ATTACHMENT 2
Markup of Proposed Operating License, Technical Specifications, and UFSAR Pages

Limerick Generating Station, Units 1 and 2

Facility Operating License Nos. NPF-39 and NPF-85

REVISED OPERATING LICENSE AND TECHNICAL SPECIFICATIONS PAGES

Operating License, Page 3 (Units 1 and 2)
TS Page 1-6 (Units 1 and 2)
TS Page 2-4 (Units 1 and 2)
TS Page 3/4 1-19 (Units 1 and 2)
TS Page 3/4 3-5 (Units 1 and 2)
TS Page 3/4 3-7 (Units 1 and 2)
TS Page 3/4 3-8 (Units 1 and 2)
TS Page 3/4 3-46 (Units 1 and 2)
TS Page 3/4 3-48 (Units 1 and 2)
TS Page 3/4 3-60 (Units 1 and 2)
TS Page 3/4 4-1 (Units 1 and 2)
TS Page 3/4 4-2 (Units 1 and 2)

REVISED UFSAR PAGES
7.4-12
7.6-43
9.3-23

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

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

(5) Pursuant to the Act and 10 CFR Parts 30, 40 and 70, to possess, but not separate, such byproduct and special nuclear materials as may be produced by the operation of the facility, and to receive and possess, but not separate, such source, byproduct, and special nuclear materials as contained in the fuel assemblies and fuel channels from the Shoreham Nuclear Power Station.

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 (except as exempted from compliance in Section 2.D. below) and is subject to all applicable provisions of the Act and to the rules, regulations, and orders of the Commission now or hereafter in effect; and is subject to the additional conditions specified or incorporated below:

(1) Maximum Power Level

Exelon Generation Company is authorized to operate the facility at reactor core power levels not in excess of ~~3458~~ 3515 megawatts thermal (100% rated power) in accordance with the conditions specified herein and in Attachment 1 to this license. The items identified in Attachment 1 to this license shall be completed as specified. Attachment 1 is hereby incorporated into this license.

(2) Technical Specifications

The Technical Specifications contained in Appendix A and the Environmental Protection Plan contained in Appendix B, as revised through Amendment No. 198, are hereby incorporated into this license. Exelon Generation Company shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

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

(5) Pursuant to the Act and 10 CFR Parts 30, 40 and 70, to possess, but not separate, such byproduct and special nuclear materials as may be produced by the operation of the facility, and to receive and possess, but not separate, such source, byproduct, and special nuclear materials as contained in the fuel assemblies and fuel channels from the Shoreham Nuclear Power Station.

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 (except as exempted from compliance in Section 2.D. below) and is subject to all applicable provisions of the Act and to the rules, regulations, and orders of the Commission now or hereafter in effect; and is subject to the additional conditions specified or incorporated below:

(1) Maximum Power Level

3515

Exelon Generation Company is authorized to operate the facility at reactor core power levels of ~~3458~~ megawatts thermal (100 percent rated power) in accordance with the conditions specified herein.

(2) Technical Specifications

The Technical Specifications contained in Appendix A and the Environmental Protection Plan contained in Appendix B, as revised through Amendment No. 159, are hereby incorporated into this license. Exelon Generation Company shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

(3) Fire Protection (Section 9.5, SSER-2, -4)*

Exelon Generation Company shall implement and maintain in effect all provisions of the approved Fire Protection Program as described in the Updated Final Safety Analysis Report for the facility, and as approved in the NRC Safety Evaluation Report dated August 1983 through Supplement 9, dated August 1989, and Safety Evaluation dated November 20, 1995, subject to the following provision:

The licensee may make changes to the approved fire protection program without prior approval of the Commission only if those changes would not adversely affect the ability to achieve and maintain safe shutdown in the event of a fire.

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

PURGE - PURGING

1.31 PURGE or PURGING shall be the controlled process of discharging air or gas from a confinement to maintain temperature, pressure, humidity, concentration or other operating condition, in such a manner that replacement air or gas is required to purify the confinement.

RATED THERMAL POWER

1.32 RATED THERMAL POWER shall be a total reactor core heat transfer rate to the reactor coolant of ~~3458~~ 3515 MWt.

REACTOR ENCLOSURE SECONDARY CONTAINMENT INTEGRITY

1.33 REACTOR ENCLOSURE SECONDARY CONTAINMENT INTEGRITY shall exist when:

a. All reactor enclosure secondary containment penetrations required to be closed during accident conditions are either:

1. Capable of being closed by an OPERABLE secondary containment automatic isolation system, or

2. Closed by at least one manual valve, blind flange, slide gate damper, or deactivated automatic valve secured in its closed position, except as provided by Specification 3.6.5.2.1.

b. All reactor enclosure secondary containment hatches and blowout panels are closed and sealed.

c. The standby gas treatment system is in compliance with the requirements of Specification 3.6.5.3.

d. The reactor enclosure recirculation system is in compliance with the requirements of Specification 3.6.5.4.

e. At least one door in each access to the reactor enclosure secondary containment is closed.

f. The sealing mechanism associated with each reactor enclosure secondary containment penetration, e.g., welds, bellows, or O-rings, is OPERABLE.

g. The pressure within the reactor enclosure secondary containment is less than or equal to the value required by Specification 4.6.5.1.1a.

REACTOR PROTECTION SYSTEM RESPONSE TIME

1.34 REACTOR PROTECTION SYSTEM RESPONSE TIME shall be the time interval from when the monitored parameter exceeds its trip setpoint at the channel sensor until de-energization of the scram pilot valve solenoids. The response time may be measured by any series of sequential, overlapping or total steps such that the entire response time is measured.

RECENTLY IRRADIATED FUEL

1.35 RECENTLY IRRADIATED FUEL is fuel that has occupied part of a critical reactor core within the previous 24 hours.

REFUELING FLOOR SECONDARY CONTAINMENT INTEGRITY

1.36 REFUELING FLOOR SECONDARY CONTAINMENT INTEGRITY shall exist when:

a. All refueling floor secondary containment penetrations required to be closed during accident conditions are either:

PURGE - PURGING

1.31 PURGE or PURGING shall be the controlled process of discharging air or gas from a confinement to maintain temperature, pressure, humidity, concentration or other operating condition, in such a manner that replacement air or gas is required to purify the confinement.

RATED THERMAL POWER

1.32 RATED THERMAL POWER shall be a total reactor core heat transfer rate to the reactor coolant of ~~3458~~ 3515 MWt.

REACTOR ENCLOSURE SECONDARY CONTAINMENT INTEGRITY

1.33 REACTOR ENCLOSURE SECONDARY CONTAINMENT INTEGRITY shall exist when:

a. All reactor enclosure secondary containment penetrations required to be closed during accident conditions are either:

1. Capable of being closed by an OPERABLE secondary containment automatic isolation system, or

2. Closed by at least one manual valve, blind flange, slide gate damper or deactivated automatic valve secured in its closed position, except as provided by Specification 3.6.5.2.1.

b. All reactor enclosure secondary containment hatches and blowout panels are closed and sealed.

c. The standby gas treatment system is in compliance with the requirements of Specification 3.6.5.3.

d. The reactor enclosure recirculation system is in compliance with the requirements of Specification 3.6.5.4.

e. At least one door in each access to the reactor enclosure secondary containment is closed.

f. The sealing mechanism associated with each reactor enclosure secondary containment penetration, e.g., welds, bellows, or O-rings, is OPERABLE.

g. The pressure within the reactor enclosure secondary containment is less than or equal to the value required by Specification 4.6.5.1.1a.

REACTOR PROTECTION SYSTEM RESPONSE TIME

1.34 REACTOR PROTECTION SYSTEM RESPONSE TIME shall be the time interval from when the monitored parameter exceeds its trip setpoint at the channel sensor until de-energization of the scram pilot valve solenoids. The response time may be measured by any series of sequential, overlapping or total steps such that the entire response time is measured.

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1.36 REFUELING FLOOR SECONDARY CONTAINMENT INTEGRITY shall exist when:

a. All refueling floor secondary containment penetrations required to be closed during accident conditions are either:

TABLE 2.2.1-1

REACTOR PROTECTION SYSTEM INSTRUMENTATION SETPOINTS

FUNCTIONAL UNIT	TRIP SETPOINT	ALLOWABLE VALUES
1. Intermediate Range Monitor, Neutron Flux-High	≤ 120/125 divisions of full scale	≤ 122/125 divisions of full scale
2. Average Power Range Monitor:		
a. Neutron Flux-Upscale (Setdown)	≤ 15.0% of RATED THERMAL POWER	≤ 20.0% of RATED THERMAL POWER
b. Simulated Thermal Power - Upscale:		
- Two Recirculation Loop Operation	≤ ~~0.66 W + 62.8%~~ 0.65 W + 61.7% and ≤ 116.6% of RATED THERMAL POWER	≤ ~~0.66 W + 63.3%~~ 0.65 W + 62.2% and 117.0% of RATED THERMAL POWER
- Single Recirculation Loop Operation***	≤ ~~0.66 (W-7.6%) + 62.8%~~ 0.65 (W-7.6%) + 61.5% and ≤ 116.6% of RATED THERMAL POWER	≤ ~~0.66 (W-7.6%) + 63.3%~~ 0.65 (W-7.6%) + 62.0% and ≤ 117.0% of RATED THERMAL POWER
c. Neutron Flux - Upscale	118.3% of RATED THERMAL POWER	118.7% of RATED THERMAL POWER
d. Inoperative	N.A.	N.A.
e. 2-Out-Of-4 Voter	N.A.	N.A.
f. OPRM Upscale	****	N.A.
3. Reactor Vessel Steam Dome Pressure - High	≤ 1096 psig	≤ 1103 psig
4. Reactor Vessel Water Level - Low, Level 3	≥ 12.5 inches above instrument zero*	≥ 11.0 inches above instrument zero
5. Main Steam Line Isolation Valve - Closure	≤ 8% closed	≤ 12% closed
6. DELETED	DELETED	DELETED
7. Drywell Pressure - High	≤ 1.68 psig	≤ 1.88 psig
8. Scram Discharge Volume Water Level - High		
a. Level Transmitter	≤ 260' 9 5/8" elevation**	≤ 261' 5 5/8" elevation
b. Float Switch	≤ 260' 9 5/8" elevation**	≤ 261' 5 5/8" elevation
9. Turbine Stop Valve - Closure	≤ 5% closed	≤7% closed
10. Turbine Control Valve Fast Closure, Trip Oil Pressure - Low	≥ 500 psig	≥ 465 psig
11. Reactor Mode Switch Shutdown Position	N.A.	N.A.
12. Manual Scram	N.A.	N.A.

* See Bases Figure B 3/4.3-1.
** Equivalent to 25.45 gallons/scram discharge volume.
*** The 7.6% flow "offset" for Single Loop Operation (SLO) is applied for W ≥ 7.6%. For flows W < 7.6%, the (W-7.6%) term is set equal to zero.
**** See COLR for OPRM period based detection algorithm trip setpoints. OPRM Upscale trip output auto-enable (not bypassed) setpoints shall be APRM Simulated Thermal Power ≥ ~~30%~~ 29.5% and recirculation drive flow < 60%.

TABLE 2.2.1-1

REACTOR PROTECTION SYSTEM INSTRUMENTATION SETPOINTS

FUNCTIONAL UNIT	TRIP SETPOINT	ALLOWABLE VALUES
1. Intermediate Range Monitor, Neutron Flux-High	≤ 120/125 divisions of full scale	≤ 122/125 divisions of full scale
2. Average Power Range Monitor:		
a. Neutron Flux-Upscale (Setdown)	≤ 15.0% of RATED THERMAL POWER	≤ 20.0% of RATED THERMAL POWER
b. Simulated Thermal Power - Upscale:		
- Two Recirculation Loop Operation	≤ ~~0.66 W + 62.8%~~ 0.65 W + 61.7% and ≤ 116.6% of RATED THERMAL POWER	≤ ~~0.66 W + 63.3%~~ 0.65 W + 62.2% and ≤ 117.0% of RATED THERMAL POWER
- Single Recirculation Loop Operation***	≤ ~~0.66 (W-7.6%) + 62.8%~~ 0.65 (W-7.6%) + 61.5% and ≤ 116.6% of RATED THERMAL POWER	≤ ~~0.66 (W-7.6%) + 63.3%~~ 0.65 (W-7.6%) + 62.0% and ≤ 117.0% of RATED THERMAL POWER
c. Neutron Flux - Upscale	118.3% of RATED THERMAL POWER	118.7% of RATED THERMAL POWER
d. Inoperative	N.A.	N.A.
e. 2-Out-Of-4 Voter	N.A.	N.A.
f. OPRM Upscale	****	N.A.
3. Reactor Vessel Steam Dome Pressure - High	≤ 1096 psig	≤ 1103 psig
4. Reactor Vessel Water Level - Low, Level 3	≥ 12.5 inches above instrument zero*	≥ 11.0 inches above instrument zero
5. Main Steam Line Isolation Valve - Closure	≤ 8% closed	≤ 12% closed
6. DELETED	DELETED	DELETED
7. Drywell Pressure - High	≤ 1.68 psig	≤ 1.88 psig
8. Scram Discharge Volume Water Level - High		
a. Level Transmitter	≤ 261' 1 1/4" elevation**	≤ 261' 9 1/4" elevation
b. Float Switch	≤ 261' 1 1/4" elevation**	≤ 261' 9 1/4" elevation

* See Bases Figure B 3/4.3-1.

** Equivalent to 25.58 gallons/scram discharge volume.

*** The 7.6% flow "offset" for Single Loop Operation (SLO) is applied for W ≥ 7.6%. For flows W < 7.6%, the (W-7.6%) term is set equal to zero.

**** See COLR for OPRM period based detection algorithm trip setpoints. OPRM Upscale trip output auto-enable (not bypassed) setpoints shall be APRM Simulated Thermal Power ≥ ~~30%~~ 29.5% and recirculation drive flow < 60%.

3/4.1.5 STANDBY LIQUID CONTROL SYSTEM

LIMITING CONDITION FOR OPERATION

3.1.5 The standby liquid control system shall be OPERABLE and consist of ~~a minimum of~~ the following:

a. In OPERATIONAL CONDITIONS 1 and 2, two pumps and corresponding flow paths,

b. In OPERATIONAL CONDITION 3, one pump and corresponding flow path.

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2, and 3

ACTION:

a. With only one pump and corresponding explosive valve OPERABLE, in OPERATIONAL CONDITION 1 or 2, restore one inoperable pump and corresponding explosive valve to OPERABLE status within 7 days or be in at least HOT SHUTDOWN within the next 12 hours.

b. With standby liquid control system otherwise inoperable, in OPERATIONAL CONDITION 1, 2, or 3, restore the system to OPERABLE status within 8 hours or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the next 24 hours.

SURVEILLANCE REQUIREMENTS

4.1.5 The standby liquid control system shall be demonstrated OPERABLE:

a. In accordance with the Surveillance Frequency Control Program by verifying that:

1. The temperature of the sodium pentaborate solution is within the limits of Figure 3.1.5-1.

2. The available volume of sodium pentaborate solution is at least 3160 gallons.

3. The temperature of the pump suction piping is within the limits of Figure 3.1.5-1 for the most recent concentration analysis.

3/4.1.5 STANDBY LIQUID CONTROL SYSTEM

LIMITING CONDITION FOR OPERATION

3.1.5 The standby liquid control system shall be OPERABLE and consist of ~~a minimum of~~ the following:

a. In OPERATIONAL CONDITIONS 1 and 2, two pumps and corresponding flow paths,

b. In OPERATIONAL CONDITION 3, one pump and corresponding flow path.

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2 and 3

ACTION:

a. With only one pump and corresponding explosive valve OPERABLE, in OPERATIONAL CONDITION 1 or 2, restore one inoperable pump and corresponding explosive valve to OPERABLE status within 7 days or be in at least HOT SHUTDOWN within the next 12 hours.

b. With standby liquid control system otherwise inoperable, in OPERATIONAL CONDITION 1, 2, or 3, restore the system to OPERABLE status within 8 hours or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the next 24 hours.

SURVEILLANCE REQUIREMENTS

4.1.5 The standby liquid control system shall be demonstrated OPERABLE:

a. In accordance with the Surveillance Frequency Control Program by verifying that:

1. The temperature of the sodium pentaborate solution is within the limits of Figure 3.1.5-1.

2. The available volume of sodium pentaborate solution is at least 3160 gallons.

3. The temperature of the pump suction piping is within the limits of Figure 3.1.5-1 for the most recent concentration analysis.

TABLE 3.3.1-1 (Continued)

REACTOR PROTECTION SYSTEM INSTRUMENTATION

TABLE NOTATIONS

(a) A channel may be placed in an inoperable status for up to 6 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.

(b) This function shall be automatically bypassed when the reactor mode switch is in the Run position.

(c) DELETED

(d) The noncoincident NMS reactor trip function logic is such that all channels go to both trip systems. Therefore, when the "shorting links" are removed, the Minimum OPERABLE Channels Per Trip System is 6 IRMs.

(e) An APRM channel is inoperable if there are less than 3 LPRM inputs per level or less than 20 LPRM inputs to an APRM channel, or if more than 9 LPRM inputs to the APRM channel have been bypassed since the last APRM calibration (weekly gain calibration).

(f) This function is not required to be OPERABLE when the reactor pressure vessel head is removed per Specification 3.10.1.

(g) This function shall be automatically bypassed when the reactor mode switch is not in the Run position.

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

(i) With any control rod withdrawn. Not applicable to control rods removed per Specification 3.9.10.1 or 3.9.10.2.

(j) This function shall be automatically bypassed when turbine first stage pressure is equivalent to a THERMAL POWER of less than ~~30%~~ 29.5% of RATED THERMAL POWER.

(k) Also actuates the EOC-RPT system.

(l) DELETED

(m) Each APRM channel provides inputs to both trip systems.

(n) DELETED

(o) With THERMAL POWER ≥ 25% RATED THERMAL POWER. The OPRM Upscale trip output shall be automatically enabled (not bypassed) when APRM Simulated Thermal Power is ≥ ~~30%~~ 29.5% and recirculation drive flow is < 60%. The OPRM trip output may be automatically bypassed when APRM Simulated Thermal Power is < ~~30%~~ 29.5% or recirculation drive flow is ≥ 60%.

(p) A minimum of 23 cells, each with a minimum of 2 OPERABLE LPRMs, must be OPERABLE for an OPRM channel to be OPERABLE.

TABLE 3.3.1-1 (Continued)

REACTOR PROTECTION SYSTEM INSTRUMENTATION

TABLE NOTATIONS

(a) A channel may be placed in an inoperable status for up to 6 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.

(b) This function shall automatically be bypassed when the reactor mode switch is in the Run position.

(c) DELETED

(d) The noncoincident NMS reactor trip function logic is such that all channels go to both trip systems. Therefore, when the "shorting links" are removed, the Minimum OPERABLE Channels Per Trip System is 6 IRMs.

(e) An APRM channel is inoperable if there are less than 3 LPRM inputs per level or less than 20 LPRM inputs to an APRM channel, or if more than 9 LPRM inputs to the APRM channel have been bypassed since the last APRM calibration (weekly gain calibration).

(f) This function is not required to be OPERABLE when the reactor pressure vessel head is removed per Specification 3.10.1.

(g) This function shall be automatically bypassed when the reactor mode switch is not in the Run position.

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

(i) With any control rod withdrawn. Not applicable to control rods removed per Specification 3.9.10.1 or 3.9.10.2.

(j) This function shall be automatically bypassed when turbine first stage pressure is equivalent to a THERMAL POWER of less than ~~30%~~ 29.5% of RATED THERMAL POWER.

(k) Also actuates the EOC-RPT system.

(l) DELETED

(m) Each APRM channel provides inputs to both trip systems.

(n) DELETED

(o) With THERMAL POWER ≥ 25% RATED THERMAL POWER. The OPRM Upscale trip output shall be automatically enabled (not bypassed) when APRM Simulated Thermal Power is ≥ ~~30%~~ 29.5% and recirculation drive flow is < 60%. The OPRM trip output may be automatically bypassed when APRM Simulated Thermal Power is < ~~30%~~ 29.5% or recirculation drive flow is ≥ 60%.

(p) A minimum of 23 cells, each with a minimum of 2 OPERABLE LPRMs, must be OPERABLE for an OPRM channel to be OPERABLE.

TABLE 4.3.1.1-1

REACTOR PROTECTION SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT	CHANNEL CHECK(n)	CHANNEL FUNCTIONAL TEST(n)	CHANNEL CALIBRATION(a)(n)	OPERATIONAL CONDITIONS FOR WHICH SURVEILLANCE REQUIRED
1. Intermediate Range Monitors:				
a. Neutron Flux - High	(b)			2
		(j)		3(i), 4(i), 5(i)
b. Inoperative	N.A.	(j)	N.A.	2, 3(i), 4(i), 5(i)
2. Average Power Range Monitor(f):				
a. Neutron Flux - Upscale (Setdown)	(b)	(l)	, (o), (p)	2
b. Simulated Thermal Power - Upscale		(e)	(d), (g)	1
c. Neutron Flux - Upscale			(d)	1
d. Inoperative	N.A.		N.A.	1, 2
e. 2-Out-Of-4 Voter			N.A.	1, 2
f. OPRM Upscale		(e)	(c)(g)	1(m)
3. Reactor Vessel Steam Dome Pressure - High				1, 2(h)
4. Reactor Vessel Water Level-Low, Level 3				1, 2
5. Main Steam Line Isolation Valve - Closure	N.A.			1
6. DELETED				
7. Drywell Pressure - High				1, 2
8. Scram Discharge Volume Water Level - High				
a. Level Transmitter				1, 2, 5(i)
b. Float Switch	N.A.			1, 2, 5(i)

TABLE 4.3.1.1-1

REACTOR PROTECTION SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT	CHANNEL CHECK (n)	CHANNEL FUNCTIONAL TEST (n)	CHANNEL CALIBRATION(a)(n)	OPERATIONAL CONDITIONS FOR WHICH SURVEILLANCE REQUIRED
1. Intermediate Range Monitors:				
a. Neutron Flux - High	(b)			2
		(j)		3(i), 4(i), 5(i)
b. Inoperative	N.A.	(j)	N.A.	2, 3(i), 4(i), 5(i)
2. Average Power Range Monitor(f):				
a. Neutron Flux - Upscale (Setdown)	(b)	(l)	, (o), (p)	2
b. Simulated Thermal Power - Upscale		(e)	(d), (g)	1
c. Neutron Flux - Upscale			(d)	1
d. Inoperative	N.A.		N.A.	1, 2
e. 2-Out-Of-4 Voter			N.A.	1, 2
f. OPRM Upscale		(e)	(c)(g)	1(m)
3. Reactor Vessel Steam Dome Pressure - High				1, 2(h)
4. Reactor Vessel Water Level-Low, Level 3				1, 2
5. Main Steam Line Isolation Valve - Closure	N.A.			1
6. DELETED				
7. Drywell Pressure - High				1, 2
8. Scram Discharge Volume Water Level - High				
a. Level Transmitter				1, 2, 5(i)
b. Float Switch	N.A.			1, 2, 5(i)

TABLE 4.3.1.1-1 (Continued)

REACTOR PROTECTION SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT	CHANNEL CHECK(n)	CHANNEL FUNCTIONAL TEST(n)	CHANNEL CALIBRATION(a)(n)	OPERATIONAL CONDITIONS FOR WHICH SURVEILLANCE REQUIRED
9. Turbine Stop Valve - Closure	N.A.			1
10. Turbine Control Valve Fast Closure, Trip Oil Pressure - Low	N.A.			1
11. Reactor Mode Switch Shutdown Position	N.A.		N.A.	1, 2, 3, 4, 5
12. Manual Scram	N.A.		N.A.	1, 2, 3, 4, 5

29.5%

(a) Neutron detectors may be excluded from CHANNEL CALIBRATION.
(b) The IRM and SRM channels shall be determined to overlap for at least 1/2 decades during each startup after entering OPERATIONAL CONDITION 2 and the IRM and APRM channels shall be determined to overlap for at least 1/2 decades during each controlled shutdown, if not performed within the previous 7 days.
(c) Calibration includes verification that the OPRM Upscale trip auto-enable (not-bypass) setpoint for APRM Simulated Thermal Power is ≥ ~~30%~~ and for recirculation drive flow is < 60%.
(d) The more frequent calibration shall consist of the adjustment of the APRM channel to conform to the power values calculated by a heat balance during OPERATIONAL CONDITION 1 when THERMAL POWER ≥25% of RATED THERMAL POWER. Adjust the APRM channel if the absolute difference is greater than 2% of RATED THERMAL POWER.
(e) CHANNEL FUNCTIONAL TEST shall include the flow input function, excluding the flow transmitter.
(f) The LPRMs shall be calibrated at least once per 2000 effective full power hours (EFPH).
(g) The less frequent calibration includes the flow input function.
(h) This function is not required to be OPERABLE when the reactor pressure vessel head is removed per Specification 3.10.1.
(i) With any control rod withdrawn. Not applicable to control rods removed per Specification 3.9.10.1 or 3.9.10.2.
(j) If the RPS shorting links are required to be removed per Specification 3.9.2, they may be reinstalled for up to 2 hours for required surveillance. During this time, CORE ALTERATIONS shall be suspended, and no control rod shall be moved from its existing position.
(k) DELETED
(l) Not required to be performed when entering OPERATIONAL CONDITION 2 from OPERATIONAL CONDITION 1 until 12 hours after entering OPERATIONAL CONDITION 2.
(m) With THERMAL POWER ≥ 25% of RATED THERMAL POWER.
(n) Frequencies are specified in the Surveillance Frequency Control Program unless otherwise noted in the table.

Insert 1

TABLE 4.3.1.1-1 (Continued)

REACTOR PROTECTION SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT	CHANNEL CHECK (n)	CHANNEL FUNCTIONAL TEST (n)	CHANNEL CALIBRATION(a)(n)	OPERATIONAL CONDITIONS FOR WHICH SURVEILLANCE REQUIRED
9. Turbine Stop Valve - Closure	N.A.			1
10. Turbine Control Valve Fast Closure, Trip Oil Pressure - Low	N.A.			1
11. Reactor Mode Switch Shutdown Position	N.A.		N.A.	1, 2, 3, 4, 5
12. Manual Scram	N.A.		N.A.	1, 2, 3, 4, 5

(a) Neutron detectors may be excluded from CHANNEL CALIBRATION. 29.5%
(b) The IRM and SRM channels shall be determined to overlap for at least 1/2 decades during each startup after entering OPERATIONAL CONDITION 2 and the IRM and APRM channels shall be determined to overlap for a least 1/2 decades during each controlled shutdown, if not performed within the previous 7 days.
(c) Calibration includes verification that the OPRM Upscale trip auto-enable (not-bypass) setpoint for APRM Simulated Thermal Power is ≥ ~~30%~~ and for recirculation drive flow is < 60%.
(d) The more frequent calibration shall consist of the adjustment of the APRM channel to conform to the power values calculated by a heat balance during OPERATIONAL CONDITION 1 when THERMAL POWER ≥ 25% of RATED THERMAL POWER. Adjust the APRM channel if the absolute difference is greater than 2% of RATED THERMAL POWER.
(e) CHANNEL FUNCTIONAL TEST shall include the flow input function, excluding the flow transmitter.
(f) The LPRMs shall be calibrated at least once per 2000 effective full power hours (EFPH).
(g) The less frequent calibration includes the flow input function.
(h) This function is not required to be OPERABLE when the reactor pressure vessel head is removed per Specification 3.10.1.
(i) With any control rod withdrawn. Not applicable to control rods removed per Specification 3.9.10.1 or 3.9.10.2.
(j) If the RPS shorting links are required to be removed per Specification 3.9.2, they may be reinstalled for up to 2 hours for required surveillance. During this time, CORE ALTERATIONS shall be suspended, and no control rod shall be moved from its existing position.
(k) DELETED
(l) Not required to be performed when entering OPERATIONAL CONDITION 2 from OPERATIONAL CONDITION 1 until 12 hours after entering OPERATIONAL CONDITION 2.
(m) With THERMAL POWER ≥ 25% of RATED THERMAL POWER.
(n) Frequencies are specified in the Surveillance Frequency Control Program unless otherwise noted in the table.

Insert 1

Insert 1

(o) If the as-found channel setpoint is outside its predefined as-found tolerance, then the channel shall be evaluated to verify that it is functioning as required before returning the channel to service.

(p) The instrument channel setpoint shall be reset to a value that is within the as-left tolerance around the Trip Setpoint at the completion of the surveillance; otherwise, the channel shall be declared inoperable. Setpoints more conservative than the Trip Setpoint are acceptable provided that the as-found and as-left tolerances apply to the actual setpoint implemented in the surveillance procedures (field setting) to confirm channel performance. The methodologies used to determine the as-found and the as-left tolerances are specified in the associated Technical Specifications Bases.

END-OF-CYCLE RECIRCULATION PUMP TRIP SYSTEM INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.4.2 The end-of-cycle recirculation pump trip (EOC-RPT) system instrumentation channels shown in Table 3.3.4.2-1 shall be OPERABLE with their trip setpoints set consistent with the values shown in the Trip Setpoint column of Table 3.3.4.2-2 and with the END-OF-CYCLE RECIRCULATION PUMP TRIP SYSTEM RESPONSE TIME as shown in Table 3.3.4.2-3.

APPLICABILITY: OPERATIONAL CONDITION 1, when THERMAL POWER is greater than or equal to ~~30%~~ 29.5% of RATED THERMAL POWER.

ACTION:

a. With an end-of-cycle recirculation pump trip system instrumentation channel trip setpoint less conservative than the value shown in the Allowable Values column of Table 3.3.4.2-2, declare the channel inoperable until the channel is restored to OPERABLE status with the channel setpoint adjusted consistent with the Trip Setpoint value.

b. With the number of OPERABLE channels one less than required by the Minimum OPERABLE Channels per Trip System requirement for one or both trip systems, place the inoperable channel(s) in the tripped condition within 12 hours.

c. With the number of OPERABLE channels two or more less than required by the Minimum OPERABLE Channels per Trip System requirement for one trip system and:

   1. If the inoperable channels consist of one turbine control valve channel and one turbine stop valve channel, place both inoperable channels in the tripped condition within 12 hours.

   2. If the inoperable channels include two turbine control valve channels or two turbine stop valve channels, declare the trip system inoperable.

d. With one trip system inoperable, restore the inoperable trip system to OPERABLE status within 72 hours or take the ACTION required by Specification 3.2.3.

e. With both trip systems inoperable, restore at least one trip system to OPERABLE status within one hour or take the ACTION required by Specification 3.2.3.

END-OF-CYCLE RECIRCULATION PUMP TRIP SYSTEM INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.4.2 The end-of-cycle recirculation pump trip (EOC-RPT) system instrumentation channels shown in Table 3.3.4.2-1 shall be OPERABLE with their trip setpoints set consistent with the values shown in the Trip Setpoint column of Table 3.3.4.2-2 and with the END-OF-CYCLE RECIRCULATION PUMP TRIP SYSTEM RESPONSE TIME as shown in Table 3.3.4.2-3.

APPLICABILITY: OPERATIONAL CONDITION 1, when THERMAL POWER is greater than or equal to ~~30%~~ 29.5% of RATED THERMAL POWER.

ACTION:

a. With an end-of-cycle recirculation pump trip system instrumentation channel trip setpoint less conservative than the value shown in the Allowable Values column of Table 3.3.4.2-2, declare the channel inoperable until the channel is restored to OPERABLE status with the channel setpoint adjusted consistent with the Trip Setpoint value.

b. With the number of OPERABLE channels one less than required by the

   Minimum OPERABLE Channels per Trip System requirement for one or both trip systems, place the inoperable channel(s) in the tripped condition within 12 hours.

c. With the number of OPERABLE channels two or more less than required by the Minimum OPERABLE Channels per Trip System requirement for one trip system and:

   1. If the inoperable channels consist of one turbine control valve channel and one turbine stop valve channel, place both inoperable channels in the tripped condition within 12 hours.

   2. If the inoperable channels include two turbine control valve channels or two turbine stop valve channels, declare the trip system inoperable.

d. With one trip system inoperable, restore the inoperable trip system to OPERABLE status within 72 hours or take the ACTION required by Specification 3.2.3.

e. With both trip systems inoperable, restore at least one trip system to OPERABLE status within one hour or take the ACTION required by Specification 3.2.3.

TABLE 3.3.4.2-1

END-OF-CYCLE RECIRCULATION PUMP TRIP SYSTEM INSTRUMENTATION

TRIP FUNCTION	MINIMUM OPERABLE CHANNELS PER TRIP SYSTEM*
1. Turbine Stop Valve - Closure	2**
2. Turbine Control Valve-Fast Closure	2**

* A trip system may be placed in an inoperable status for up to 6 hours for required surveillance provided that the other trip system is OPERABLE.

** This function shall be automatically bypassed when turbine first stage pressure is equivalent to THERMAL POWER LESS than ~~30%~~ 29.5% of RATED THERMAL POWER.

TABLE 3.3.4.2-1

END-OF-CYCLE RECIRCULATION PUMP TRIP SYSTEM INSTRUMENTATION

TRIP FUNCTION	MINIMUM OPERABLE CHANNELS PER TRIP SYSTEM*
1. Turbine Stop Valve - Closure	2**
2. Turbine Control Valve-Fast Closure	2**

* A trip system may be placed in an inoperable status for up to 6 hours for required surveillance provided that the other trip system is OPERABLE.

** This function shall be automatically bypassed when turbine first stage pressure is equivalent to THERMAL POWER LESS than ~~30%~~ 29.5% of RATED THERMAL POWER.

TABLE 3.3.6-2
CONTROL ROD BLOCK INSTRUMENTATION SETPOINTS

TRIP FUNCTION	TRIP SETPOINT	ALLOWABLE VALUE
1. ROD BLOCK MONITOR		
a. Upscale[a]		
1) Low Trip Setpoint (LTSP)	*	*
2) Intermediate Trip Setpoint (ITSP)	*	*
3) High Trip Setpoint (HTSP)	*	*
b. Inoperative	N/A	N/A
c. Downscale (DTSP)	*	*
d. Power Range Setpoint[b]		
1) Low Power Setpoint (LPSP)	28.1% RATED THERMAL POWER	28.4% RATED THERMAL POWER
2) Intermediate Power Setpoint (IPSP)	63.1% RATED THERMAL POWER	63.4% RATED THERMAL POWER
3) High Power Setpoint (HPSP)	83.1% RATED THERMAL POWER	83.4% RATED THERMAL POWER
2. APRM		
a. Simulated Thermal Power - Upscale:		
- Two Recirculation Loop Operation	≤ ~~0.66 W + 55.2%~~ 0.65 W + 54.3% and ≤ 108.0% of RATED THERMAL POWER	≤ ~~0.66 W + 55.7%~~ 0.65 W + 54.7% and ≤ 108.4% of RATED THERMAL POWER
- Single Recirculation Loop Operation****	≤ ~~0.66 (W-7.6%) + 55.2%~~ 0.65 (W-7.6%) + 54.1% and ≤ 108.0% of RATED THERMAL POWER	≤ ~~0.66 (W-7.6%) + 55.7%~~ 0.65 (W-7.6%) + 54.5% and ≤ 108.4% of RATED THERMAL POWER
b. Inoperative	N.A.	N.A.
c. Neutron Flux - Downscale	≥ 3.2% of RATED THERMAL POWER	≥ 2.8% of RATED THERMAL POWER
d. Simulated Thermal Power - Upscale (Setdown)	≤ 12.0% of RATED THERMAL POWER	≤ 13.0% of RATED THERMAL POWER
e. Recirculation Flow - Upscale	*	*
f. LPRM Low Count	< 20 per channel < 3 per axial level	< 20 per channel < 3 per axial level
3. SOURCE RANGE MONITORS		
a. Detector not full in	N.A.	N.A.
b. Upscale	≤ 1×10^5 cps	≤ 1.6×10^5 cps
c. Inoperative	N.A.	N.A.
d. Downscale	≥ 3 cps**	≥ 1.8 cps**

TABLE 3.3.6-2
CONTROL ROD BLOCK INSTRUMENTATION SETPOINTS

TRIP FUNCTION	TRIP SETPOINT	ALLOWABLE VALUE
1. ROD BLOCK MONITOR		
a. Upscale[a]		
1) Low Trip Setpoint (LTSP)	*	*
2) Intermediate Trip Setpoint (ITSP)	*	*
3) High Trip Setpoint (HTSP)	*	*
b. Inoperative	N/A	N/A
c. Downscale (DTSP)	*	*
d. Power Range Setpoint[b]		
1) Low Power Setpoint (LPSP)	28.1% RATED THERMAL POWER	28.4% RATED THERMAL POWER
2) Intermediate Power Setpoint (IPSP)	63.1% RATED THERMAL POWER	63.4% RATED THERMAL POWER
3) High Power Setpoint (HPSP)	83.1% RATED THERMAL POWER	83.4% RATED THERMAL POWER
2. APRM		
a. Simulated Thermal Power - Upscale:		
- Two Recirculation Loop Operation	≤ ~~0.66 W + 55.2%~~ 0.65 W + 54.3% and ≤ 108.0% of RATED THERMAL POWER	≤ ~~0.66 W + 55.7%~~ 0.65 W + 54.7% and ≤ 108.4% of RATED THERMAL POWER
- Single Recirculation Loop Operation****	≤ ~~0.66 (W-7.6%) + 55.2%~~ 0.65 (W-7.6%) + 54.1% and ≤ 108.0% of RATED THERMAL POWER	≤ ~~0.66 (W-7.6%) + 55.7%~~ 0.65 (W-7.6%) + 54.5% and ≤ 108.4% of RATED THERMAL POWER
b. Inoperative	N.A.	N.A.
c. Neutron Flux - Downscale	≥ 3.2% of RATED THERMAL POWER	≥ 2.8% of RATED THERMAL POWER
d. Simulated Thermal Power - Upscale (Setdown)	≤ 12.0% of RATED THERMAL POWER	≤ 13.0% of RATED THERMAL POWER
e. Recirculation Flow - Upscale	*	*
f. LPRM Low Count	< 20 per channel < 3 per axial level	< 20 per channel < 3 per axial level
3. SOURCE RANGE MONITORS		
a. Detector not full in	N.A.	N.A.
b. Upscale	≤ 1×10^5 cps	≤ 1.6×10^5 cps
c. Inoperative	N.A.	N.A.
d. Downscale	≥ 3 cps**	≥ 1.8 cps**

3/4.4 REACTOR COOLANT SYSTEM

3/4.4.1 RECIRCULATION SYSTEM

RECIRCULATION LOOPS

LIMITING CONDITION FOR OPERATION

3.4.1.1 Two reactor coolant system recirculation loops shall be in operation.

APPLICABILITY: OPERATIONAL CONDITIONS 1* and 2*.

ACTION:

a. With one reactor coolant system recirculation loop not in operation:

1. Within 4 hours:

a. Place the recirculation flow control system in the Local Manual mode, and

b. Reduce THERMAL POWER to ≤ ~~76.2%~~ 74.9% of RATED THERMAL POWER, and,

c. Limit the speed of the operating recirculation pump to less than or equal to 90% of rated pump speed, and

d. Verify that the differential temperature requirements of Surveillance Requirement 4.4.1.1.5 are met if THERMAL POWER is ≤ 30% of RATED THERMAL POWER or the recirculation loop flow in the operating loop is ≤ 50% of rated loop flow, or suspend the THERMAL POWER or recirculation loop flow increase.

*See Special Test Exception 3.10.4.

3/4.4 REACTOR COOLANT SYSTEM

3/4.4.1 RECIRCULATION SYSTEM

RECIRCULATION LOOPS

LIMITING CONDITION FOR OPERATION

3.4.1.1 Two reactor coolant system recirculation loops shall be in operation.

APPLICABILITY: OPERATIONAL CONDITIONS 1* and 2*.

ACTION:

a. With one reactor coolant system recirculation loop not in operation:

1. Within 4 hours:

a. Place the recirculation flow control system in the Local Manual mode, and

b. Reduce THERMAL POWER to ≤ ~~76.2%~~ 74.9% of RATED THERMAL POWER, and,

c. Limit the speed of the operating recirculation pump to less than or equal to 90% of rated pump speed, and

d. Verify that the differential temperature requirements of Surveillance Requirement 4.4.1.1.5 are met if THERMAL POWER is ≤ 30% of RATED THERMAL POWER or the recirculation loop flow in the operating loop is ≤ 50% of rated loop flow, or suspend the THERMAL POWER or recirculation loop flow increase.

*See Special Test Exception 3.10.4.

SURVEILLANCE REQUIREMENTS

4.4.1.1.1 DELETED

4.4.1.1.2 DELETED

4.4.1.1.3 DELETED

4.4.1.1.4 With one reactor coolant system recirculation loop not in operation, in accordance with the Surveillance Frequency Control Program, verify that:

a. Reactor THERMAL POWER is ≤ ~~76.2%~~ 74.9% of RATED THERMAL POWER,

b. The recirculation flow control system is in the Local Manual mode, and

c. The speed of the operating recirculation pump is ≤ 90% of rated pump speed.

4.4.1.1.5 With one reactor coolant system recirculation loop not in operation, within 15 minutes prior to either THERMAL POWER increase or recirculation loop flow increase, verify that the following differential temperature requirements are met if THERMAL POWER is ≤ 30% of RATED THERMAL POWER or the recirculation loop flow in the operating recirculation loop is ≤ 50% of rated loop flow.

a. ≤ 145°F between reactor vessel steam space coolant and bottom head drain line coolant,

b. ≤ 50°F between the reactor coolant within the loop not in operation and the coolant in the reactor pressure vessel, and

c. ≤ 50°F between the reactor coolant within the loop not in operation and the operating loop.

The differential temperature requirements of Specification 4.4.1.1.5b. and c. do not apply when the loop not in operation is isolated from the reactor pressure vessel.

SURVEILLANCE REQUIREMENTS

4.4.1.1.1 DELETED

4.4.1.1.2 DELETED

4.4.1.1.3 DELETED

4.4.1.1.4 With one reactor coolant system recirculation loop not in operation, in accordance with the Surveillance Frequency Control Program, verify that:

a. Reactor THERMAL POWER is ≤ ~~76.2%~~ 74.9% of RATED THERMAL POWER,

b. The recirculation flow control system is in the Local Manual mode, and

c. The speed of the operating recirculation pump is ≤ 90% of rated pump speed.

4.4.1.1.5 With one reactor coolant system recirculation loop not in operation, within 15 minutes prior to either THERMAL POWER increase or recirculation loop flow increase, verify that the following differential temperature requirements are met if THERMAL POWER is ≤ 30% of RATED THERMAL POWER or the recirculation loop flow in the operating recirculation loop is ≤ 50% of rated loop flow.

a. ≤ 145°F between reactor vessel steam space coolant and bottom head drain line coolant,

b. ≤ 50°F between the reactor coolant within the loop not in operation and the coolant in the reactor pressure vessel, and

c. ≤ 50°F between the reactor coolant within the loop not in operation and the operating loop.

The differential temperature requirements of Specification 4.4.1.1.5b. and c. do not apply when the loop not in operation is isolated from the reactor pressure vessel.

The SLCS is a special "plant capability" event system. No single active component failure of any plant system or component would necessitate the operational function of the SLCS. It is included for two special cases:

a. Plant capability to shut down the reactor without control rods from normal operation (Section 15.9)

b. Plant capability to shut down the reactor without control rods from a transient incident (Sections 15.8 and 15.9)

c. Plant capability to maintain suppression pool water inventory at a pH of 7.0 or greater post LOCA in accordance with Regulatory Guide 1.183, Alternative Source Terms.

Although this system has been designed to a high degree of reliability with many safety system features, it is not required to meet the safety design basis requirements of the safety systems.

7.4.1.2.3.2 SLCS Initiating Circuits

The SLCS is manually initiated in the control room, when the operator determines that normal reactivity control systems have not shutdown the reactor as required, by turning the key-lock switch to the run position for ~~Loop A, B, or C. The key is removable in the center normal position.~~ Indicator lights located near each key-lock switch apprise the operator of the selected system initiation.

either one or both of the two in-service pumps.

Two

~~All three~~ loops of the SLCS can also be automatically initiated by the RRCS after a time delay, provided that APRM power is not downscale. This automatic initiation overrides the manual initiation signal; however, the manual shutoff signal overrides the automatic initiation signals. Section 7.6.1.8.3.4 describes the RRCS automatic initiation of the SLCS.

and the key-lock switch is in the center normal position

7.4.1.2.3.3 SLCS Logic and Sequencing

a

(manually or automatically)

manually

When SLCS pump is started, both squibs on its explosive- operated valve fire~~, opening the valve~~. Firing of either or both of the two squibs installed on each valve will open the valve. The SLCS is initiated in the control room by turning the key-locked switch for ~~system A, B, or C~~ to the run position. ~~The key is removable in the center normal position.~~

UFSAR insert 1

either one or both of the two in-service pumps

~~For automatic operation, both squibs on each injection valve fire, all three SLCS pumps receive a start signal when both channels A and B of either RRCS division are tripped.~~

UFSAR insert 2

The SLCS is provided with instrumentation and control to automatically shut off the SLCS pumps when the solution level in the storage tank is below the low level limit. This low level pump shutoff signal is provided by two-out-of-two logic. Three sets of storage tank level monitoring devices are provided to automatically shut off the SLCS pumps. Each set consists of two independent transmitters and trip units. There is a separate external line for each set of transmitters; this prevents a single instrument line problem from affecting all three SLCS pumps.

7.4.1.2.3.4 SLCS Bypasses and Interlocks

UFSAR Insert 1

Only two SLCS pumps are started to prevent lifting the pressure relief valve(s) on high discharge header pressure with three SLCS pump operation.

UFSAR Insert 2

SLCS operates automatically when both channels A and B of either RRCS division are tripped. Normally, only SLCS pumps A and B are aligned to receive the start signal (key-lock switches in the normal position), with the SLCS pump C automatic start signal blocked (key-lock switch in the stop position and the key removed). However, when either the A or B SLCS pump is out-of-service, then the C SLCS pump can be aligned to receive the automatic start signal (key-lock switch positioned to normal).

Both sensors in either division (i.e., two level sensors in one division or two pressure sensors in one division) are required to generate a trip signal. The ATWS RPT pump breakers are the same ones used in the EOC RPT. There are two breakers in series in each pump motor feed; the control logic of each breaker is assigned to a separate safety division.

Manual initiation of RRCS without reactor high pressure or reactor low level 2 does not trip the recirculation pump drive motor breaker; however, after manual initiation of RRCS, the breaker trip will occur if either reactor high pressure or low level 2 occur.

The ATWS RPT trip circuitry is separate from and independent of the EOC RPT trip circuitry. Separate trip coils are used in each breaker (one for ATWS RPT and one for EOC RPT). The trip coils are fed from RPS power supplies.

The trip circuits, including the sensors and the pump breakers, are Class 1E. The entire trip circuits may be tested during plant operation, except for opening of the pump breakers. ATWS RPT circuitry is separated from non-Class 1E circuitry in accordance with the LGS separation criteria.

Indicators and annunciators in the control room provide the status of the trip coils and the mechanical position of the pump circuit breakers. Actuation of the ATWS-RPT is recorded in the control room.

7.6.1.8.3.3 RRCS Feedwater Runback

The feedwater runback function mitigates the consequences of an ATWS event by stopping feedwater flow into the vessel, which reduces the core subcooling, thereby reducing the core power generation.

Reactor high pressure combined with a 25 second time delay and APRM power not downscale will initiate a feedwater runback. Feedwater flow will be reduced to 0% within 15 seconds. The logic to initiate feedwater runback is energized to trip and can be manually overridden 30 seconds after runback initiation. The runback reduces the input of cooler water flowing to the vessel. As average core coolant temperature increases, voids increase, reactivity decreases, and power is reduced.

The RRCS feedwater runback will occur whether the feed pumps are in automatic or in manual mode of control. The normal loss of signal interlock that prohibits changes in feedwater pump output during loss of signal conditions is disabled during ATWS.

7.6.1.8.3.4 Standby Liquid Control System Initiation

Low water level 2, reactor high pressure, or manual initiation of the RRCS immediately starts a timer. A signal will be sent to initiate the SLCS if, at the expiration of a 118 second time delay, the core power is not downscale as measured by the APRM system. Initiation of the SLCS requires start signals from both channels A and B of either division of RRCS. Receipt of these signals causes the squibs to fire, ~~opens~~ the explosive valves~~, and starts the SLCS pumps~~. ~~All three~~ pumps will inject borated water into the vessel until the storage tank low level sensors, arranged in two-out-of-two logic trip~~,~~ the pumps.

associated

opening

Both

starts the two in-service pumps and

The SLCS pump control switches can be used to manually stop SLCS pump injection.

neutron absorber solution, can be injected into the reactor to test the operation of all components of the system.

e. The neutron absorber will be dispersed within the reactor core in sufficient quantity to provide a reasonable margin for dilution, leakage, and imperfect mixing.

f. The system is reliable to a degree consistent with its role as a special safety system; the possibility of unintentional or accidental shutdown of the reactor by this system is minimized.

g. The system shall be capable of injecting sodium pentaborate into the reactor vessel to maintain the suppression pool water inventory at a pH of 7.0 or higher following a LOCA.

9.3.5.2 System Description

The SLCS (drawing M-48) is designed to be manually initiated from the control room to cause a sodium pentaborate solution to be pumped into the reactor if the operator determines that the reactor cannot be shut down or kept shut down with the control rods.

The SLCS is also designed to be manually initiated from the control room to pump sodium pentaborate into the reactor within 13 hours of the onset of a large break LOCA to maintain suppression pool pH at a level of 7.0 or higher. (Ref. 9.3-4)

The SLCS is also automatically initiated upon receipt of a signal from the RRCS logic. The sodium pentaborate solution is injected through the core spray line and sparger, which is used by the HPCI system. With two pumps operating, the SLCS can begin to deliver the control liquid to the RPV within about 53 seconds after actuation. ~~To further ensure the system performance, the~~ The actuation signal starts ~~all three~~ the two in-service pumps and actuates ~~all three~~ the associated explosive valves. However, the SLCS is not expected to be actuated because the CRD system, backed up by the ARI design, can shut down the reactor when required. Low vessel water level (level 2), high vessel pressure, or manual initiation of the RRCS starts a timer. If the core power is not downscale at the end of a predetermined time-delay, the SLCS injection as described above is initiated.

The boron solution tank, the test water tank, the three positive displacement pumps, the three explosive valves, the motor-operated stop-check shutoff valve, and associated local valves and controls are located in the reactor enclosure. The solution is pumped into the core spray line downstream of the air-operated testable check valve that leads to sparger B. It is sprayed radially over the top of the core (Sections 3.9.3, 3.9.5, 5.3, and 6.3.2).

The boron absorbs thermal neutrons and, when present in sufficient quantity in the reactor, will cause the reactor to become subcritical.

The specified neutron absorber solution is an aqueous solution of sodium pentaborate decahydrate ($Na_2B_{10}O_{16}\bullet 10H_20$). It is prepared by dissolving either sodium pentaborate decahydrate or stoichiometric quantities of borax and boric acid in the SLCS tank with demineralized water so that the solution fills the tank to at least the low level alarm point. The solution can be diluted with water to allow for evaporation losses or to lower the solution saturation temperature, provided that solution concentration requirements are met. An air sparger is provided in the tank for mixing. To prevent system plugging, the tank outlet is raised above the bottom of the tank.