned (remote or local) in the shutdown cooling mode for removal of decay heat. The provisions of Specification 3.0.D are not applicable.

b The RHR shutdown cooling subsystem may be removed from operation during hydrostatic testing.

c Whenever the two required RHR SDC mode subsystems are inoperable, if unable to attain COLD SHUTDOWN as required by this ACTION, maintain reactor coolant temperature as low as practical by use of alternate heat removal methods.

3.6 - LIMITING CONDITIONS FOR OPERATION

P. Residual Heat Removal - COLD SHUTDOWN

Two shutdown cooling mode subsystems of the residual heat removal (RHR) system shall be OPERABLE[a] and, unless at least one recirculation pump is in operation, at least one shutdown cooling mode subsystem shall be capable of circulating reactor coolant[b] with each subsystem consisting of at least:

1. One OPERABLE RHR pump, and

2. One OPERABLE RHR heat exchanger.

APPLICABILITY:

OPERATIONAL MODE 4.

ACTION:

1. With less than the above required RHR shutdown cooling mode subsystems OPERABLE, within 1 hour and at least once per 24 hours thereafter, demonstrate the operability of at least one alternate method capable of decay heat removal for each inoperable RHR shutdown cooling mode subsystem.

4.6 - SURVEILLANCE REQUIREMENTS

P. Residual Heat Removal - COLD SHUTDOWN

At least one shutdown cooling mode subsystem of the residual heat removal system, recirculation pump or alternate method shall be verified to be capable of circulating reactor coolant at least once per 12 hours.

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a Each shutdown cooling subsystem is considered OPERABLE if it can be manually aligned (remote or local) in the shutdown cooling mode for removal of decay heat.

b The RHR shutdown cooling subsystem may be removed from operation during hydrostatic testing.

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assurance that the recirculation flow is not bypassing the core through inactive or broken jet pumps. The change in the flow rate of the failed jet pump produces a change in the indicated flow rate of that pump relative to the other pumps in that loop. Comparison of the data with a normal pattern provides the indication necessary to detect a failed jet pump. Allowable deviations from the established patterns have been developed based on operation. Since refueling activities (fuel assembly replacement or shuffle, as well as any modification to fuel support orifice size or core plate bypass flow) can affect the relationship between core flow, jet pump flow, and recirculation loop flow, these relationships may need to be reestablished each cycle. Similarly, initial entry into extended single loop operation may also require establishment of these relationships. During the initial weeks of operation under such conditions, while base-lining new "established patterns," engineering judgement of the daily surveillance results is used to detect significant abnormalities which could indicate jet pump failure.

The accuracy of the core flow measurement system is assumed in the derivation of the Safety Limit MINIMUM CRITICAL POWER RATIO. An analysis assuming a loss of flow indication for three jet pumps resulted in uncertainties within the values assumed for the core flow measurement system in the Safety Limit MINIMUM CRITICAL POWER RATIO calculation for both two loop operation and single loop operation. Therefore, plant operation with loss of flow indication in up to two jet pumps is acceptable as long as each jet pump is on a separate riser and no more than one calibrated double tap jet pump per loop is affected.

Recirculation pump speed mismatch limits are in compliance with the ECCS LOCA analysis design criteria. For some limited low probability events with the recirculation loop operating with large speed differences, it is possible for the LPCI loop selection logic to select the wrong loop for injection. Above 80% of RATED THERMAL POWER, the LPCI selection logic is expected to function at a speed differential of 15%. Below 80% of RATED THERMAL POWER, the loop select logic would be expected to function at a speed differential of 20%. Therefore, this specification provides a margin of 5% in pump speed differential before a problem could arise.

In order to prevent undue stress on the vessel nozzles and bottom head region, the recirculation loop temperatures shall be within 50°F of each other prior to startup of an idle loop. The loop temperature must also be within 50°F of the reactor pressure vessel steam space coolant temperature to prevent thermal shock to the recirculation pump and recirculation nozzles. Since the coolant in the bottom of the vessel is at a lower temperature than the coolant in the upper regions of the core, undue stress on the vessel would result if the temperature difference was greater than 145°F. Additionally, asymmetric speed operation of the recirculation pumps during idle loop startup induces levels of jet pump riser vibration that are higher than normal. The specific limitation of 45% of rated pump speed for the operating recirculation pump prior to the start of the idle recirculation pump ensures that the recirculation pump speed mismatch requirements are maintained.

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3/4.6.E Safety Valves

3/4.6.F Relief Valves

The American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code requires the reactor pressure vessel be protected from overpressure during upset conditions by self-actuated safety valves. As part of the nuclear pressure relief system, the size and number of safety valves are selected such that peak pressure in the nuclear system will not exceed the ASME Code limits for the reactor coolant pressure boundary. The overpressure protection system must accommodate the most severe pressurization transient. Evaluations have determined that the most severe transient is the closure of all the main steam line isolation valves followed by a reactor scram on high neutron flux. The analysis results demonstrate that the design safety valve capacity is capable of maintaining reactor pressure below the ASME Code limit of 110% of the reactor pressure vessel design pressure.

The relief valve function is not assumed to operate in response to any accident, but are provided to remove the generated steam flow upon turbine stop valve closure coincident with failure of the turbine bypass system. The relief valve opening pressure settings are sufficiently low to prevent the need for safety valve actuation following such a transient.

Each of the five relief valves discharge to the suppression chamber via a dedicated relief valve discharge line. Steam remaining in the relief valve discharge line following closure can condense, creating a vacuum which may draw suppression pool water up into the discharge line. This condition is normally alleviated by the vacuum breakers; however, subsequent actuation in the presence of an elevated water leg can result in unacceptably high thrust loads on the discharge piping. To prevent this, the relief valves have been designed to ensure that each valve which closes will remain closed until the normal water level in the relief valve discharge line is restored. The opening and closing setpoints are set such that all pressure induced subsequent actuation are limited to the two lowest set valves. These two valves are equipped with additional logic which functions in conjunction with the setpoints to inhibit valve reopening during the elevated water leg duration time following each closure.

Each safety/relief valve is equipped with diverse position indicators which monitor the tailpipe acoustic vibration and temperature. Either of these provide sufficient indication of safety/relief valve position for normal operation.

3/4.6.G Leakage Detection Systems

The RCS leakage detection systems required by this specification are provided to monitor and detect leakage from the reactor coolant pressure boundary. Limits on leakage from the reactor coolant pressure boundary are required so that appropriate action can be taken before the integrity of the reactor coolant pressure boundary is impaired. Leakage detection systems for the reactor coolant system are provided to alert the operators when leakage rates above the normal background levels are detected and also to supply quantitative measurement of leakage rates.

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Leakage from the reactor coolant pressure boundary inside the drywell is detected by at least one or two independently monitored variables, such as sump level changes and drywell atmosphere radioactivity levels. The means of quantifying leakage in the drywell is the drywell floor drain sump pumps. With the drywell floor drain sump pump system inoperable, no other form of monitoring can provide the equivalent information. However, primary containment atmosphere sampling for radioactivity can provide indication of changes in leakage rates.

3/4.6.H Operational Leakage

The allowable leakage rates from the reactor coolant system have been based on the predicted and experimentally observed behavior of cracks in pipes. The normally expected background leakage due to equipment design and the detection capability of the instrumentation for determining system leakage was also considered. The evidence obtained from experiments suggests that for leakage somewhat greater than that specified for UNIDENTIFIED LEAKAGE the probability is small that the imperfection or crack associated with such leakage would grow rapidly. However, in all cases, if the leakage rates exceed the values specified or the leakage is located and known to be PRESSURE BOUNDARY LEAKAGE, the reactor will be shutdown to allow further investigation and corrective action.

An UNIDENTIFIED LEAKAGE increase of more than 2 gpm within a 24 hour period is an indication of a potential flaw in the reactor coolant pressure boundary and must be quickly evaluated. Although the increase does not necessarily violate the absolute UNIDENTIFIED LEAKAGE limit, IGSCC susceptible components must be determined not to be the source of the leakage within the required completion time.

3/4.6.I Chemistry

The water chemistry limits of the reactor coolant system are established to prevent damage to the reactor materials in contact with the coolant. Chloride limits are specified to prevent stress corrosion cracking of the stainless steel. The effect of chloride is not as great when the oxygen concentration in the coolant is low, thus the 0.2 ppm limit on chlorides is permitted during POWER OPERATION.

Conductivity measurements are required on a continuous basis since changes in this parameter are an indication of abnormal conditions. When the conductivity is within limits, the pH, chlorides and other impurities affecting conductivity must also be within their acceptable limits. With the conductivity meter inoperable, additional samples must be analyzed to ensure that the chlorides are not exceeding the limits.

Action 1 permits temporary operation with chemistry limits outside of the limits required in OPERATIONAL MODE 1 without requiring Commission notification. The surveillance requirements provide adequate assurance that concentrations in excess of the limits will be detected in sufficient time to take corrective action.

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3/4.6.J Specific Activity

The limitations on the specific activity of the primary coolant ensure that the 2 hour thyroid and whole body doses resulting from a main steam line failure outside the containment during steady state operation will not exceed small fractions of the dose guidelines of 10 CFR 100. The values for the limits on specific activity represent interim limits based upon a parametric evaluation by the NRC of typical site locations. These values are conservative in that specific site parameters, such as site boundary location and meteorological conditions, were not considered in this evaluation.

The ACTION statement permitting POWER OPERATION to continue for limited time periods with the primary coolant's specific activity greater than 0.2 microcuries per gram DOSE EQUIVALENT I-131, but less than or equal to 4.0 microcuries per gram DOSE EQUIVALENT I-131, accommodates possible iodine spiking phenomenon which may occur following changes in THERMAL POWER. Information obtained on iodine spiking will be used to assess the parameters associated with spiking phenomena. A reduction in frequency of isotopic analysis following power changes may be permissible if justified by the data obtained.

Closing the main steam line isolation valves prevents the release of activity to the environs should a steam line rupture occur outside containment. The surveillance requirements provide adequate assurance that excessive specific activity levels in the reactor coolant will be detected in sufficient time to take corrective action.

3/4.6.K Pressure/Temperature Limits

All components in the reactor coolant system are designed to withstand the effects of cyclic loads due to system temperature and pressure changes. These cyclic loads are introduced by normal load transients, reactor trips, and startup and shutdown operations. The various categories of load cycles used for design purposes are provided in Section 3.9.1.1.1 of the UFSAR. During startup and shutdown, the rates of temperature and pressure changes are limited so that the maximum specified heatup and cooldown rates are consistent with the design assumptions and satisfy the stress limits for cyclic operation.

During heatup, the thermal gradients in the reactor vessel wall produce thermal stresses which vary from compressive at the inner wall to tensile at the outer wall. These thermal induced compressive stresses tend to alleviate the tensile stresses induced by the internal pressure. Therefore, a pressure temperature curve based on steady state conditions, i.e., no thermal stresses, represents a lower bound of all similar curves for finite heatup rates when the inner wall of the vessel is treated as the governing location.

The heatup analysis also covers the determination of pressure-temperature limitations for the case in which the outer wall of the vessel becomes the controlling location. The thermal gradients established during heatup produce tensile stresses which are already present. The thermal induced stresses at the outer wall of the vessel are tensile and are dependent on both the rate of heatup and the time along the heatup ramp; therefore, a lower bound curve similar to that described for

BASES

the heatup of the inner wall cannot be defined. Subsequently, for the cases in which the outer wall of the vessel becomes the stress controlling location, each heatup rate of interest must be analyzed on an individual basis.

The pressure-temperature limit lines shown in Figure 3.6.K-1, for operating conditions; Inservice Hydrostatic Testing (curve A), Non-Nuclear Heatup/Cooldown (curve B), and Core Critical Operation (curve C). The curves have been established to be in conformance with Appendix G to 10 CFR Part 50 and Regulatory Guide 1.99 Revision 2, and take into account the change in reference nil-ductility transition temperature (RT_{NDT}) as a result of neutron embrittlement. The adjusted reference temperature (ART) of the limiting vessel material is used to account for irradiation effects.

Three vessel regions are considered for the development of the pressure-temperature curves: 1) the core beltline region; 2) the non-beltline region (other than the closure flange region); and 3) the closure flange region. The beltline region is defined as that region of the reactor vessel that directly surrounds the effective height of the reactor core and is subject to an RT_{NDT} adjustment to account for radiation embrittlement. The non-beltline and closure flange regions receive insufficient fluence to necessitate an RT_{NDT} adjustment. These regions contain components which include; the reactor vessel nozzles, closure flanges, top and bottom head plates, control rod drive penetrations, and shell plates that do not directly surround the reactor core. Although the closure flange region is a non-beltline region, it is treated separately for the development of the pressure-temperature curves to address 10CFR Part 50 Appendix G requirements.

In evaluating the adequacy of the steel which comprises the reactor vessel, it is necessary that the following be established: 1) the RT_{NDT} for all vessel and adjoining materials; 2) the relationship between RT_{NDT} and integrated neutron flux (fluence, at energies greater than one Mev); and 3) the fluence at the location of a postulated flaw.

Boltup Temperature

The initial RT_{NDT} of the main closure flanges, the shell and head materials connecting to these flanges, and connecting welds is 10°F; however, the vertical electroslag welds which terminate immediately below the vessel flange have an RT_{NDT} of 40°F. Therefore, the minimum allowable boltup temperature is established as 100°F (RT_{NDT} + 60°F) which includes a 60°F conservatism required by the original ASME Code of construction.

Curve A - Hydrotesting

As indicated in curve A of Figure 3.6.K-1 for system hydrotesting, the minimum metal temperature of the reactor vessel shell is 100°F for reactor pressures less than 312 psig. This 100°F minimum boltup temperature is based on a RT_{NDT} of 40°F for the electroslag weld immediately below the vessel flange and a 60°F conservatism required by the original ASME Code of construction. At reactor pressures greater than 312 psig, the minimum vessel metal temperature is established as 130°F. The 130°F minimum temperature is based on a closure flange region RT_{NDT} of 40°F and a 90°F conservatism required by 10CFR Part 50 Appendix G

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for pressure in excess of 20% of the preservice hydrostatic test pressure (1563 psig). At approximately 650 psig the effects of pressurization are more limiting than the boltup stresses at the closure flange region, hence a family of non-linear curves intersect the 130°F vertical line. Beltline as well as non-beltline curves have been provided to allow separate monitoring of the two regions. Beltline curves as a function of vessel exposure for 12, 14 and 16 effective full power years (EFPY) are presented to allow the use of the appropriate curve up to 16 EFPY of operation.

A typical sequence involved in pressure testing is a heatup to the required temperature and then pressurization to the required pressure for the inspection. During the heatup, at 100°F/hour or less, Curve B is the governing curve. Since the vessel is not pressurized during the heatup, Curves A and B are the same. When temperatures are stabilized to within 20°F/hour rates, at temperatures above those required by curve A, pressurization begins, at which point Curve A is the governing curve. During the inspection period with the vessel at the required pressure, temperature changes are limited to 20°F/hour.

Curve B - Non-Nuclear Heatup/Cooldown

Curve B of Figure 3.6.K-1 applies during heatups with non-nuclear heat (e.g., recirculation pump heat) and during cooldowns when the reactor is not critical (e.g., following a scram). The curve provides the minimum reactor vessel metal temperatures based on the most limiting vessel stress. As indicated by the vertical 100°F line, the boltup stresses at the closure flange region are most limiting for reactor pressures below approximately 110 psig. For reactor pressures greater than approximately 110 psig, pressurization and thermal stresses become more limiting than the boltup stresses, which is reflected by the nonlinear portion of curve B. The non-linear portion of the curve is dependent on non-beltline and beltline regions, with the beltline region temperature limits having been adjusted to account for vessel irradiation (up to a vessel exposure of 16 EFPY). The non-beltline region is limiting between approximately 110 psig and 830 psig. Above approximately 803 psig, the beltline region becomes limiting.

Curve C - Core Critical Operation

Curve C, the core critical operation curve shown in Figure 3.6.K-1, is generated in accordance with 10CFR Part 50 Appendix G which requires core critical pressure-temperature limits to be 40°F above any curve A or B limits. Since curve B is more limiting, (curve C is curve B plus 40°F.

The actual shift in RT_{NDT} of the vessel material will be established periodically during operation by removing and evaluating, in accordance with ASTM E185-73 and 10CFR Part 50, Appendix H, irradiated reactor vessel material specimens installed near the inside wall of the reactor vessel in the core area. The irradiated specimens can be used with confidence in predicting reactor vessel material transition temperature shift. The operating limit curves of Figure 3.6.K-1 shall be adjusted, as required, on the basis of the specimen data and recommendations of Regulatory Guide 1.99, Revision 2.

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3/4.6.L Reactor Steam Dome Pressure

The reactor steam dome pressure is an assumed initial condition of Design Basis Accidents and transients and is also an assumed value in the determination of compliance with reactor pressure vessel overpressure protection criteria. The reactor steam dome pressure of ≤1005 psig is an initial condition of the vessel overpressure protection analysis. This analysis assumes an initial maximum reactor steam dome pressure and evaluates the response of the pressure relief system, primarily the safety valves, during the limiting pressurization transient. The determination of compliance with the overpressure criteria is dependent on the initial reactor steam dome pressure; therefore, the limit on this pressure ensures that the assumptions of the overpressure protection analysis are conserved.

3/4.6.M Main Steam Line Isolation Valves

Double isolation valves are provided on each of the main steam lines to minimize the potential leakage paths from the containment in case of a line break. Only one valve in each line is required to maintain the integrity of the containment, however, single failure considerations require that two valves be OPERABLE. The surveillance requirements are based on the operating history of this type of valve. The maximum closure time has been selected to contain fission products and to ensure the core is not uncovered following line breaks. The minimum closure time is consistent with the assumptions in the safety analyses to prevent pressure surges.

3/4.6.N Structural Integrity

The inspection programs for ASME Code Class 1, 2 and 3 components ensure that the structural integrity of these components will be maintained at an acceptable level throughout the life of the plant.

The inservice inspection program for ASME Code Class 1, 2 and 3 components will be performed in accordance with Section XI of the ASME Boiler and Pressure Vessel Code and applicable addenda as required by 10 CFR Part 50.55a(g) except where specific written relief has been granted by the NRC pursuant to 10 CFR Part 50.55a(g)(6)(i).

3/4.6.O Residual Heat Removal - HOT SHUTDOWN

3/4.6.P Residual Heat Removal - COLD SHUTDOWN

Irradiated fuel in the reactor pressure vessel generates decay heat during normal and abnormal shutdown conditions, potentially resulting in an increase in the temperature of the reactor coolant. This decay heat is required to be removed such that the reactor coolant temperature can be reduced in preparation for performing refueling, maintenance operations or for maintaining the

3.7 - LIMITING CONDITIONS FOR OPERATION

A. PRIMARY CONTAINMENT INTEGRITY

PRIMARY CONTAINMENT INTEGRITY shall be maintained.

APPLICABILITY:

OPERATIONAL MODE(s) 1, 2[a] and 3.

ACTION:

Without PRIMARY CONTAINMENT INTEGRITY, restore PRIMARY CONTAINMENT INTEGRITY within 1 hour or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

4.7 - SURVEILLANCE REQUIREMENTS

A. PRIMARY CONTAINMENT INTEGRITY

PRIMARY CONTAINMENT INTEGRITY shall be demonstrated:

1. After each closing of each penetration subject to Type B testing, except the primary containment air locks, if opened following Type A or B test, by leak rate testing the seals with gas at $\geq P_a$ (48 psig), and verifying that when the measured leakage rate for these seals is added to the leakage rates determined pursuant to Surveillance Requirement 4.7.B.4 for all other Type B and C penetrations, the combined leakage rate is $\leq 0.60\ L_a$.
2. At least once per 31 days by verifying that all primary containment penetrations[b] not capable of being closed by OPERABLE containment automatic isolation valves and required to be closed during accident conditions are closed, except for valves that are open under administrative control as permitted by Specification 3.7.D.
3. By verifying each primary containment air lock is in compliance with the requirements of Specification 3.7.C.
4. By verifying the suppression chamber is in compliance with the requirements of Specification 3.7.K.

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a See Special Test Exception 3.12.A.

b Except valves, blind flanges, and deactivated automatic valves which are located inside the containment. Valves and blind flanges in high radiation areas may be verified by use of administrative controls. These penetrations shall be verified closed during each COLD SHUTDOWN except such verification need not be performed when the primary containment has not been de-inerted since the last verification or more often than once per 92 days.

3.7 - LIMITING CONDITIONS FOR OPERATION

B. Primary Containment Leakage

Primary containment leakage rates shall be limited to:

1. An overall integrated leakage rate of $\leq L_a$ which is defined as 1.0 percent by weight of the containment air per 24 hours at P_a (48 psig).

2. A combined leakage rate of $\leq 0.60\ L_a$ for all primary containment penetrations, except[a] for main steam line isolation valves, subject to Type B and C tests when pressurized to P_a (48 psig).

3. $\leq$11.5 scfh for any one main steam line isolation valve when tested at P_t (25 psig)[a].

APPLICABILITY:

When PRIMARY CONTAINMENT INTEGRITY is required per Specification 3.7.A.

ACTION:

With the measured combined leakage rate for all primary containment penetrations subject to Type B and C tests $>0.60\ L_a$, restore the combined leakage rate to $\leq 0.60\ L_a$, within 1 hour. Otherwise, be in HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

4.7 - SURVEILLANCE REQUIREMENTS

B. Primary Containment Leakage

The primary containment leakage rates shall be demonstrated at the following test schedule and shall be determined in conformance with the criteria, methods and provisions specified in Appendix J of 10CFR Part 50, as modified by approved exemptions:

1. Three Type A overall integrated containment leakage rate tests shall be conducted at approximately equal intervals during shutdown at $\geq P_a$ (48 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.

2. If the results of any periodic Type A test are $>0.75\ L_a$, the test schedule for subsequent Type A tests shall be reviewed and approved by the Commission. If the results of two consecutive Type A tests are $>0.75\ L_a$, a Type A test shall be performed at intervals in accordance with 10 CFR Part 50, Appendix J, as modified by approved exemptions, until the results of two consecutive Type A tests are $\leq 0.75\ L_a$, at which time the above test schedule may be resumed.

3. The accuracy of each Type A test shall be verified by a supplemental test which:

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

a Exemption from Appendix J to 10CFR Part 50.

3.7 - LIMITING CONDITIONS FOR OPERATION

4.7 - SURVEILLANCE REQUIREMENTS

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

c. Requires the quantity of gas to be bled from the containment during the supplemental test to be equivalent to at least 25% of the total measured leakage at $\geq P_a$ (48 psig).

4. Type B and C tests shall be conducted with gas at $\geq P_a$ (48 psig) at intervals in accordance with 10 CFR 50, Appendix J, as modified by approved exemptions, except for tests involving:

   a. Air locks which shall be leak tested in accordance with Surveillance Requirement 4.7.C,

   b. Main steam line isolation valves[a] which shall be leak tested at $\geq P_t$ (25 psig)[a], and

   c. Bolted double-gasketed seals which shall be leak tested at $\geq P_a$ (48 psig) following each closure of the seal and at intervals in accordance with 10 CFR 50, Appendix J, as modified by approved exemptions.

a Exemption from Appendix J to 10CFR Part 50.

3.7 - LIMITING CONDITIONS FOR OPERATION

c. Otherwise, be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

2. With the primary containment air lock interlock mechanism inoperable, restore the air lock interlock mechanism to OPERABLE status within 24 hours, or lock at least one air lock door closed and verify that the door is locked closed at least once per 31 days. Personnel entry and exit through the airlock is permitted provided one OPERABLE air lock door remains locked closed at all times and an individual is dedicated to assure that both air lock doors are not opened simultaneously.

3. With the primary containment air lock inoperable, except as a result of an inoperable air lock door or interlock mechanism, maintain at least one air lock door closed; restore the inoperable air lock to OPERABLE status within 24 hours or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

4.7 - SURVEILLANCE REQUIREMENTS

3.7 - LIMITING CONDITIONS FOR OPERATION

H. Drywell - Suppression Chamber Differential Pressure

Differential pressure between the drywell and the suppression chamber shall be ≥1.0 psid[a].

APPLICABILITY:

OPERATIONAL MODE 1, during the time period:

1. Beginning within 24 hours after THERMAL POWER is >15% of RATED THERMAL POWER following startup, and
2. Ending within 24 hours prior to reducing THERMAL POWER to ≤15% of RATED THERMAL POWER preliminary to a scheduled reactor shutdown.

ACTION:

1. With the drywell - suppression chamber differential pressure less than the above limit, restore the required differential pressure within 24 hours or reduce THERMAL POWER to <15% RATED THERMAL POWER within the next 8 hours.
2. With the drywell - suppression chamber differential pressure instrumentation CHANNEL inoperable, restore the inoperable CHANNEL to OPERABLE status within 30 days or reduce THERMAL POWER to <15% RATED THERMAL POWER within the next 8 hours.

4.7 - SURVEILLANCE REQUIREMENTS

H. Drywell - Suppression Chamber Differential Pressure

1. The drywell - suppression chamber differential pressure shall be demonstrated to be within limits by verifying the differential pressure at least once per 12 hours.
2. At least one drywell - suppression chamber differential pressure instrumentation CHANNEL, and at least one drywell pressure and one suppression chamber pressure instrumentation CHANNEL shall be demonstrated OPERABLE by performance of a:
   a. CHANNEL CHECK at least once per 24 hours,
   b. CHANNEL CALIBRATION at least once per 6 months.

a Except for up to 4 hours for required surveillance which reduces the differential pressure.

3.7 - LIMITING CONDITIONS FOR OPERATION

I. DELETED

4.7 - SURVEILLANCE REQUIREMENTS

I. DELETED

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3.7 - LIMITING CONDITIONS FOR OPERATION

within 1 hour or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

2. In OPERATIONAL MODE(s) 1 or 2 with the suppression pool average water temperature >95°F, except as permitted above, restore the average temperature to ≤95°F within 24 hours or reduce THERMAL POWER to ≤1% RATED THERMAL POWER within the next 12 hours.

3. With the suppression pool average water temperature >105°F during testing which adds heat to the suppression pool, except as permitted above, stop all testing which adds heat to the suppression pool and restore the average temperature to ≤95°F within 24 hours or reduce THERMAL POWER to ≤1% RATED THERMAL POWER within the next 12 hours.

4. With the suppression pool average water temperature >110°F, immediately place the reactor mode switch in the Shutdown position and operate at least one residual heat removal loop in the suppression pool cooling mode.

5. With the suppression pool average water temperature >120°F, depressurize the reactor pressure vessel to <150 psig (reactor steam dome pressure) within 12 hours.

4.7 - SURVEILLANCE REQUIREMENTS

3. Deleted.

4. Deleted.

5. At least once per 18 months by conducting a drywell to suppression chamber bypass leak test at an initial differential pressure of 1.0 psid and verifying that the measured leakage is within the specified limit. If any drywell to suppression chamber bypass leak test fails to meet the specified limit, the test schedule for subsequent tests shall be reviewed and approved by the Commission. If two consecutive tests fail to meet the specified limit, a test shall be performed at least every 9 months until two consecutive tests meet the specified limit, at which time the 18 month test schedule may be resumed.

3.7 - LIMITING CONDITIONS FOR OPERATION

N. SECONDARY CONTAINMENT INTEGRITY

SECONDARY CONTAINMENT INTEGRITY shall be maintained.

APPLICABILITY:

OPERATIONAL MODE(s) 1, 2, 3 and *.

ACTION:

1. Without SECONDARY CONTAINMENT INTEGRITY in OPERATIONAL MODES(s) 1, 2 or 3, restore SECONDARY CONTAINMENT INTEGRITY within 4 hours or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

2. Without SECONDARY CONTAINMENT INTEGRITY in OPERATIONAL MODE *, suspend handling of irradiated fuel in the secondary containment, CORE ALTERATION(s), and operations with a potential for draining the reactor vessel. The provisions of Specification 3.0.C are not applicable.

4.7 - SURVEILLANCE REQUIREMENTS

N. SECONDARY CONTAINMENT INTEGRITY

SECONDARY CONTAINMENT INTEGRITY shall be demonstrated by:

1. Verifying at least once per 24 hours that the pressure within the secondary containment is ≥0.25 inches of vacuum water gauge.

2. Verifying at least once per 31 days that:

   a. At least one door in each secondary containment air lock is closed.

   b. All secondary containment penetrations[a] not capable of being closed by OPERABLE secondary containment automatic isolation dampers and required to be closed during accident conditions are closed.

3. At least once per 18 months by operating one standby gas treatment subsystem at a flow rate ≤4000 cfm for one hour and maintaining ≥0.25 inches of vacuum water gauge in the secondary containment.

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* When handling irradiated fuel in the secondary containment, during CORE ALTERATION(s), and operations with a potential for draining the reactor vessel.

a Valves and blind flanges in high-radiation areas may be verified by use of administrative controls. Normally locked or sealed-closed penetrations may be opened intermittently under administrative control.

3.7 - LIMITING CONDITIONS FOR OPERATION

P. Standby Gas Treatment System

Two independent standby gas treatment subsystems shall be OPERABLE.

APPLICABILITY:

OPERATIONAL MODE(s) 1, 2, 3 and *.

ACTION:

1. With one standby gas treatment subsystem inoperable, restore the inoperable subsystem to OPERABLE status within 7 days, or:
   a. In OPERATIONAL MODE(s) 1,2 or 3, be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.
   b. In OPERATIONAL MODE *, suspend handling of irradiated fuel in the secondary containment, CORE ALTERATION(s), and operations with a potential for draining the reactor vessel. The provisions of Specification 3.0.C are not applicable.
2. Deleted.

4.7 - SURVEILLANCE REQUIREMENTS

P. Standby Gas Treatment System

Each standby gas treatment subsystem shall be demonstrated OPERABLE:

1. At least once per 31 days by initiating, from the control room, flow through the HEPA filters and charcoal adsorbers and verifying that the subsystem operates for at least 10 hours with the heaters operating.
2. 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 subsystem by:
   a. Verifying that the subsystem satisfies the in-place penetration and bypass leakage testing acceptance criteria of $<1\%$ 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 4000 cfm $\pm 10\%$.
   b. 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 ASTM-D-3803-89, for a methyl iodide penetration of $<10\%$, when tested at 30°C and 70% relative humidity; and

* When handling irradiated fuel in the secondary containment, during CORE ALTERATION(s), and operations with a potential for draining the reactor vessel.

3.7 - LIMITING CONDITIONS FOR OPERATION

3. Deleted.

4.7 - SURVEILLANCE REQUIREMENTS

c. Verifying a subsystem flow rate of 4000 cfm ±10% during system operation when tested in accordance with ANSI N510-1980.

3. After every 1440 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 ASTM-D-3803-89, for a methyl iodide penetration of <10%, when tested at 30°C and 70% relative humidity.

4. At least once per 18 months by:

   a. Verifying that the pressure drop across the combined HEPA filters and charcoal adsorber banks is <6 inches water gauge while operating the filter train at a flow rate of 4000 cfm ±10%.

   b. Verifying that the filter train starts and isolation dampers open on each of the following test signals:

      1) Manual initiation from the control room, and

      2) Simulated automatic initiation signal.

   c. Verifying that the heaters dissipate 30 ±3 kw when tested in accordance with ANSI N510-1989. This reading shall include the appropriate correction for variations from 480 volts at the bus.

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which temporarily increases the suppression chamber pressure and reduces the differential pressure. Only one direct suppression chamber to drywell differential pressure instrumentation CHANNEL is provided. However, any pair of the redundant drywell and suppression chamber pressure instrumentation CHANNEL(s) are sufficient to determine the differential pressure.

3/4.7.I DELETED

3/4.7.J Primary Containment Oxygen Concentration

All nuclear reactors must be designed to withstand events that generate hydrogen either due to the zirconium metal-water reaction in the core or due to radiolysis. The primary method to control hydrogen is to inert the primary containment. With the primary containment inerted, that is, oxygen concentration less than 4.0 volume percent, a combustible mixture cannot be present in the primary containment for any hydrogen concentration. The Design Basis Accident (DBA) loss-of-coolant accident (LOCA) analysis assumes that the primary containment is inerted when the DBA occurs. Thus, the hydrogen assumed to be released to the primary containment as a result of a metal-water reaction in the reactor core will not produce combustible gas mixtures in the primary containment.

The primary containment oxygen concentration must be within the specified limit when primary containment is inerted, except as allowed by the relaxations during startup and shutdown. The primary containment must be inert in OPERATIONAL MODE 1, since this is the condition with the highest probability of an event that could produce hydrogen. Inerting the primary containment is an operational problem because it prevents containment access without an appropriate breathing apparatus. Therefore, the primary containment is inerted as late as possible in the plant startup and de-inerted as soon as possible in the plant shutdown. As long as reactor power is below 15% of RATED THERMAL POWER, the potential for an event that generates significant hydrogen is low and the primary containment does not need to be inert. Furthermore, the probability of an event that generates hydrogen occurring within the first 24 hours of a reactor startup or within the last 24 hours before a shutdown is low enough that these windows, when the primary containment is not inerted, are also justified. The 24 hour time frame is a reasonable amount of time to allow plant personnel to perform inerting or de-inerting.

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3/4.7.N SECONDARY CONTAINMENT INTEGRITY

The function of the secondary containment is to isolate and contain fission products that escape from primary containment following a Design Basis Accident (DBA), to confine the postulated release of radioactive material within the requirements of 10CFR Part 100, and to isolate and contain fission products that are released during certain operations that take place inside primary containment, when primary containment is not required to be OPERABLE, or that take place outside of primary containment. The reactor building and associated structures provide secondary containment during normal operation when the drywell is sealed and in service. At other times the drywell may be open and, when required, secondary containment integrity is specified. There are two principal accidents for which credit is taken for secondary containment OPERABILITY. These are a LOCA and fuel-handling accident inside secondary containment. The secondary containment performs no active function in response to each of these limiting events; however, its leak tightness is required to limit offsite radiation doses to below those required by 10CFR Part 100. Maintaining secondary containment OPERABLE ensures that the release of radioactive materials from the primary containment is restricted to those leakage paths and associated leakage rates assumed in the accident analysis and that fission products entrapped within the secondary containment structure will be treated prior to discharge to the environment. Establishing and maintaining a vacuum in the reactor building with the standby gas treatment system during testing, along with the surveillance of the doors, hatches, dampers and valves, is adequate to ensure that there are no violations of the integrity of the secondary containment. This surveillance is normally conducted during periods of calm winds (<5 mph), but may be conducted under higher wind conditions with appropriate application of correction factors.

Valves and blind flanges located in high radiation areas may be verified by use of administrative controls. Allowing verification by administrative controls is considered acceptable, since access to these areas is typically restricted during MODES 1, 2, and 3 for ALARA reasons. Therefore, the probability of misalignment of these SCIVs, once they have been verified to be in the proper position, is low. Normally locked or sealed closed penetrations may be opened intermittently under administrative controls. These administrative controls consist of stationing a dedicated operator, who is in continuous communication with the control room, at the controls of the penetration. In this way, the penetration can be rapidly isolated when a valid secondary containment isolation signal is indicated.

3/4.7.O Secondary Containment Automatic Isolation Dampers

The function of the secondary containment ventilation system automatic isolation dampers, in combination with other accident-mitigation systems, is to limit fission-product release during and following postulated Design Basis Accidents (DBA) such that offsite radiation exposures are maintained within the requirements of 10CFR Part 100. Secondary containment isolation ensures that fission products that escape from primary containment following a DBA, or which are released during certain operations when primary containment is not required, or take place outside primary containment, are maintained within applicable limits. The OPERABILITY requirements for the secondary containment ventilation system isolation dampers help ensure that adequate secondary

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containment leak tightness is maintained during and after an accident by minimizing potential paths to the environment.

3/4.7.P Standby Gas Treatment System

The standby gas treatment system (SBGT) is required to ensure that radioactive materials that leak from the primary containment into the secondary containment following a Design Basis Accident (DBA) are filtered and adsorbed prior to exhausting to the environment. This system reduces the potential releases of radioactive material, principally iodine, to within values specified in 10CFR Part 100.

The standby gas treatment system is designed to filter and exhaust the reactor building atmosphere to the main chimney during secondary containment isolation conditions, with a minimum release of radioactive materials from the reactor building to the environment. One standby gas treatment fan is designed to automatically start upon secondary containment isolation and to maintain the reactor building pressure to approximately a negative ¼ inch water gauge pressure; all leakage should be in-leakage. Should the fan fail to start, the redundant alternate fan and filter subsystem is designed to start automatically.

The OPERABILITY of the standby gas treatment system reduces the potential release of radioactive material, principally iodine, following a design basis accident. The reduction in containment iodine inventory reduces the resulting site boundary radiation doses associated with containment leakage. The operation of this system and resultant iodine removal capacity are consistent with the assumptions used in the LOCA analyses. Periodic operation of the system with the heaters is sufficient to reduce the buildup of moisture on the adsorbers and HEPA filters.

Since the standby gas treatment subsystems are shared by both units, one subsystem is powered by the unit diesel generator power source of each unit. The emergency power supply OPERABILITY requirements for the standby gas treatment system are addressed within Specification 3.9.A, Actions. For example, if conducting the alternate offsite power source cross-tie surveillance were to require the inoperability of both unit diesel generator power sources, neither of the standby gas treatment subsystems would have an OPERABLE diesel generator power source and the appropriate ACTION would have to be entered.

3.8 - LIMITING CONDITIONS FOR OPERATION

A. Residual Heat Removal Service Water System

At least the following independent residual heat removal service water (RHRSW) subsystems, with each subsystem comprised of:

1. Two OPERABLE RHRSW pumps, and
2. An OPERABLE flow path capable of taking suction from the ultimate heat sink and transferring the water:
   a. Through one RHR heat exchanger, and separately,
   b. To the associated safety related equipment,

shall be OPERABLE:

1. In OPERATIONAL MODE(s) 1, 2 and 3, two subsystems.
2. In OPERATIONAL MODE(s) 4, 5 and * the subsystem(s) associated with subsystems/loops and components required OPERABLE by Specifications 3.6.O, 3.6.P, 3.8.D, 3.10.K and 3.10.L.

APPLICABILITY:

OPERATIONAL MODE(s) 1, 2, 3, 4, 5 and *.

4.8 - SURVEILLANCE REQUIREMENTS

A. Residual Heat Removal Service Water System

Each of the required RHRSW subsystems shall be demonstrated OPERABLE at least once per 31 days by verifying that each valve, manual or power operated, in the flow path that is not locked, sealed or otherwise secured in position, is in its correct position.

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* When handling irradiated fuel in the secondary containment, during CORE ALTERATION(s), and operations with a potential for draining the reactor vessel.

3.8 - LIMITING CONDITIONS FOR OPERATION

D. Control Room Emergency Ventilation System

The control room emergency ventilation system shall be OPERABLE, with the system comprised of an OPERABLE control room emergency filtration system and an OPERABLE refrigeration control unit (RCU).

APPLICABILITY:

OPERATIONAL MODE(s) 1, 2, 3, and *.

ACTION:

1. In OPERATIONAL MODE(s) 1, 2 or 3:

   a. With the control room emergency filtration system inoperable, restore the inoperable system to OPERABLE status within 7 days or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

   b. With the refrigeration control unit (RCU) inoperable, restore the inoperable RCU to OPERABLE status within 30 days or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

4.8 - SURVEILLANCE REQUIREMENTS

D. Control Room Emergency Ventilation System

The control room emergency ventilation system shall be demonstrated OPERABLE:

1. At least once per 18 months by verifying that the RCU has the capability to remove the required heat load.

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

3. 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:

   a. Verifying that the system satisfies the in-place penetration and bypass leakage testing acceptance criteria of <0.05% 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 scfm ±10%.

* When handling irradiated fuel in the secondary containment, during CORE ALTERATION(s), and operations with a potential for draining the reactor vessel.

3.8 - LIMITING CONDITIONS FOR OPERATION

2. In OPERATIONAL MODE *, with the control room emergency filtration system or the RCU inoperable, immediately suspend CORE ALTERATION(s), handling of irradiated fuel in the secondary containment and operations with a potential for draining the reactor vessel.

3. The provisions of Specification 3.0.C are not applicable in OPERATIONAL MODE *.

4.8 - SURVEILLANCE REQUIREMENTS

b. 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 laboratorytesting criteria of ASTM-D-3803-89, for a methyl iodide penetration of <0.50%, when tested at 30°C and 70% relative humidity; and

c. Verifying a system flow rate of 2000 scfm ±10% during system operation when tested in accordance with ANSI N510-1980.

4. After every 1440 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 ASTM-D-3803-89, for a methyl iodide penetration of <0.50%, when tested at 30°C and 70% relative humidity.

5. At least once per 18 months by:

   a. Verifying that the pressure drop across the combined HEPA filters and charcoal adsorber banks is <6 inches water gauge while operating the filter train at a flow rate of 2000 scfm ±10%.

3.8 - LIMITING CONDITIONS FOR OPERATION

4.8 - SURVEILLANCE REQUIREMENTS

b. Verifying that the isolation dampers close on each of the following signals:

    1) Manual initiation from the control room, and

    2) Simulated automatic isolation signal.

c. Verifying that during the pressurization mode of operation, control room positive pressure is maintained at ≥1/8 inch water gauge relative to adjacent areas during system operation at a flow rate ≤2000 scfm.

d. Verifying that the heaters dissipate 12 ±1.2 kw when tested in accordance with ANSI N510-1980. This reading shall include the appropriate correction for variations from 480 volts at the bus.

6. After each complete or partial replacement of an HEPA filter bank by verifying that the HEPA filter bank satisfies the in-place penetration and leakage testing acceptance criteria of <0.05% in accordance with ANSI N510-1980 while operating the system at a flow rate of 2000 scfm ±10%.

7. After each complete or partial replacement of an charcoal adsorber bank by verifying that the charcoal adsorber bank satisfies the in-place penetration and leakage testing acceptance criteria of <0.05% in accordance with ANSI N510-1980 for a halogenated hydrocarbon refrigerant test gas while operating the system at flow rate of 2000 scfm ±10%.

3.8 - LIMITING CONDITIONS FOR OPERATION

4.8 - SURVEILLANCE REQUIREMENTS

the snubber is required to be OPERABLE. The parts replacements shall be documented and the documentation shall be retained.

3.8 - LIMITING CONDITIONS FOR OPERATION

I. Main Condenser Offgas Activity

The release rate of the sum of the activities of the noble gases measured prior to the offgas holdup line shall be limited to ≤100 μCi/sec/MWt, after 30 minutes decay.

APPLICABILITY:

OPERATIONAL MODE(s) 1, 2[a] and 3[a].

ACTION:

With the release rate of the sum of the activities of the noble gases in the main condenser air ejector effluent (as measured prior to the offgas holdup line) >100 μCi/sec/MWt, after 30 minutes decay, restore the release rate to within its limit within 72 hours or be in at least STARTUP with the main steam isolation valves closed within the next 8 hours.

4.8 - SURVEILLANCE REQUIREMENTS

I. Main Condenser Offgas Activity

1. The release rate of noble gases from the main condenser air ejector shall be continuously monitored in accordance with the ODCM.

2. The release rate of the sum of the activities from noble gases from the main condenser air ejector shall be determined to be within the limits of Specification 3.8.I at the following frequencies[b] by performing an isotopic analysis of a representative sample of gases taken at the recombiner outlet, or the air ejector outlet, if the recombiner is bypassed:

   a. At least once per 31 days, and

   b. Within 4 hours following the determination of an increase of >50%.

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a When the main condenser air ejector is in operation.

b The provisions of Specification 4.0.D are not applicable.

3.8 LIMITING CONDITIONS FOR OPERATION

J. Safe Shutdown Makeup Pump

The Safe Shutdown Makeup Pump (SSMP) shall be OPERABLE.

APPLICABILITY:

OPERATIONAL MODE(s) 1, 2 and 3 with reactor steam dome pressure greater than 150 psig.

ACTION:

1. With the SSMP system inoperable, restore the inoperable SSMP system to OPERABLE status within 67 days, or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

4.8 - SURVEILLANCE REQUIREMENTS

J. Safe Shutdown Makeup Pump

The SSMP system shall be demonstrated OPERABLE:

1. At least once per 31 days by:
   a. 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. Verifying that the pump flow controller is in the correct position.
2. At least once per 92 days by verifying that the SSMP develops a flow of greater than or equal to 400 gpm against a system head corresponding to reactor vessel pressure of greater than 1120 psig.

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3/4.8.A Residual Heat Removal Service Water System

The residual heat removal service water system, with the ultimate heat sink, provides sufficient cooling capacity for continued operation of the residual heat removal system and of other safety-related equipment, e.g., RHRSW vault coolers and the control room emergency ventilation system refrigeration units, during normal and accident conditions. The redundant cooling capacity of the system, assuming a single failure, is consistent with the assumptions used in the safety analysis to keep the accident conditions within acceptable limits. Since only one of the four pumps is required to provide the necessary cooling capacity, a thirty day repair period is allowed for one pump out of service. OPERABILITY of this system is also dependent upon special measures for protection from flooding in the condenser pit area.

3/4.8.B Diesel Generator Cooling Water System

The diesel generator cooling water system, with the ultimate heat sink, provides sufficient cooling capacity for continued operation of the diesel generators during normal and accident conditions. The cooling capacity of the system is consistent with the assumptions used in the safety analysis to keep the accident conditions within acceptable limits. OPERABILITY of this system is also dependent upon special measures for protection from flooding in the condenser pit area.

3/4.8.C Ultimate Heat Sink

The Mississippi River provides an ultimate heat sink with sufficient cooling capacity to either provide normal cooldown of the units, or to mitigate the effects of accident conditions within acceptable limits for one unit while conducting a normal cooldown on the other unit.

3/4.8.D Control Room Emergency Ventilation System

The control room emergency filtration system maintains habitable conditions for operations personnel during and following all design basis accident conditions. This system, in conjunction with control room design, is based on limiting the radiation exposure to personnel occupying the room to five rem or less whole body, or its equivalent.

The frequency of tests and sample analysis is necessary to show that the HEPA filters and charcoal adsorbers can perform as evaluated. The control room emergency filtration system in-place testing procedures are established utilizing applicable sections of ANSI N510-1980 standard. Operation of the system with the heaters OPERABLE for ten hours a month is sufficient to reduce the buildup of moisture on the adsorbers and HEPA filters. The charcoal adsorber efficiency test procedures allow for the removal of one representative sample cartridge and testing in accordance with the guidelines of ASTM-D-3803-89. The sample is at least two inches in diameter and has a length equivalent to the thickness of the bed. If the iodine removal efficiency test results are

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unacceptable, all adsorbent in the system is replaced. HEPA filter particulate removal efficiency is verified to be at least 99% by in-place testing with a DOP testing medium.

The control room refrigeration control unit (RCU) provides conditioned air for personnel comfort, safety and equipment reliability. The testing of the control room RCU system verifies that the heat-removal capability of the system is sufficient to remove sufficient heat load from the control room such that the control room air temperature is ≤ 95 °F. The test frequency is appropriate since significant degradation of the control room RCU system is not expected over this time period.

3/4.8.E Flood Protection

Flood protection measures are provided to protect the systems and equipment necessary for safe shutdown during high water conditions. The equipment necessary to implement the appropriate measures, as detailed in plant procedures, is required to be available, but not necessarily onsite, to implement the procedures in a timely manner. The selected water levels are based on providing timely protection from the design basis flood of the river.

3/4.8.F Snubbers

Mechanical snubbers are provided to ensure that the structural integrity of the reactor coolant system and all other safety-related systems is maintained during and following a seismic event or other event initiating dynamic loads. Snubbers are classified and grouped by design, manufacturer and accessibility. A list of individual snubbers with information of snubber location, classification or group, and system affected is maintained at the plant. The accessibility of each snubber is determined and documented for each snubber. The determination is based upon the existing radiation levels and the expected time to perform a visual inspection in each snubber location as well as other factors associated with accessibility during plant operation (e.g., temperature, atmosphere, location, etc.), and the recommendations of Regulatory Guides 8.8 and 8.10.

The visual inspection frequency is based upon maintaining a constant level of snubber protection to the systems. Therefore, the required inspection interval varies with the number of unacceptable snubbers found during the previous inspection, the total population or category size for each snubber type, and the previous inspection interval. A snubber is considered unacceptable if it fails to satisfy the acceptance criteria of the visual inspection. Snubbers may be categorized, based upon their accessibility during power operation, as accessible or inaccessible. These categories may be examined separately or jointly as determined and documented prior to the inspections. The categorization is used as the basis for determining the next inspection interval for that category.

BASES

If a review and evaluation can not justify continued operation with an unacceptable snubber, the snubber is declared inoperable and the applicable action taken. To determine the next surveillance interval, the unacceptable snubber may be reclassified as acceptable if it can be demonstrated that the snubber is OPERABLE in its as-found condition by the performance of a functional test. The next visual inspection interval may be twice, the same, or reduced by as much as two-thirds of the previous inspection interval, depending on the number of unacceptable snubbers found in proportion to the size of the population or category for each type of snubber included in the previous inspection. The inspection interval may be as long as 48 months and the provisions of Specification 4.0.B may be applied.

When a snubber is found to be inoperable, an engineering evaluation is performed, in addition to the determination of the snubber mode of failure, in order to determine if any safety-related component or system has been adversely affected by the inoperability of the snubber. The engineering evaluation shall determine whether or not the snubber mode of failure has imparted a significant effect or degradation on the supported component or system.

To provide additional assurance of snubber functional reliability , a representative sample of the installed snubbers will be functionally tested at 18 month intervals. This sample is identified using one of three methods:

1. Functionally test 10% of a type of snubber with an additional 10% tested for each functional testing failure, or

2. Functionally test a sample size and determine sample acceptance or rejection using Figure 4.8.F-1, or

3. Functionally test a representative sample size and determine sample acceptance or rejection using the stated equation.

Figure 4.8.F-1 was developed using "Wald's Sequential Probability Ratio Plan" as described in "Quality Control and Industrial Statistics" by Acheson J. Duncan.

Permanent or other exemptions from the surveillance program for individual snubbers may be granted by the NRC if a justifiable basis for exemption is presented and, if applicable, snubber life destructive testing was performed to qualify the snubber for the applicable design conditions at either the completion of their fabrication or at a subsequent date. Snubbers so exempted are listed in the list of individual snubbers indicating the extent of the exemptions.

The service life of a snubber is established via manufacturer input and information through consideration of the snubber service conditions and associated installation and maintenance records (newly installed snubbers, seal replace, spring replaced, in high radiation area, in high temperature area, etc.). The requirement to monitor the snubber service life is included to ensure that the snubbers periodically undergo a performance evaluation in view of their age and operating conditions. These records provide statistical bases for future consideration of snubber service life.

BASES

3/4.8.G Sealed Source Contamination

The limitations on removable contamination for sources requiring leak testing, including alpha emitters, is based on 10 CFR 70.39(c) limits for plutonium. This limitation will ensure that leakage from byproduct, source, and special nuclear material sources will not exceed allowable intake values. Sealed sources, including startup sources and fission detectors, are classified into three groups according to their use, with surveillance requirements commensurate with the probability of damage to a source in that group. Those sources which are frequently handled are required to be tested more often than those which are not. Sealed sources which are continuously enclosed within a shielded mechanism, i.e., sealed sources within radiation monitoring or boron measuring devices, are considered to be stored and need not be tested unless they are removed from the shielded mechanism.

3/4.8.H Explosive Gas Mixture

This specification is provided to ensure that the concentration of potentially explosive gas mixtures contained in the offgas holdup system is maintained below the flammability limits of hydrogen and oxygen. Maintaining the concentration of hydrogen and oxygen below their flammability limits provides assurance that the releases of radioactive materials will be controlled in conformance with the requirements of General Design Criterion 60 of Appendix A to 10CFR Part 50.

3/4.8.I Main Condenser Offgas Activity

Restricting the gross radioactivity rate of noble gases from the main condenser provides reasonable assurance that the total body exposure to an individual at the exclusion area boundary will not exceed a small fraction of the limits of 10CFR Part 100 in the event this effluent is inadvertently discharged directly to the environment without treatment. This specification implements the requirements of General Design Criteria 60 and 64 of Appendix A to 10CFR Part 50. The release rates are determined at a more expeditious frequency following the determination of an increase of greater than 50%, as indicated by the air ejector noble gas monitor, after factoring out increases due to changes in THERMAL POWER level and off-gas flow in the nominal steady-state fission gas release from the primary coolant.

3/4.8.J Safe Shutdown Makeup Pump System (SSMP)

The SSMP system provides a common backup to the Unit 1 and 2 RCIC systems to satisfy the requirements of 10 CFR 50, Appendix R, Section III.G, "Fire Protection of Safe Shutdown Capability." The system bypasses fire zones which could theoretically disable the RCIC system.

In the event that the reactor vessel becomes isolated, and the feedwater supply becomes unavailable, makeup can be provided by manually initiating the SSMP system to supply demineralized makeup water from the CCST or as an alternate source, makeup water from the fire

BASES

header. The flow rate of the SSMP system is approximately equal to the reactor water boil-off rate 15 minutes after shutdown.

The SSMP system is required to be OPERABLE when either Unit 1 or Unit 2 is in OPERATIONAL MODE(s) 1, 2 or 3 with reactor steam dome pressure greater than 150 psig. With the SSMP system inoperable, a 67-day allowable out-of-service (AOT) is provided to restore the inoperable system to OPERABLE status before the Unit(s) must be shut down. (Reference: Fire Protection Plan Documentation Package (FPPDP), "Fire Protection Reports," Volume 2, Tab 4, Safe Shutdown Analysis.)

The surveillance requirements provide adequate assurance that the SSMP system will be OPERABLE when required. A design flow test can be performed during plant operation using a full flow test return line to the CCST.

3.9 - LIMITING CONDITIONS FOR OPERATION

A. A.C. Sources - Operating

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

1. Two physically independent circuits between the offsite transmission network and the onsite Class 1E distribution system, and

2. Two separate and independent diesel generators, each with:

   a. A separate fuel oil day tank containing ≥205 gallons of available fuel,

   b. A separate bulk fuel storage system containing ≥10,000 gallons of available fuel, and

   c. A separate fuel oil transfer pump.

APPLICABILITY:

OPERATIONAL MODE(s) 1, 2, and 3.

ACTION:

1. With one of the above required offsite circuit power sources inoperable:

   a. Demonstrate the OPERABILITY of the remaining offsite circuit by performing Surveillance Requirement 4.9.A.1.a within 1 hour and at least once per 8 hours thereafter.

4.9 - SURVEILLANCE REQUIREMENTS

A. A.C Sources - Operating

1. Each of the required independent circuits between the offsite transmission network and the onsite Class 1E distribution system shall be determined OPERABLE:

   a. At least once per 7 days by verifying correct breaker alignments and indicated power availability, and

   b. At least once per 18 months by manually transferring the power supply from the normal circuit to the alternate circuit.

2. Each of the required diesel generators shall be demonstrated OPERABLE[a] at least once per 31 days by:

   a. Verifying the fuel levels in both the fuel oil day tank and the bulk fuel storage tank.

   b. Verifying the fuel transfer pump starts and transfers fuel from the bulk fuel storage system to the fuel oil day tank.

---

a All diesel generator starts may be preceded by an engine prelube period. All diesel generator starts that require loading may be preceded by an engine prelube period and followed by a warmup period prior to loading. Diesel generator loadings may include gradual loading as recommended by the manufacturer/vendor.

3.9 - LIMITING CONDITIONS FOR OPERATION

b. Restore the inoperable offsite circuit to OPERABLE status within 7 days or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

2. With one of the above required diesel generator power sources inoperable:

   a. Demonstrate the OPERABILITY of the offsite circuit power sources by performing Surveillance Requirement 4.9.A.1.a within 1 hour and at least once per 8 hours thereafter.

   b. If the diesel generator is inoperable due to any cause other than an inoperable support system, an independently testable component, or preplanned maintenance or testing, demonstrate the OPERABILITY of the remaining OPERABLE diesel generator by performing Surveillance Requirement 4.9.A.2.c[b] within 24 hours unless the absence of any potential common mode failure for the remaining diesel generator is demonstrated (if it has not been successfully tested within the past 24 hours) and within the subsequent 72 hours, and

4.9 - SURVEILLANCE REQUIREMENTS

c. Verifying[c] the diesel starts and accelerates to synchronous speed with generator voltage and frequency at 4160 ± 420 volts and 60 ± 1.2 Hz, respectively.

d. Verifying the diesel generator is synchronized, loaded to between 2375 and 2500 kW[d] in accordance with the manufacturer's/vendor's recommendations, and operates with this load for ≥ 60 minutes.

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

f. Verifying the pressure in required starting air receiver tanks to be ≥ 230 psig.

3. Each of the required diesel generators shall be demonstrated OPERABLE at least once per 31 days and after each operation of the diesel where the period of operation was ≥ 1 hour by removing any accumulated water from the day tank.

4. Each of the required diesel generators shall be demonstrated OPERABLE at least once per 92 days by checking for and removing accumulated water from the fuel oil bulk storage tanks.

---

b Contrary to the provisions of Specification 3.0.B, this test is required to be completed regardless of when the inoperable diesel generator is restored to OPERABILITY for failures that are potentially generic to the remaining diesel generator and for which appropriate alternative testing cannot be designed.

c ~~Surveillance Requirement 4.9.A.7 may be substituted for Surveillance Requirement 4.9.A.2.c.~~

d Momentary transients outside of the load range do not invalidate this test. Diesel generator loadings may include gradual loading as recommended by the manufacturer/vendor. This surveillance shall be conducted on only one diesel generator at a time.

3.9 - LIMITING CONDITIONS FOR OPERATION

c. Restore the diesel generator to OPERABLE status within 7 days or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

3. With one of the above offsite circuit power sources and one of the above required diesel generator power sources inoperable:

   a. Demonstrate the OPERABILITY of the remaining offsite circuit power source by performing Surveillance Requirement 4.9.A.1.a within 1 hour and at least once per 8 hours thereafter.

   b. If the diesel generator is inoperable due to any cause other than preplanned maintenance or testing, demonstrate the OPERABILITY[e] of the remaining OPERABLE diesel generator by performing Surveillance Requirement 4.9.A.2.c[b] within 8 hours unless the absence of any potential common mode failure for the remaining diesel generator is demonstrated (if it has not been successfully tested within the past 24 hours) and within the subsequent 72 hours for each OPERABLE diesel generator.

4.9 - SURVEILLANCE REQUIREMENTS

5. Each of the required diesel generators shall be demonstrated OPERABLE by:

   a. Sampling new fuel oil prior to addition to the storage tanks in accordance with applicable ASTM standards, and

   b. Verifying prior to addition to the storage tanks that the sample meets the applicable ASTM standards for API gravity, water and sediment, and the visual test for free water and particulate contamination[h], and

   c. Verifying within 31 days of obtaining the sample that the kinematic viscosity is within applicable ASTM limits.

6. Each of the required diesel generators shall be demonstrated OPERABLE by:

   a. Sampling and analyzing the bulk fuel storage tanks at least once per 31 days in accordance with applicable ASTM standards, and

   b. Verifying that the sample meets the applicable ASTM standards for water and sediment, kinematic viscosity, and ASTM particulate contaminant[h] is <10 mg/liter.

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e A successful test of OPERABILITY per Surveillance Requirement 4.9.A.2.c under this ACTION statement satisfies the diesel generator test requirements of ACTION(s) 1 or 2 above.

b Contrary to the provisions of Specification 3.0.B, this test is required to be completed regardless of when the inoperable diesel generator is restored to OPERABILITY for failures that are potentially generic to the remaining diesel generator and for which appropriate alternative testing cannot be designed.

h The particulate contamination surveillance is not required for No. 1 fuel oil. It is required for No. 2 fuel oil and for blends.

3.9 - LIMITING CONDITIONS FOR OPERATION

c. Restore at least one of the inoperable A.C. power sources to OPERABLE status within 12 hours or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours, and

d. Restore both offsite circuits and both diesel generators to OPERABLE status within 7 days from the time of the initial loss or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

4. With one of the above required diesel generator power sources inoperable, in addition to ACTION 2 or 3, as applicable:

   a. Verify within 2 hours that at least one of the required two systems, subsystems, trains, components and devices in two train systems is OPERABLE including its emergency power supply.

   b. Otherwise, take the applicable ACTIONs for both systems, subsystems, trains, components or devices inoperable, or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

4.9 - SURVEILLANCE REQUIREMENTS

7. Each of the required diesel generators shall be demonstrated OPERABLE[a] at least once per 184 days by verifying[c] the diesel starts and accelerates to synchronous speed in ≤10 seconds. The generator voltage and frequency shall be verified to reach 4160 ±420 volts and 60 ±1.2 Hz, respectively, in ≤10 seconds after the start signal.

8. Each of the required diesel generators shall be demonstrated OPERABLE[a] at least once per 18 months by:

   a. Deleted.

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a All diesel generator starts may be preceded by an engine prelube period. All diesel generator starts that require loading may be preceded by an engine prelube period and followed by a warmup period prior to loading. Diesel generator loadings may include gradual loading as recommended by the manufacturer/vendor.

c Surveillance Requirement 4.9.A.7 may be substituted for Surveillance Requirement 4.9.A.2.c.

3.9 - LIMITING CONDITIONS FOR OPERATION

5. With two of the above required offsite circuit power sources inoperable:

   a. Restore at least one of the inoperable offsite circuits to OPERABLE status within 24 hours or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

   b. Restore at least two offsite circuits to OPERABLE status within 7 days from the time of initial loss or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

6. With both of the above required diesel generator power sources inoperable:

   a. Demonstrate the OPERABILITY of the offsite circuit power sources by performing Surveillance Requirement 4.9.A.1.a within 1 hour and at least once per 8 hours thereafter.

4.9 - SURVEILLANCE REQUIREMENTS

b. Verifying the diesel generator capability to reject its largest single emergency load (≥725 kW) while maintaining speed ≤1001 rpm and voltage at 4160 ±420 volts.

c. Verifying the diesel generator capability to reject a load between 2375 and 2500 kW[d], without tripping on overspeed. The generator voltage shall not exceed 5000 volts during or following the load rejection.

d. Simulating a loss of offsite power by itself, and:

   1) Verifying de-energization of the emergency buses, and load shedding from the emergency buses.

   2) Verifying the diesel starts on the auto-start signal, energizes the emergency buses with permanently connected loads in ≤10 seconds, energizes the auto-connected shutdown loads, and operates with this load for ≥5 minutes. After energization, the steady-state voltage and frequency of the emergency busses shall be maintained at 4160 ±420 volts and 60 ±1.2 Hz, respectively, during this test.

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d Momentary transients outside of the load range do not invalidate this test. Diesel generator loadings may include gradual loading as recommended by the manufacturer/vendor. This surveillance shall be conducted on only one diesel generator at a time.

3.9 - LIMITING CONDITIONS FOR OPERATION

b. Within 2 hours, restore at least one of the above required diesel generators to OPERABLE[e] status and verify that at least one of the required two systems, subsystems, trains, components and devices in two train systems is OPERABLE including its emergency power supply. Otherwise, take the applicable ACTIONs for both systems, subsystems, trains, components or devices inoperable, or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

c. Demonstrate the continued OPERABILITY of the restored diesel generator by performing Surveillance Requirement 4.9.A.2.c within the subsequent 72 hours, and

d. Restore at least two required diesel generators to OPERABLE status within 7 days from the time of initial loss or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

7. With the fuel oil contained in the bulk fuel storage tank(s) not meeting the properties specified in Surveillance Requirements 4.9.A.5 and 4.9.A.6, restore the fuel oil properties to within the specified limits within 7 days. Otherwise, declare the associated diesel generator(s) inoperable.

4.9 - SURVEILLANCE REQUIREMENTS

e. Verifying that on an ECCS actuation test signal, without loss of offsite power, the diesel generator starts on the auto-start signal and operates on standby for ≥5 minutes. The generator voltage and frequency shall be 4160 ±420 volts and 60 ±1.2 Hz, respectively, in ≤10 seconds after the auto-start signal; the steady state generator voltage and frequency shall be maintained within these limits during this test.

f. Simulating a loss of offsite power in conjunction with an ECCS actuation test signal , and

1) Verifying de-energization of the emergency buses, and load shedding from the emergency buses.

2) Verifying the diesel starts on the auto-start signal, energizes the emergency buses with permanently connected loads in ≤10 seconds, energizes the auto-connected emergency loads through the load sequencer, and operates with this load for ≥5 minutes. After energization, the steady-state voltage and frequency of the emergency busses shall be maintained at 4160 ±420 volts and 60 ±1.2 Hz, respectively, during this test.

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e A successful test of OPERABILITY per Surveillance Requirement 4.9.A.2.c under this ACTION statement satisfies the diesel generator test requirements of ACTION(s) 1 or 2 above.

3.9 - LIMITING CONDITIONS FOR OPERATION

4.9 - SURVEILLANCE REQUIREMENTS

g. Verifying that all automatic diesel generator trips, except engine overspeed and generator differential current are automatically bypassed upon an emergency actuation signal.

h. Verifying the diesel generator operates for ≥24 hours. During the first 2 hours of this test, the diesel generator shall be loaded to between 2625 and 2750 kW[d] and during the remaining 22 hours of this test, the diesel generator shall be loaded to between 2375 and 2500 kW[d]. The generator voltage and frequency shall be 4160 ±420 volts and 60 ±1.2 Hz, respectively, in ≤10 seconds after the start signal; the steady state generator voltage and frequency shall be maintained within these limits during this test. Within 5 minutes after completing this 24 hour test, perform Surveillance Requirement 4.9.A.2.c[f].

i. Verifying that the auto-connected loads to each diesel generator do not exceed the 2000 hour rating of 2850 kW.

d Momentary transients outside of the load range do not invalidate this test. Diesel generator loadings may include gradual loading as recommended by the manufacturer/vendor. This surveillance shall be conducted on only one diesel generator at a time.

f If Surveillance Requirement 4.9.A.2.c is not satisfactorily completed, it is not necessary to repeat the preceding 24 hour test. Instead, the diesel generator may be operated at approximately full load for 2 hours or until the operating temperature has stabilized.

3.9 - LIMITING CONDITIONS FOR OPERATION

4.9 - SURVEILLANCE REQUIREMENTS

j. Verifying the diesel generator's capability to:

   1) synchronize with the offsite power source while the generator is loaded with its emergency loads upon a simulated restoration of offsite power,

   2) transfer its loads to the offsite power source, and

   3) be restored to its standby status.

k. Verifying that the automatic load sequence logic is OPERABLE with the interval between each load block within ±10% of its design interval.

9. Each of the required diesel generators shall be demonstrated OPERABLE[a] at least once per 10 years or after any modifications which could affect diesel generator interdependence by starting both diesel generators simultaneously, and verifying that both diesel generators accelerate to ≥900 rpm in ≤10 seconds.

10. Each of the required diesel generators shall be demonstrated OPERABLE at least once per 10 years by draining each fuel oil storage tank, removing the accumulated sediment and cleaning the tank.

a All diesel generator starts may be preceded by an engine prelube period. All diesel generator starts that require loading may be preceded by an engine prelube period and followed by a warmup period prior to loading. Diesel generator loadings may include gradual loading as recommended by the manufacturer/vendor.

TABLE 4.9.A-1

DIESEL GENERATOR TEST SCHEDULE

(NOT USED)

3.9 - LIMITING CONDITIONS FOR OPERATION

C. D.C. Sources - Operating

As a minimum, the following D.C. electrical power sources shall be OPERABLE with the identified parameters within the limits specified in Table 4.9.C-1:

1. Two station 250 volt batteries, each with a full capacity charger.
2. Two station 125 volt batteries, each with a full capacity charger.

APPLICABILITY:

OPERATIONAL MODE(s) 1, 2, and 3.

ACTION:

1. With one of the above required 250 volt station batteries and/or chargers inoperable, restore the inoperable equipment to OPERABLE status within 72 hours.

4.9 - SURVEILLANCE REQUIREMENTS

C. D.C. Sources - Operating

Each of the required 125 volt and 250 volt batteries and chargers shall be demonstrated OPERABLE[a]:

1. At least once per 7 days by verifying that:
   a. The parameters in Table 4.9.C-1 meet Category A limits, and
   b. There is correct breaker alignment to the battery chargers and total battery terminal voltage is ≥125.9 or ≥260.4 volts, as applicable, on float charge.
2. At least once per 92 days and within 7 days after a battery discharge with a battery terminal voltage below 105 or 210 volts, as applicable, or battery overcharge with battery terminal voltage above 150 or 300 volts, as applicable, by verifying that:
   a. The parameters in Table 4.9.C-1 meet the Category B limits,
   b. There is no visible corrosion at either terminals or connectors, or the connection resistance of these items is ≤150 x10^{-6} ohms or ≤20% above baseline connection resistance, whichever is higher, and
   c. The average electrolyte temperature of all connected cells is above 60°F.

a An alternate 125 volt battery shall adhere to these same Surveillance Requirements to be considered OPERABLE.

3.9 - LIMITING CONDITIONS FOR OPERATION

2. With one of the above required 125 volt station batteries and/or chargers inoperable, within 72 hours[b], either restore the inoperable equipment to OPERABLE status, or place an OPERABLE corresponding alternate 125 volt battery (with an OPERABLE full capacity charger) in service.

3. With the provisions of either ACTION 1 or 2 above not met, be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

4. With any Category A parameter(s) outside the limit(s) shown in Table 4.9.C-1, the battery may be considered OPERABLE provided that its associated charger is OPERABLE, and 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.

5. With any Category B parameter(s) outside the limit(s) shown in Table 4.9.C-1, 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 the limit(s) within 7 days.

4.9 - SURVEILLANCE REQUIREMENTS

3. At least every 18 months by verifying that:

   a. The cells, cell plates and battery racks show no visual indication of physical damage or abnormal deterioration.

   b. The cell-to-cell and terminal connections are clean, tight, free of corrosion and coated with anti-corrosion material.

   c. The resistance of each cell-to-cell and terminal connection is $\leq 150 \times 10^{-6}$ ohms or $\leq 20\%$ above baseline connection resistance, whichever is higher.

   d. The battery chargers will supply a load equal to the manufacturer's rating for at least 4 hours.

4. At least every 18 months, by verifying that the battery capacity is adequate to supply and maintain in OPERABLE status all of the actual or simulated emergency loads for design duty cycle when the battery is subjected to a battery service test.

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b With Unit 1 and 2 in OPERATIONAL MODE(s) 1, 2 or 3, each 125 volt battery may be inoperable for up to a maximum of seven days per operating cycle for maintenance or testing provided the alternate 125 volt battery is placed into service and is OPERABLE. If it is determined that a 125 volt battery need be replaced as a result of maintenance or testing, a specific battery may be inoperable for an additional seven days provided the alternate 125 volt battery is placed into service and is OPERABLE. With the other Unit in MODE(s) 4 or 5, operations may continue with one of the two 125 volt battery systems inoperable provided the alternate 125 volt battery is placed into service and is OPERABLE.

3.9 - LIMITING CONDITIONS FOR OPERATION

6. With any Category B parameter not within its allowable value(s), immediately declare the battery inoperable.

4.9 - SURVEILLANCE REQUIREMENTS

5. At least once per 60 months, verify that the battery capacity is at least the greater of either 80% of the manufacturer's rating or the minimum acceptable battery capacity from the load profile when subjected to either a performance discharge test or a modified performance discharge test. The modified performance discharge test satisfies the requirements of both the service test and the performance test and therefore, may be performed in lieu of a service test.

6. For any battery that shows signs of degradation or has reached 85% of the service life for the expected application and delivers a capacity of less than 100% of the manufacturer's rated capacity, a performance discharge test or a modified performance test of battery capacity shall be performed at least once every 12 months or the battery shall be replaced or restored to 100% or greater of the manufacturer's rated capacity during the next refuel outage. Degradation is indicated when the battery capacity drops more than 10% from its capacity on the previous performance test, or is below 90% of the manufacturer's rating. If the battery has reached 85% of service life, delivers a capacity of 100% or greater of the manufacturer's rated capacity and has shown no signs of degradation, a performance test or a modified performance test of battery capacity shall be performed at least once every two years.

3.9 - LIMITING CONDITIONS FOR OPERATION

E. Distribution - Operating

The following power distribution systems shall be energized:

1. A.C. power distribution, consisting of:

   a. Both Unit engineered safety features 4160 volt buses:

      1) For Unit 1, Nos. 13-1 and 14-1,

      2) For Unit 2, Nos. 23-1 and 24-1.

   b. Both Unit engineered safety features 480 volt buses:

      1) For Unit 1, Nos. 18 and 19,

      2) For Unit 2, Nos. 28 and 29, and

   c. The Unit 120 volt Essential Service Bus and Instrument Bus.

2. 250 volt D.C. power distribution, consisting of:

   a. TB MCC Nos. 1 and 2, and

   b. 1) For Unit 1, RB MCC Nos. 1A and 1B,

      2) For Unit 2, RB MCC Nos. 2A and 2B.

3. For Unit 1, 125 volt D.C. power distribution, consisting of:

   a. TB Main Bus Nos. 1A, 1A-1 and 2A,

   b. TB Reserve Bus Nos. 1B and 1B-1, and

   c. RB Distribution Panel No. 1.

4.9 - SURVEILLANCE REQUIREMENTS

E. Distribution - Operating

Each of the required power distribution system divisions shall be determined energized at least once per 7 days by verifying correct breaker alignment and voltage on the busses/MCCs/panels.

3.9 - LIMITING CONDITIONS FOR OPERATION

4. For Unit 2, 125 volt D.C. power distribution, consisting of:

   a. TB Main Bus Nos. 1A, 2A and 2A-1,

   b. TB Reserve Bus Nos. 2B and 2B-1, and

   c. RB Distribution Panel No. 2.

APPLICABILITY:

OPERATIONAL MODE(s) 1, 2, and 3.

ACTIONS:

1. With one of the above required A.C. distribution systems not energized, re-energize the system within 8 hours or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

2. With one of the above required D.C. distribution systems not energized, re-energize the system within 2 hours or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

4.9 - SURVEILLANCE REQUIREMENTS

BASES

The initial conditions of design basis transient and accident analyses assume Engineering Safety Features (ESF) systems are OPERABLE. The A.C. and D.C. electrical power sources are designed to provide sufficient capacity, capability, redundancy, and reliability to ensure the availability of necessary power to ESF systems so that the fuel, reactor coolant system and containment design limits are not exceeded.

The A.C. and D.C. sources are designed to permit inspection and testing of all important areas and features, especially those that have a standby function. Periodic component tests are supplemented by extensive functional testing during refueling outages under simulated accident conditions.

3/4.9.A A.C. Sources - Operating

The OPERABILITY of the A.C. electrical power sources is consistent with the initial assumptions of the accident analyses and is based upon meeting the design basis of the plant. This includes maintaining at least one of the onsite or offsite A.C. sources, D.C. power sources and associated distribution systems OPERABLE during accident conditions concurrent with an assumed loss of all offsite power and a worst-case single failure.

There are two sources of electrical energy available, i.e., the offsite transmission system and the onsite diesel generators. Two unit reserve auxiliary transformers are available to supply the Station class 1E distribution system. The reserve auxiliary transformer is sized to carry 100% of the auxiliary load. If this reserve auxiliary transformer (the normal circuit) is lost, auxiliary power from the other unit can be obtained for one division through a 4160 volt bus tie (the alternate circuit). Additionally, two diesel generators are available to handle an accident. The allowable outage time takes into account the capacity and capability of the remaining A.C. sources, reasonable time for repairs, and the low probability of a design basis accident occurring during this period. Surveillance is required to ensure a highly reliable power source and no common cause failure mode for the remaining required offsite A.C. source.

Upon failure of one diesel generator, performance of appropriate surveillance requirements ensures a highly reliable power supply by checking the availability of the required offsite circuits, and the remaining required diesel generator. The initial surveillance is required to be completed regardless of how long the diesel inoperability persists, since the intent is that all diesel generator inoperabilities must be investigated for common cause failures. After the initial surveillance, an additional start test is required approximately mid-way through the allowed outage time to demonstrate continued OPERABILITY of the available onsite power sources. The diesel generator surveillance is limited to the normal start testing, since for cases in which less than a full complement of A.C. sources may be available, paralleling of two of the remaining A.C. sources may compromise the A.C. source independence. Additionally, the action provisions ensure that continued plant operation is not allowed when a complete loss of a required safety function (i.e., certain required components) would occur upon a loss of offsite power. These certain components which are critical to accomplishment of the required safety functions may be identified in advance and administratively controlled and/or evaluated on a case-by-case basis.

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Surveillance Requirements are also provided for demonstrating the OPERABILITY of the diesel generators. The specified testing is based on the guidance provided in Regulatory Guide 1.9, Revision 3 (7/93), Regulatory Guide 1.108, Revision 1, and Regulatory Guide 1.137, Revision 1, as modified by plant specific analysis, diesel generator manufacturer/vendor recommendations and responses to Generic Letter 84-15.

The diesel generators are equipped with a prelubrication system which maintains a continuous flow of oil to the diesel engine moving parts while the engine is shutdown. The purpose of this system is to increase long term diesel generator reliability by reducing the stress and wear caused by frequent dry starting of the diesel generator. The diesel generator prelube may be accomplished either through normal operation of the installed prelubrication system or by manual prelubrication of the diesel generator in accordance with the manufacturer's/vendor's instructions. Performance of an idle start of the diesel generator is not considered to be a means of prelubrication.

A periodic "start test" of the diesel generators demonstrates proper startup from standby conditions, and verifies that the required generator voltage and frequency is attained. For this test, the diesel generator may be slow started and reach rated speed on a prescribed schedule that is selected to minimize stress and wear. In cases where this Surveillance Requirement is being used to identify a possible common mode failure in accordance with the action provisions, this test eliminates the risk of paralleling two of the remaining A.C. sources, which may compromise the A.C. source independence.

A "load-run test" normally follows the periodic "start test" of the diesel generator to demonstrate operation at or near the continuous rating. This surveillance should only be conducted on one diesel generator at a time in order to avoid common mode failures that might result from offsite circuit or grid perturbations. A minimum run time of 60 minutes is required to stabilize engine temperatures. Actual run time should be in accordance with vendor recommendations with regard to good operating practice and should be sufficient to ensure that cooling and lubrication are adequate for extended periods of operation, while minimizing the time that the diesel generator is connected to the offsite source. This Surveillance Requirement may include gradual loading, as recommended by the manufacturer, so that mechanical stress and wear on the diesel engine are minimized. A load band is provided to avoid routine overloading of the diesel generators. Momentary transients outside the load band because of changing bus loads do not impact the validity of this test.

A periodic surveillance requirement is provided to assure the diesel generator is aligned to provide standby power on demand. Periodic surveillance requirements also verify that, without the aid of the refill compressor, sufficient air start capacity for each diesel generator is available. With either pair of air receiver tanks at the minimum specified pressure, there is sufficient air in the tanks to start the associated diesel generator.

The periodicity of surveillance requirements for the shared diesel generators shall be equivalent to those required for the unit diesel generators. For example, it is not the intention to perform surveillances for the shared diesel generators twice during the specified surveillance interval in order to satisfy each unit's diesel generator surveillance requirements. By appropriately staggering

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the surveillance intervals between all three (3) diesel generators further ensures that for any loaded diesel generator surveillances, not more than one diesel generator is rendered inoperable at any given time in order to perform such testing.

Surveillance requirements provide verification that there is an adequate inventory of fuel oil in the storage tanks that is sufficient to provide time to place the facility in a safe shutdown condition and to bring in replenishment fuel from an offsite location. Additional diesel fuel can normally be obtained and delivered to the site within an eight hour period; thus a two day supply provides for adequate margin. The operation of each required fuel oil transfer pump is demonstrated by transferring fuel oil from its associated storage tank to its associated day tank. This surveillance provides assurance that the fuel oil transfer pump is OPERABLE, the fuel oil piping system is intact, the fuel delivery piping is not obstructed, and the necessary fuel oil day tank instrumentation is OPERABLE.

A comprehensive surveillance program is provided to ensure the availability of high quality fuel oil for the diesel generators which is necessary to ensure proper operation. Water content should be minimized, because water in the fuel would contribute to excessive corrosion of the system, causing decreased reliability. The growth of micro-organisms results in slime formations, which are one of the chief causes of jellying in hydrocarbon fuels. Therefore, minimizing such slimes is also essential to assuring high reliability.

Sampling of both new diesel fuel oil and the bulk fuel oil storage tanks is in accordance with the American Society for Testing Materials (ASTM) standard D4057. Testing for API gravity is in accordance with ASTM D1298, water and sediment is in accordance with ASTM D1796, and the visual test for free water and particulate contamination (clear and bright) is in accordance with ASTM D4176. Testing for kinematic viscosity is in accordance with ASTM D445 and particulate contaminant testing is in accordance with ASTM D2276. Parameter limits are in accordance with ASTM D396 for API gravity, ASTM D975 for water and sediment and for kinematic viscosity, and ASTM D4176 for "clear and bright." The specific revision in use for each of these standards is controlled by procedure.

The diesel fuel oil day tanks are not equipped with the capability to obtain samples. Any accumulated water is removed by partially draining the day tank to the bulk fuel oil storage tank on a routine basis. Monthly sampling of the bulk fuel oil storage tank is then used to detect the presence of any water.

Fuel oil testing may indicate that such fuel oil is not within the required parameters. However, continued operation is acceptable while measures are taken to restore the properties of the fuel oil to within its limits since the properties of interest, even if they were not within the required limits, would not have an immediate effect on diesel generator operation. If the fuel oil properties cannot be returned to within their limits in the allowed time, the associated diesel generator(s) is (are) declared inoperable and the appropriate ACTION(s) taken.

A semi-annual surveillance is provided to verify the diesel generator can "fast start" from standby conditions and achieve the required voltage and frequency within the timing assumptions of the

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design basis loss of coolant accident safety analysis. Conducting this test on a semi-annual frequency is consistent with the intent of the reduction of cold testing identified in Generic Letter 84-15.

Additional surveillance requirements provide for periodic inspections and demonstration of the diesel generator capabilities, some are conducted in conjunction with a simulated loss of offsite power and/or a simulated ESF actuation signal. These tests of the diesel generator are expected to be conducted during an outage to functionally test the system. This testing is consistent with the intent of the diesel generator reliability programs recommended by Regulatory Guide 1.155.

3/4.9.B A.C. Sources - Shutdown

The A.C. sources required during Cold Shutdown, Refueling, when handling irradiated fuel and during operations with a potential for draining the reactor vessel provide assurance that:

1. Systems to provide adequate coolant inventory makeup are available for the irradiated fuel in the core in case of an inadvertent draindown of the reactor vessel;

2. Systems needed to mitigate a fuel handling accident are available;

3. Systems necessary to mitigate the effects of events that can lead to core damage during shutdown are OPERABLE; and

4. Instrumentation and control capability is available for monitoring and maintaining the unit in a cold shutdown condition and refueling condition.

With one or more of the required A.C. electrical power sources inoperable, the action provisions require a suspension of activities that will preclude the occurrence of actions that could potentially initiate the postulated events. However, timely suspension of these activities is not intended to preclude completion of actions necessary to establish a safe, conservative condition.

The Surveillance Requirements for A.C. Source Shutdown are the same as those for operation, with the exception of the periodic "load-run test" which is not required due to the limited redundancy of A.C. power sources.

3/4.9.C D.C. Sources - Operating

The station D.C. electrical power system provides the A.C. emergency power system with control power. It also provides both motive and control power to selected safety-related equipment. During normal operation, the D.C. electrical loads are powered from the battery chargers with batteries floating on the system. In case of loss of normal power to the battery charger, the D.C. load is automatically powered from the station batteries.

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Each battery of the D.C. electrical power systems is sized to start and carry the normal D.C. loads plus all D.C. loads required for safe shutdown on one unit and operations required to limit the consequences of a design basis event on the other unit for a period of 4 hours following loss of all A.C. sources. The battery chargers are sized to restore the battery to full charge under normal (non-emergency) load conditions. A normally disconnected alternate 125 volt battery is also provided as a backup for each normal battery. If both units are operating , the normal 125 volt battery must be returned to service within the specified time frame since the design configuration of the alternate battery circuit is susceptible to single failure and, hence, is not as reliable as the normal station circuit. During times when the other unit is in a Cold Shutdown or Refuel condition, an alternate 125 volt battery is available to replace a normal station 125 volt battery on a continuous basis to provide a second available power source. With the alternate 125 volt battery in service, the normally open breaker on the DC Reserve Bus is placed in the open position and posted, i.e., "tagged out."

With one of the required D.C. electrical power subsystems inoperable the remaining system has the capacity to support a safe shutdown and to mitigate an accident condition. However, a subsequent worst-case single failure would result in complete loss of ESF functions. Therefore, an allowed outage time is provided based on a reasonable time to assess plant status as a function of the inoperable D.C. electrical power subsystem and, if the D.C. electrical power subsystem is not restored to OPERABLE status, prepare to effect an orderly and safe plant shutdown.

Inoperable chargers do not necessarily indicate that the D.C. systems are not capable of performing their post-accident functions as long as the batteries are within their specified parameter limits. With both the required charger inoperable and the battery degraded, prompt action is required to assure an adequate D.C. power supply.

ACTION(s) are provided to delineate the measurements and time frames needed to continue to assure OPERABILITY of the Station batteries when battery parameters are outside their identified limits.

Battery surveillance requirements are based on the defined battery cell parameter values. Category A defines the normal parameter limit for each designated pilot cell in each battery. The pilot cells are the average cells in the battery based on previous test results. These cells are monitored closely as an indication of battery performance. Category B defines the normal parameter limits for each connected cell. The term "connected cell" excludes any battery cell that may be jumpered out because of a degraded condition or for any other reason. Category B also defines allowable values for each connected cell. These values, although reduced, provide assurance that sufficient capacity exists to perform the intended function and maintain a margin of safety. When any battery parameter is outside the Category B allowable value, the assurance of sufficient capacity as described above no longer exists and the battery must be declared inoperable.

Verifying battery terminal voltage while on float charge for the batteries helps to ensure the effectiveness of the charging system and the ability of the batteries to perform their intended function. The voltage requirements are based on the nominal design voltage of the battery and are consistent with the initial voltages assumed in the battery sizing calculations.

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Visual inspection to detect corrosion of the battery cells and connections, or measurement of the resistance of each connection provides an indication of physical damage or abnormal deterioration that could potentially degrade battery performance. The limits established for this Surveillance Requirement shall be no more than 20% above the resistance as measured during installation or not above the ceiling value established by the manufacturer.

Verifying an acceptable average temperature of battery cells is consistent with the recommendations of IEEE-450 and ensures that lower than normal temperatures do not act to inhibit or reduce battery capacity.

Verifying that the chargers will provide the manufacturer's rated current and voltage for four hours ensures that charger deterioration has not occurred and that the charger will provide the necessary capacity to restore the battery to a fully charged state.

A battery service test is a special test of the battery's capability "as found" to satisfy the design requirements of the D.C. electrical power system. The discharge rate and test length should correspond to the design duty cycle requirements.

A battery modified performance test is a test of the battery capacity and the battery's ability to meet the loads that exceed the constant current discharge rate of the battery (high rate short duration loads) of the battery's duty cycle. This test satisfies the requirements of both a service test and a performance test and is intended to detect any change in capacity and to determine overall batery degradation due to age and usage. The 125 volt batteries have a rated capacity of 125% of the load expected at the end of their service life allowing for a minimum battery capacity of at least 80% of the manufacturer's rating. A battery capacity of 80% indicates that the battery rate of deterioration is increasing, even if there is ample capacity to meet the load requirements. The 250 volt batteries do not have a rated capacity of 125% of the load expected at the end of their service therefore, the minimum allowable battery capacity is based on the capacity margin calculated from the design load profile for the battery.

3/4.9.D D.C. Sources - Shutdown

The D.C. sources required to be OPERABLE during Cold Shutdown, Refueling, when handling irradiated fuel and during operations with a potential for draining the reactor vessel provide assurance that:

1. Systems to provide adequate coolant inventory makeup are available for the irradiated fuel in the core in case of an inadvertent drain down of the reactor vessel;

2. Systems needed to mitigate a fuel-handling accident are available;

3. Systems necessary to mitigate the effects of events that can lead to core damage during shutdown are OPERABLE;

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4. Instrumentation and control capability is available for monitoring and maintaining the unit in a cold shutdown condition and refueling condition.

With one or more of the required D.C. electrical power sources inoperable, the action provisions require a suspension of activities that will preclude the occurrence of actions that could potentially initiate the postulated events. However, timely suspension of these activities is not intended to preclude completion of actions necessary to establish a safe, conservative condition.

3/4.9.E Distribution - Operating

The OPERABILITY of the A.C. and D.C. onsite power distribution systems ensures that sufficient power will be available to the safety related equipment required for (1) the safe shutdown of the facility and (2) the mitigation and control of accident conditions within the facility.

The surveillance requirements verify that the A.C. and D.C. electrical power distribution systems are functioning properly, with all the required circuit breakers closed and the buses energized from normal power. The verification of proper voltage availability on the buses ensures that the required power is readily available for motive as well as control functions for critical system loads connected to these buses. The frequency takes into account the redundant capability of the A.C. and D.C. electrical power distribution subsystems, and other indications available in the control room that will alert the operator to subsystem malfunctions.

3/4.9.F Distribution - Shutdown

The OPERABILITY of the minimum specified A.C. and D.C. onsite power distribution systems, during Cold Shutdown and Refueling and when handling irradiated fuel in the secondary containment, ensures that the facility can be maintained in these conditions for extended time periods and sufficient instrumentation and control capability is available for monitoring and maintaining the unit status. Requiring OPERABILITY of the minimum specified onsite power distribution systems when handling irradiated fuel in the secondary containment helps to ensure that systems needed to mitigate a fuel handling accident are available.

3/4.9.G RPS Power Monitoring

Specifications are provided to ensure the OPERABILITY of the reactor protection system (RPS) bus electrical protection assemblies (EPAs). Each RPS motor generator (MG) set and the alternate power source has 2 EPA CHANNEL(s) wired in series. A trip of either CHANNEL from either overvoltage, undervoltage, or underfrequency will disconnect the associated MG set or alternate power source.

BASES

The associated surveillance requirements provide for demonstration of the OPERABILITY of the RPS EPA's. The setpoints for overvoltage, undervoltage, and underfrequency have been chosen based on analysis (ref. February 4, 1983 letter to H. Denton from T. Rausch).

3.10 - LIMITING CONDITIONS FOR OPERATION

H. Water Level - Spent Fuel Storage Pool

The pool water level shall be maintained at a level of 33 feet.

APPLICABILITY:

Whenever irradiated fuel assemblies are in the spent fuel storage pool.

ACTION:

With the requirements of the above specification not satisfied, suspend all movement of fuel assemblies and crane operations with loads in the spent fuel storage pool area after placing the fuel assemblies and crane load in a safe condition. The provisions of Specification 3.0.C are not applicable.

4.10 - SURVEILLANCE REQUIREMENTS

H. Water Level - Spent Fuel Storage Pool

The water level in the spent fuel storage pool shall be determined to be at least at its minimum required depth at least once per 7 days.

3.11 - LIMITING CONDITIONS FOR OPERATION

C. MINIMUM CRITICAL POWER RATIO

The MINIMUM CRITICAL POWER RATIO (MCPR) shall be equal to or greater than the MCPR operating limit specified in the CORE OPERATING LIMITS REPORT.

APPLICABILITY:

OPERATIONAL MODE 1, when THERMAL POWER is greater than or equal to 25% of RATED THERMAL POWER.

ACTION:

With MCPR less than the applicable MCPR operating limit as determined for one of the conditions specified in the CORE OPERATING LIMITS REPORT:

1. Initiate corrective ACTION within 15 minutes, and
2. Restore MCPR to within the required limit within 2 hours.

With the provisions of the ACTION above not met, reduce THERMAL POWER to less than 25% of RATED THERMAL POWER within the next 4 hours.

4.11 - SURVEILLANCE REQUIREMENTS

C. MINIMUM CRITICAL POWER RATIO

MCPR shall be determined to be equal to or greater than the applicable MCPR operating limit specified in the CORE OPERATING LIMITS REPORT.

1. At least once per 24 hours,
2. Within 12 hours after completion of a THERMAL POWER increase of at least 15% of RATED THERMAL POWER, and
3. Initially and at least once per 12 hours when the reactor is operating with a LIMITING CONTROL ROD PATTERN for MCPR.
4. The provisions of Specification 4.0.D are not applicable.

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3/4.11.A AVERAGE PLANAR LINEAR HEAT GENERATION RATE

This specification assures that the peak cladding temperature following the postulated design basis loss-of-coolant accident will not exceed the limit specified in 10 CFR 50.46. The specification also assures that fuel rod mechanical integrity is maintained during normal and transient operations.

The peak cladding temperature (PCT) following a postulated loss-of-coolant accident is primarily a function of the average heat generation rate of all the rods of a fuel assembly at any axial location and is dependent only secondarily on the rod-to-rod power distribution within an assembly. The peak clad temperature is calculated assuming a LINEAR HEAT GENERATION RATE (LHGR) for the highest powered rod which is equal to or less than the design LHGR corrected for densification. The APLHGR limits specified are equivalent to the LHGR of the highest powered fuel rod assumed in the LOCA analysis divided by its local peaking factor. A conservative multiplier is applied to the LHGR assumed in the LOCA analysis to account for the uncertainty associated with the measurement of the APLHGR.

The calculational procedure used to establish the maximum APLHGR values uses NRC approved calculational models which are consistent with the requirements of Appendix K of 10 CFR Part 50. The approved calculational models are listed in Specification 6.9.

The daily requirement for calculating APLHGR when THERMAL POWER is greater than or equal to 25% of RATED THERMAL POWER is sufficient since power distribution shifts are very slow when there have not been significant power or control rod changes. The requirement to calculate APLHGR within 12 hours after the completion of a THERMAL POWER increase of at least 15% of RATED THERMAL POWER ensures thermal limits are met after power distribution shifts while still allotting time for the power distribution to stabilize. The requirement for calculating APLHGR after initially determining a LIMITING CONTROL ROD PATTERN exists ensures that APLHGR will be known following a change in THERMAL POWER or power shape, that could place operation above a thermal limit.

3/4.11.B APRM SETPOINTS

The fuel cladding integrity Safety Limits of Specification 2.1 were based on a power distribution which would yield the design LHGR at RATED THERMAL POWER. The flow biased neutron flux - high scram setting and control rod block functions of the APRM instruments for both two recirculation loop operation and single recirculation loop operation must be adjusted to ensure that the MCPR does not become less than the fuel cladding safety limit or that ≥1% plastic strain does not occur in the degraded situation. The scram settings and rod block settings are adjusted in accordance with the formula in this specification when the value of MFLPD indicates a higher peaked power distribution to ensure that an LHGR transient would not be increased in the degraded condition.

BASES

3/4.11.C MINIMUM CRITICAL POWER RATIO

The required operating limit MCPR at steady state operating conditions as specified in Specification 3.11.C are derived from the established fuel cladding integrity Safety Limit MCPR, and an analysis of abnormal operational transients. For any abnormal operating transient analysis evaluation with the initial condition of the reactor being at the steady state operating limit, it is required that the resulting MCPR does not decrease below the Safety Limit MCPR at any time during the transient assuming instrument trip setting given in Specification 2.2.

To assure that the fuel cladding integrity Safety Limit is not exceeded during any anticipated abnormal operational transient, the most limiting transients have been analyzed to determine which result in the largest reduction in the CRITICAL POWER RATIO (CPR). The type of transients evaluated were change of flow, increase in pressure and power, positive reactivity insertion, and coolant temperature decrease. The limiting transient yields the largest delta MCPR. When added to the Safety Limit MCPR, the required minimum operating limit MCPR of Specification 3.11.C is obtained and presented in the CORE OPERATING LIMITS REPORT.

The steady state values for MCPR specified were determined using NRC-approved methodology listed in Specification 6.9.

The purpose of the MCPR multiplicative factor specified in the CORE OPERATING LIMITS REPORT is to define MCPR operating limits at other than rated core flow conditions. At less than 100% of rated flow, the required MCPR is the product of the MCPR and the off rated flow MCPR multiplier factor. The MCPR multiplier assures that the Safety Limit MCPR will not be violated.

At THERMAL POWER levels less than or equal to 25% of RATED THERMAL POWER, the reactor will be operating at minimum recirculation pump speed and the moderator void content will be very small. For all designated control rod patterns which may be employed at this point, operating plant experience indicates that the resulting MCPR value has considerable margin. Thus, the demonstration of MCPR below this power level is unnecessary. The daily requirement for calculating MCPR when THERMAL POWER is greater than or equal to 25% of RATED THERMAL POWER is sufficient since power distribution shifts are very slow when there have not been significant power or control rod changes. The requirement for calculating MCPR after initially determining that a LIMITING CONTROL ROD PATTERN exists ensures that MCPR will be known following a change in THERMAL POWER or power shape, regardless of magnitude, that could place operation above a thermal limit.

ATTACHMENT D

SIGNIFICANT HAZARDS EVALUATIONS FOR DRESDEN AND QUAD CITIES NUCLEAR POWER STATIONS FOR PROPOSED TSUP CLEAN-UP CHANGES LICENSE NOS. DPR-19, DPR-25, DPR-29, AND DPR-30

ComEd has evaluated this proposed supplemental amendment and determined that it involves no significant hazards consideration. According to 10 CFR 50.92(c), a proposed amendment to an operating license involves no significant hazards consideration if operation of the facility, in accordance with the proposed amendment, would not:

1) Involve a significant increase in the probability or consequences of an accident previously evaluated; or

2) Create the possibility of a new or different kind of accident from any accident previously evaluated; or

3) Involve a significant reduction in a margin of safety.

ATTACHMENT D

The proposed changes do not involve a significant increase in the probability or consequences of an accident previously evaluated because:

In general, the proposed amendment represents the conversion of current requirements to a more generic format, or the addition of requirements which are based on the current safety analysis. Implementation of these changes will provide increased reliability of equipment assumed to operate in the current safety analysis, or provide continued assurance that specified parameters remain within their acceptance limits, and as such, will not significantly increase the probability or consequences of a previously evaluated accident.

Some of the proposed changes represent minor curtailments of the current requirements which are based on generic guidance or previously approved provisions for other stations. The proposed amendment for Dresden and Quad Cities Station's Technical Specifications are based on STS guidelines or later operating BWR plants' NRC accepted changes. Any deviations from STS requirements do not significantly increase the probability or consequences of any previously evaluated accidents for Dresden or Quad Cities Stations. The proposed amendment is consistent with the current safety analyses and has been previously determined to represent sufficient requirements for the assurance and reliability of equipment assumed to operate in the safety analysis, or provide continued assurance that specified parameters remain within their acceptance limits. As such, these changes will not significantly increase the probability or consequences of a previously evaluated accident.

The associated systems related to this proposed amendment are not assumed in any safety analysis to initiate any accident sequence for Dresden or Quad Cities Stations; therefore, the probability of any accident previously evaluated is not increased by the proposed amendment. In addition, the proposed surveillance requirements for the proposed amendments to these systems are generally more prescriptive than the current requirements specified within the Technical Specifications. The additional surveillance requirements improve the reliability and availability of all affected systems and therefore, reduce the consequences of any accident previously evaluated as the probability of the systems related to the TSUP open items outlined within the proposed Technical Specifications performing their intended function is increased by the additional surveillances.

The proposed changes do not create the possibility of a new or different kind of accident from any previously evaluated because:

In general, the proposed amendment represents the conversion of current requirements to a more generic format, the addition of requirements which are based on the current safety analysis, and some minor curtailments of the current requirements which are based on generic guidance or previously approved provisions for other stations. These changes do not involve revisions to the design of the station. Some of the changes may involve revision in the operation of the station; however, these provide additional restrictions which are in accordance with the current safety analysis, or are to provide for additional testing or surveillances which will not introduce new failure mechanisms beyond those already considered in the current safety analyses.

The proposed amendment for Dresden and Quad Cities Station's Technical Specification is based on STS guidelines or later operating BWR plants' NRC accepted changes. The proposed amendment has been reviewed for acceptability at the Dresden and Quad Cities Nuclear Power Stations considering similarity of system or component design versus the STS or later operating BWRs. Any deviations from STS requirements do not create the possibility of a new or different kind of accident previously evaluated for Dresden or Quad Cities Stations. No new modes of operation are introduced by the proposed changes. Surveillance requirements are changed to reflect improvements in technique, frequency of performance or operating experience at later plants. Proposed changes to action statements in many places add requirements that are not in the present technical specifications. The proposed changes maintain at least the present level of operability. Therefore, the proposed changes do not create the possibility of a new or different kind of accident from any previously evaluated.

The associated systems related to this proposed amendment are not assumed in any safety analysis to initiate any accident sequence for Dresden or Quad Cities Stations. In addition, the proposed surveillance requirements for affected systems associated with the TSUP open items are generally more prescriptive than the current requirements specified within the Technical Specifications; therefore, the proposed changes do not create the possibility of a new or different kind of accident from any previously evaluated.

ATTACHMENT D

The proposed changes do not involve a significant reduction in the margin of safety because:

In general, the proposed amendment represents the conversion of current requirements to a more generic format, the addition of requirements which are based on the current safety analysis, and some minor curtailments of the current requirements which are based on generic guidance or previously approved provisions for other stations. Some of the latter individual items may introduce minor reductions in the margin of safety when compared to the current requirements. However, other individual changes are the adoption of new requirements which will provide significant enhancement of the reliability of the equipment assumed to operate in the safety analysis, or provide enhanced assurance that specified parameters remain with their acceptance limits. These enhancements compensate for the individual minor reductions, such that taken together, the proposed changes will not significantly reduce the margin of safety.

The proposed amendment to the Technical Specifications implements present requirements, or the intent of present requirements in accordance with the guidelines set forth in the STS. Any deviations from STS requirements do not significantly reduce the margin of safety for Dresden or Quad Cities Stations. The proposed changes are intended to improve readability, usability, and the understanding of technical specification requirements while maintaining acceptable levels of safe operation. The proposed changes have been evaluated and found to be acceptable for use at Dresden or Quad Cities based on system design, safety analysis requirements and operational performance. Since the proposed changes are based on NRC accepted provisions at other operating plants that are applicable at Dresden or Quad Cities and maintain necessary levels of system or component reliability, the proposed changes do not involve a significant reduction in the margin of safety.

The proposed amendment for Dresden and Quad Cities Stations will not reduce the availability of systems associated with the TSUP open items when required to mitigate accident conditions; therefore, the proposed changes do not involve a significant reduction in the margin of safety.

ATTACHMENT D

ENVIRONMENTAL ASSESSMENT STATEMENT APPLICABILITY REVIEW

ComEd has evaluated the proposed supplemental amendment against the criteria for the identification of licensing and regulatory actions requiring environmental assessment in accordance with 10 CFR 51.20. It has been determined that the proposed supplemental changes meet the criteria for a categorical exclusion as provided under 10 CFR 51.22 (c)(9). This conclusion has been determined because the supplemental changes requested do not pose significant hazards consideration or do not involve a significant increase in the amounts, and no significant changes in the types, of any effluent that may be released offsite. Additionally, this request does not involve a significant increase in individual or cumulative occupational radiation exposure. Therefore, the Environmental Assessment Statement is not applicable for these supplemental changes.