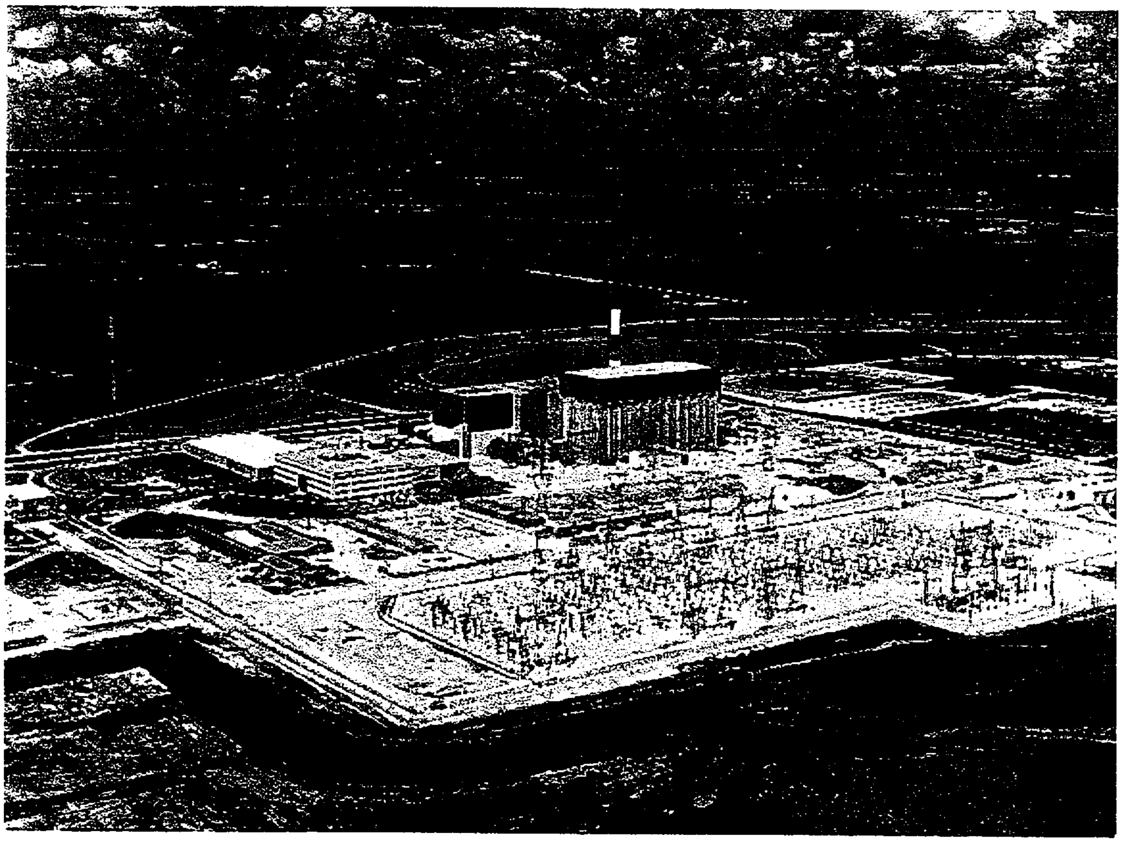
Improved Technical Specifications



LaSalle County Station

Volume 5:
Sections 3.4 and 3.5

ComEd

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.1 Recirculation Loops Operating

LCO 3.4.1 Two recirculation loops with matched flows shall be in operation within Region III of Figure 3.4.1-1,

OR

One recirculation loop shall be in operation within Region III of Figure 3.4.1-1 with the following limits applied when the associated LCO is applicable:

a. LCO 3.2.1, "AVERAGE PLANAR LINEAR HEAT GENERATION RATE (APLHGR)," single loop operation limits specified in the COLR;

b. LCO 3.2.2, "MINIMUM CRITICAL POWER RATIO (MCPR)," single loop operation limits specified in the COLR;

c. LCO 3.3.1.1, "Reactor Protection System (RPS) Instrumentation," Function 2.b (Average Power Range Monitors Flow Biased Simulated Thermal Power—Upscale), Allowable Value of Table 3.3.1.1-1 is reset for single loop operation; and

d. LCO 3.3.2.1, "Control Rod Block Instrumentation," Function 1.a (Rod Block Monitor - Upscale), Allowable Value of Table 3.3.2.1-1, specified in the COLR, is reset for single loop operation.

APPLICABILITY: MODES 1 and 2.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or two recirculation loops operating within Region II of Figure 3.4.1-1.	A.1 -----------NOTE---------- Only applicable when 3 times baseline value is > 10% peak-to-peak value. ------------------------- Verify APRM and LPRM flux noise levels ≤ 3 times baseline.	45 minutes AND Once per 12 hours thereafter AND 45 minutes from discovery of Condition A concurrent with any THERMAL POWER increase of ≥ 5% RTP.
	AND	
		(continued)

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. (continued)	A.2 -----------NOTE---------- Only applicable when 10% peak-to-peak value is ≥ 3 times baseline value. ------------------------ Verify APRM and LPRM flux noise levels ≤ 10% peak-to-peak.	45 minutes AND Once per 12 hours thereafter AND 45 minutes from discovery of Condition A concurrent with any THERMAL POWER increase of ≥ 5% RTP.
	AND A.3 Verify recirculation loop(s) are not operating in Region I of Figure 3.4.1-1.	Once per 12 hours
B. Required Action A.1 or A.2 and associated Completion Time not met.	B.1 Satisfy the requirements of the LCO.	2 hours

(continued)

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
C. One or two recirculation loops operating within Region I of Figure 3.4.1-1. <u>OR</u> Required Action A.3 and associated Completion Time not met.	C.1 Exit Region I of Figure 3.4.1-1.	2 hours
D. No recirculation loops in operation.	D.1 Verify APRM and LPRM flux noise levels $\leq$ 10% peak-to-peak. <u>AND</u> D.2 Reduce THERMAL POWER to < 36% RTP. <u>AND</u> D.3 Be in MODE 3.	Immediately 2 hours 12 hours
E. Required Action B.1 or D.1 and associated Completion Time not met.	E.1 Place the mode switch in the shutdown position.	Immediately
F. Recirculation loop flow mismatch not within limits.	F.1 Declare the recirculation loop with lower flow to be "not in operation."	2 hours

(continued)

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
G. Requirements of the LCO not met for reasons other than Condition A, C, D, and F.	G.1 Satisfy the requirements of the LCO.	12 hours
H. Required Action and associated Completion Time of Condition G not met.	H.1 Be in MODE 3.	12 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.4.1.1	-------------------NOTE------------------- Not required to be performed until 24 hours after both recirculation loops are in operation. -- Verify recirculation loop jet pump flow mismatch with both recirculation loops in operation is: a. $\leq$ 10% of rated core flow when operating at < 70% of rated core flow; and b. $\leq$ 5% of rated core flow when operating at $\geq$ 70% of rated core flow.	24 hours
SR 3.4.1.2	Verify operation is in Region III of Figure 3.4.1-1.	24 hours

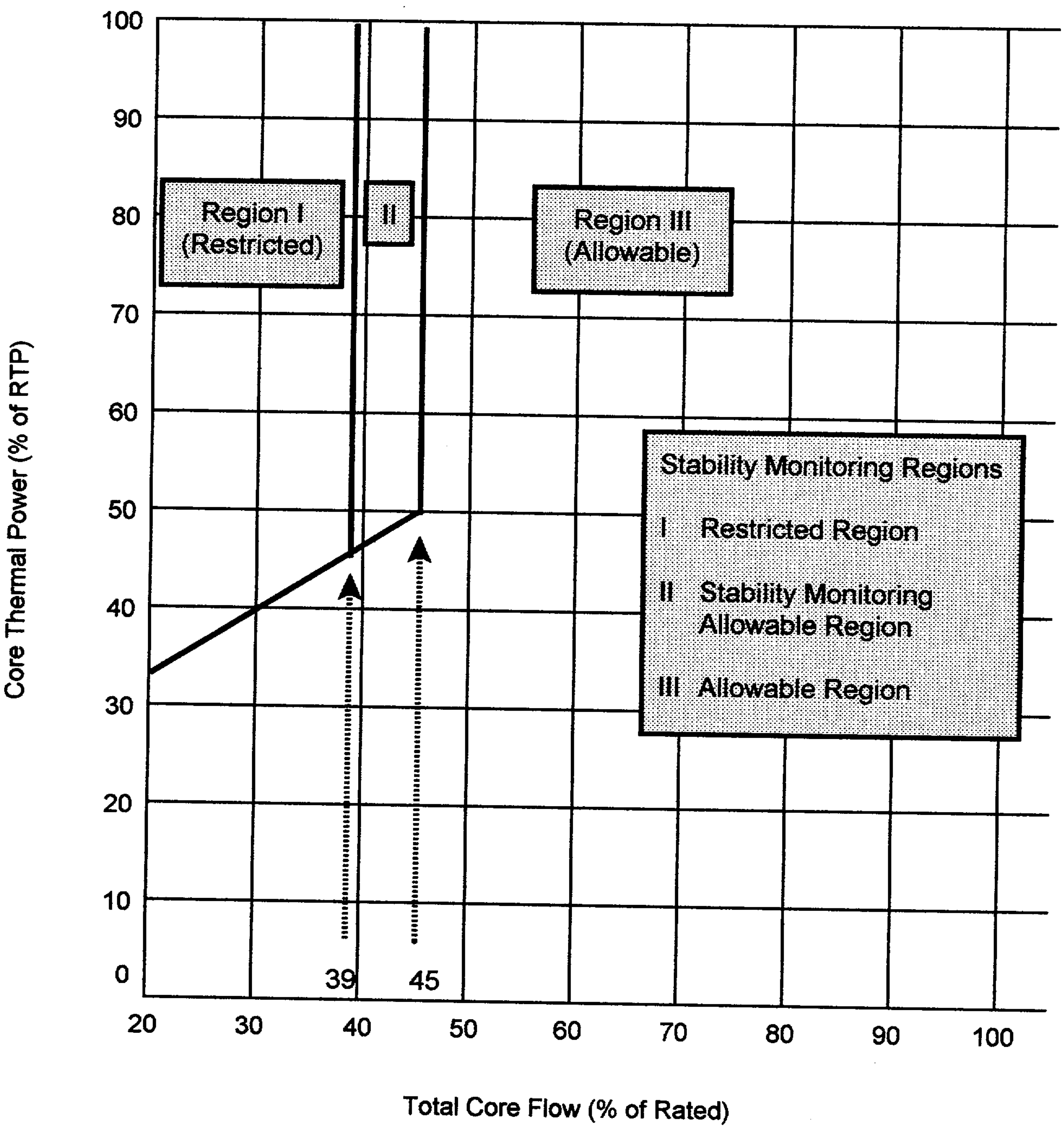


Figure 3.4.1-1 (Page 1 of 1)
Power versus Flow

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.2 Flow Control Valves (FCVs)

LCO 3.4.2 A recirculation loop FCV shall be OPERABLE in each operating recirculation loop.

APPLICABILITY: MODES 1 and 2.

ACTIONS

-----------------------------------NOTE-----------------------------------
Separate Condition entry is allowed for each FCV.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or two required FCVs inoperable.	A.1 Lock up the FCV.	4 hours
B. Required Action and associated Completion Time not met.	B.1 Be in MODE 3.	12 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
SR 3.4.2.1 Verify each FCV fails "as is" on loss of hydraulic pressure at the hydraulic unit.	24 months

(continued)

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.4.2.2	Verify average rate of each FCV movement is: a. $\leq$ 11% of stroke per second for opening; and b. $\leq$ 11% of stroke per second for closing.	24 months

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.3 Jet Pumps

LCO 3.4.3 All jet pumps shall be OPERABLE.

APPLICABILITY: MODES 1 and 2.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more jet pumps inoperable.	A.1 Be in MODE 3.	12 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.4.3.1	------------------NOTES------------------ 1. Not required to be performed until 4 hours after associated recirculation loop is in operation. 2. Not required to be performed until 24 hours after > 25% RTP. --- Verify at least two of the following criteria (a, b, and c) are satisfied for each operating recirculation loop: a. Recirculation loop drive flow versus flow control valve position differs by ≤ 10% from established patterns. b. Indicated total core flow versus calculated total core flow differs by ≤ 10% from established patterns. c. Each jet pump diffuser to lower plenum differential pressure differs by ≤ 20% from established patterns.	24 hours

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.4 Safety/Relief Valves (S/RVs)

LCO 3.4.4 The safety function of 17 S/RVs for Unit 1, and 12 S/RVs for Unit 2, shall be OPERABLE.

APPLICABILITY: MODES 1, 2, and 3.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more required S/RVs inoperable.	A.1 Be in MODE 3. AND A.2 Be in MODE 4.	12 hours 36 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.4.4.1	-------------------NOTE------------------ Less than or equal to two required S/RVs may be changed to a lower setpoint group. --- Verify the safety function lift setpoints of the required S/RVs are as follows: Number of Unit 1 S/RVs — Setpoint (psig) 4 — 1205 ± 36.1 4 — 1195 ± 35.8 4 — 1185 ± 35.5 4 — 1175 ± 35.2 2 — 1150 ± 34.5 Number of Unit 2 S/RVs — Setpoint (psig) 2 — 1205 ± 36.1 3 — 1195 ± 35.8 2 — 1185 ± 35.5 4 — 1175 ± 35.2 2 — 1150 ± 34.5 Following testing, lift settings shall be within ± 1%.	In accordance with the Inservice Testing Program

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.5 RCS Operational LEAKAGE

LCO 3.4.5 RCS operational LEAKAGE shall be limited to:

a. No pressure boundary LEAKAGE;

b. ≤ 5 gpm unidentified LEAKAGE;

c. ≤ 25 gpm total LEAKAGE averaged over the previous 24 hour period; and

d. ≤ 2 gpm increase in unidentified LEAKAGE within the previous 24 hour period in MODE 1.

APPLICABILITY: MODES 1, 2, and 3.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Unidentified LEAKAGE not within limit. <u>OR</u> Total LEAKAGE not within limit.	A.1 Reduce LEAKAGE to within limits.	4 hours
B. Unidentified LEAKAGE increase not within limit.	B.1 Reduce unidentified LEAKAGE increase to within limit. <u>OR</u>	4 hours (continued)

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
B. (continued)	B.2 Identify source of unidentified LEAKAGE increase is not intergranular stress corrosion cracking susceptible material.	4 hours
C. Required Action and associated Completion Time of Condition A or B not met. <u>OR</u> Pressure boundary LEAKAGE exists.	C.1 Be in MODE 3. <u>AND</u> C.2 Be in MODE 4.	12 hours 36 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
SR 3.4.5.1 Verify RCS unidentified and total LEAKAGE and unidentified LEAKAGE increase are within limits.	12 hours

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.6 RCS Pressure Isolation Valve (PIV) Leakage

LCO 3.4.6 The leakage from each RCS PIV shall be within limit.

APPLICABILITY: MODES 1 and 2,
MODE 3, except valves in the residual heat removal shutdown cooling flow path when in, or during the transition to or from, the shutdown cooling mode of operation.

ACTIONS

------------------------------------NOTES------------------------------------

1. Separate Condition entry is allowed for each flow path.

2. Enter applicable Conditions and Required Actions for systems made inoperable by PIVs.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more flow paths with leakage from one or more RCS PIVs not within limit.	------------NOTE------------ Each valve used to satisfy Required Action A.1 and Required Action A.2 shall have been verified to meet SR 3.4.6.1 and be in the reactor coolant pressure boundary or the high pressure portion of the system. ----------------------------	
		(continued)

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. (continued)	A.1	Isolate the high pressure portion of the affected system from the low pressure portion by use of one closed manual, deactivated automatic, or check valve.	4 hours
	<u>AND</u>		
	A.2	Isolate the high pressure portion of the affected system from the low pressure portion by use of a second closed manual, deactivated automatic, or check valve.	72 hours
B. Required Action and associated Completion Time not met.	B.1	Be in MODE 3.	12 hours
	<u>AND</u>		
	B.2	Be in MODE 4.	36 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.4.6.1	------------------NOTE------------------ Only required to be performed in MODES 1 and 2. -- Verify equivalent leakage of each RCS PIV is $\leq$ 0.5 gpm per nominal inch of valve size up to a maximum of 5 gpm, at an RCS pressure $\geq$ 950 psig and $\leq$ 1050 psig.	In accordance with the Inservice Testing Program

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.7 RCS Leakage Detection Instrumentation

LCO 3.4.7 The following RCS leakage detection instrumentation shall be OPERABLE:

a. Drywell floor drain sump flow monitoring system;

b. One channel of either drywell atmospheric particulate or atmospheric gaseous monitoring system; and

c. Drywell air cooler condensate flow rate monitoring system.

APPLICABILITY: MODES 1, 2, and 3.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Drywell floor drain sump flow monitoring system inoperable.	------------NOTE------------ LCO 3.0.4 is not applicable. -----------------------------	
	A.1 Restore drywell floor drain sump flow monitoring system to OPERABLE status.	30 days
B. Required drywell atmospheric monitoring system inoperable.	B.1 Analyze grab samples of drywell atmosphere.	Once per 12 hours

(continued)

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
C. Drywell air cooler condensate flow rate monitoring system inoperable.	------------NOTE------------ Not applicable when the required drywell atmospheric monitoring system is inoperable. ---------------------------- C.1 Perform SR 3.4.7.1.	 Once per 8 hours
D. Required drywell atmospheric monitoring system inoperable. AND Drywell air cooler condensate flow rate monitoring system inoperable.	------------NOTE------------ LCO 3.0.4 is not applicable. ---------------------------- D.1 Restore required drywell atmospheric monitoring system to OPERABLE status. OR D.2 Restore drywell air cooler condensate flow rate monitoring system to OPERABLE status.	 30 days 30 days
E. Required Action and associated Completion Time of Condition A, B, C, or D not met.	E.1 Be in MODE 3. AND E.2 Be in MODE 4.	12 hours 36 hours
F. All required leakage detection systems inoperable.	F.1 Enter LCO 3.0.3.	Immediately

SURVEILLANCE REQUIREMENTS

---------------------------------NOTE---------------------------------
When a channel is placed in an inoperable status solely for performance of required Surveillances, entry into associated Conditions and Required Actions may be delayed for up to 6 hours provided the other required leakage detection instrumentation is OPERABLE.

SURVEILLANCE		FREQUENCY
SR 3.4.7.1	Perform CHANNEL CHECK of required drywell atmospheric monitoring system.	12 hours
SR 3.4.7.2	Perform CHANNEL FUNCTIONAL TEST of required leakage detection instrumentation.	31 days
SR 3.4.7.3	Perform CHANNEL CALIBRATION of required leakage detection instrumentation.	24 months

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.8 RCS Specific Activity

LCO 3.4.8 The specific activity of the reactor coolant shall be limited to DOSE EQUIVALENT I-131 specific activity $\leq$ 0.2 μCi/gm.

APPLICABILITY: MODE 1,
MODES 2 and 3 with any main steam line not isolated.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Reactor coolant specific activity > 0.2 μCi/gm and $\leq$ 4.0 μCi/gm DOSE EQUIVALENT I-131.	-------------NOTE----------- LCO 3.0.4 is not applicable. ---------------------------	
	A.1 Determine DOSE EQUIVALENT I-131.	Once per 4 hours
	AND	
	A.2 Restore DOSE EQUIVALENT I-131 to within limits.	48 hours
B. Required Action and associated Completion Time of Condition A not met.	B.1 Determine DOSE EQUIVALENT I-131.	Once per 4 hours
OR	AND	
Reactor coolant specific activity > 4.0 μCi/gm DOSE EQUIVALENT I-131.	B.2.1 Isolate all main steam lines.	12 hours
	OR	
		(continued)

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
B. (continued)	B.2.2.1 Be in MODE 3. AND B.2.2.2 Be in MODE 4.	12 hours 36 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.4.8.1	------------------NOTE------------------- Only required to be performed in MODE 1. --- Verify reactor coolant DOSE EQUIVALENT I-131 specific activity is $\leq$ 0.2 μCi/gm.	7 days

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.9 Residual Heat Removal (RHR) Shutdown Cooling System—Hot Shutdown

LCO 3.4.9 Two RHR shutdown cooling subsystems shall be OPERABLE, and, with no recirculation pump in operation, at least one RHR shutdown cooling subsystem shall be in operation.

---------------------------NOTES---------------------------

1. Both RHR shutdown cooling subsystems and recirculation pumps may be not in operation for up to 2 hours per 8 hour period.

2. One RHR shutdown cooling subsystem may be inoperable for up to 2 hours for performance of Surveillances.

APPLICABILITY: MODE 3 with reactor vessel pressure less than the RHR cut-in permissive pressure.

ACTIONS

-----------------------------------NOTES-----------------------------------

1. LCO 3.0.4 is not applicable.

2. Separate Condition entry is allowed for each RHR shutdown cooling subsystem.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or two RHR shutdown cooling subsystems inoperable.	A.1 Initiate action to restore RHR shutdown cooling subsystem to OPERABLE status. <u>AND</u>	Immediately (continued)

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. (continued)	A.2 Verify an alternate method of decay heat removal is available for each inoperable RHR shutdown cooling subsystem. AND	1 hour
	A.3 Be in MODE 4.	24 hours
B. No RHR shutdown cooling subsystem in operation. AND No recirculation pump in operation.	B.1 Initiate action to restore one RHR shutdown cooling subsystem or one recirculation pump to operation. AND	Immediately
	B.2 Verify reactor coolant circulation by an alternate method. AND	1 hour from discovery of no reactor coolant circulation AND Once per 12 hours thereafter
	B.3 Monitor reactor coolant temperature and pressure.	Once per hour

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.4.9.1	-------------------NOTE------------------- Not required to be met until 2 hours after reactor vessel pressure is less than the RHR cut-in permissive pressure. -- Verify one RHR shutdown cooling subsystem or recirculation pump is operating.	12 hours

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.10 Residual Heat Removal (RHR) Shutdown Cooling System—Cold Shutdown

LCO 3.4.10 Two RHR shutdown cooling subsystems shall be OPERABLE, and, with no recirculation pump in operation, at least one RHR shutdown cooling subsystem shall be in operation.

---------------------------NOTES-------------------------

1. Not required to be met during hydrostatic testing.

2. Both RHR shutdown cooling subsystems and recirculation pumps may be not in operation for up to 2 hours per 8 hour period.

3. One RHR shutdown cooling subsystem may be inoperable for up to 2 hours for the performance of Surveillances.

APPLICABILITY: MODE 4.

ACTIONS

--------------------------------------NOTE--------------------------------------

Separate Condition entry is allowed for each RHR shutdown cooling subsystem.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or two RHR shutdown cooling subsystems inoperable.	A.1 Verify an alternate method of decay heat removal is available for each inoperable RHR shutdown cooling subsystem.	1 hour <u>AND</u> Once per 24 hours thereafter

(continued)

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
B. No RHR shutdown cooling subsystem in operation. AND No recirculation pump in operation.	B.1 Verify reactor coolant circulating by an alternate method.	1 hour from discovery of no reactor coolant circulation AND Once per 12 hours thereafter
	AND B.2 Monitor reactor coolant temperature and pressure.	Once per hour

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
SR 3.4.10.1 Verify one RHR shutdown cooling subsystem or recirculation pump is operating.	12 hours

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.11 RCS Pressure and Temperature (P/T) Limits

LCO 3.4.11 RCS pressure, RCS temperature, RCS heatup and cooldown rates, and the recirculation pump starting temperature requirements shall be maintained within limits.

APPLICABILITY: At all times.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. ---------NOTE--------- Required Action A.2 shall be completed if this Condition is entered. --------------------- Requirements of the LCO not met in MODE 1, 2, or 3.	A.1 Restore parameter(s) to within limits. AND A.2 Determine RCS is acceptable for continued operation.	30 minutes 72 hours
B. Required Action and associated Completion Time of Condition A not met.	B.1 Be in MODE 3. AND B.2 Be in MODE 4.	12 hours 36 hours

(continued)

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
C. ---------NOTE--------- Required Action C.2 shall be completed if this Condition is entered. --------------------- Requirements of the LCO not met in other than MODES 1, 2, and 3.	C.1	Initiate action to restore parameter(s) to within limits.	Immediately
	AND		
	C.2	Determine RCS is acceptable for operation.	Prior to entering MODE 2 or 3

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.4.11.1	------------------NOTE------------------ Only required to be performed during RCS heatup and cooldown operations, and RCS inservice leak and hydrostatic testing. --------------------------------------- Verify: a. RCS pressure and RCS temperature are within the applicable limits specified in Figures 3.4.11-1, 3.4.11-2, 3.4.11-3 for Unit 1, and Figures 3.4.11-4, 3.4.11-5, and 3.4.11-6 for Unit 2; b. RCS heatup and cooldown rates are $\leq$ 100°F in any 1 hour period; and c. RCS temperature change during system leakage and hydrostatic testing is $\leq$ 20°F in any one hour period when the RCS pressure and RCS temperature are not within the limits of Figure 3.4.11-2 for Unit 1 and Figure 3.4.11-5 for Unit 2, as applicable.	30 minutes
SR 3.4.11.2	Verify RCS pressure and RCS temperature are within the applicable criticality limits specified in Figure 3.4.11-3 for Unit 1 and Figure 3.4.11-6 for Unit 2.	Once within 15 minutes prior to control rod withdrawal for the purpose of achieving criticality

(continued)

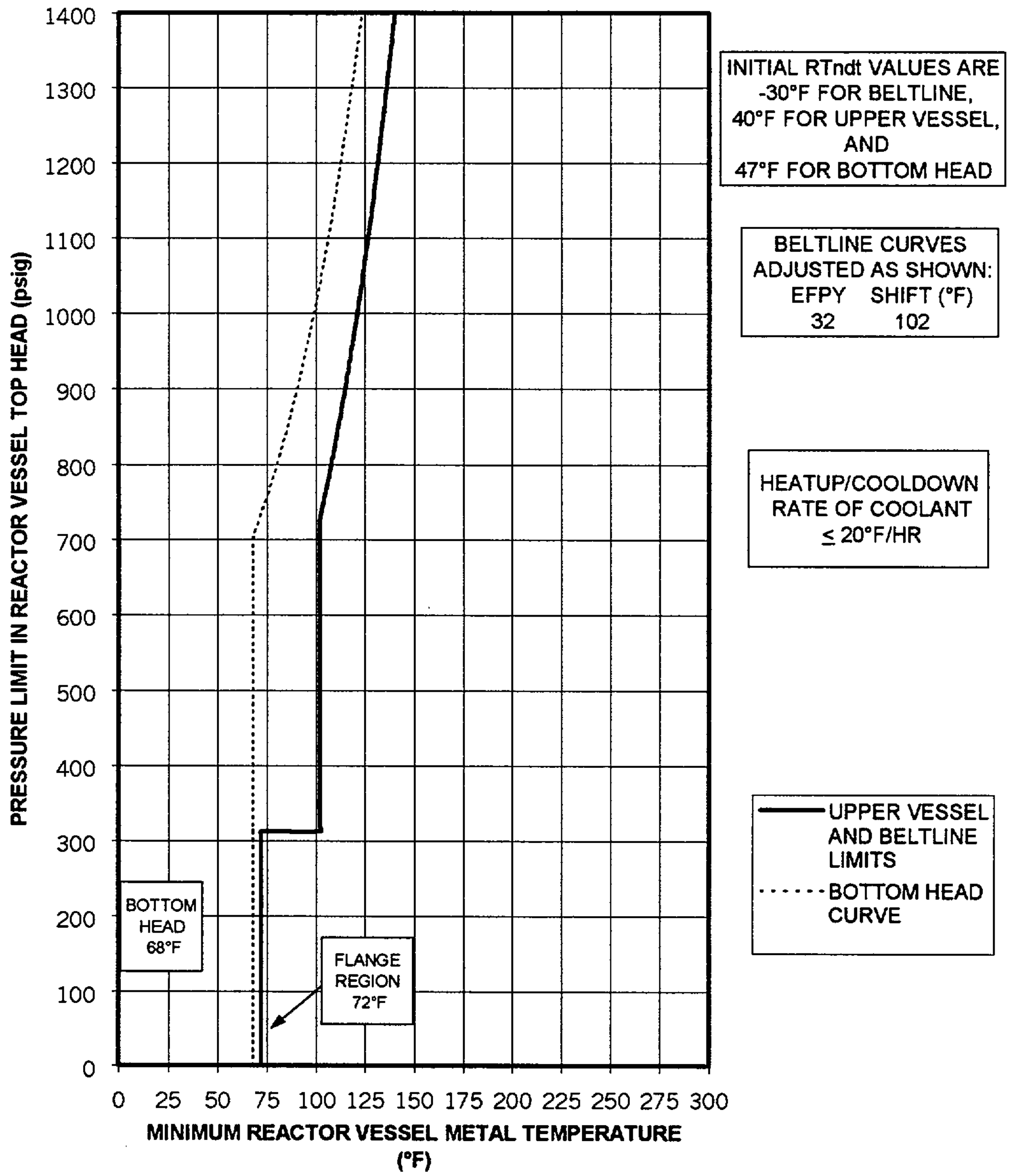
SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.4.11.3	-------------------NOTE------------------- Only required to be met in MODES 1, 2, 3, and 4 during recirculation pump startup. -- Verify the difference between the bottom head coolant temperature and the reactor pressure vessel (RPV) coolant temperature is ≤ 145°F.	Once within 15 minutes prior to each startup of a recirculation pump
SR 3.4.11.4	-------------------NOTE------------------- Only required to be met in MODES 1, 2, 3, and 4 during recirculation pump startup. -- Verify the difference between the reactor coolant temperature in the recirculation loop to be started and the RPV coolant temperature is ≤ 50°F.	Once within 15 minutes prior to each startup of a recirculation pump
SR 3.4.11.5	-------------------NOTE------------------- Only required to be performed when tensioning the reactor vessel head bolting studs. -- Verify reactor vessel flange and head flange temperatures are ≥ 72°F for Unit 1 and ≥ 86°F for Unit 2.	30 minutes

(continued)

SURVEILLANCE REQUIREMENTS

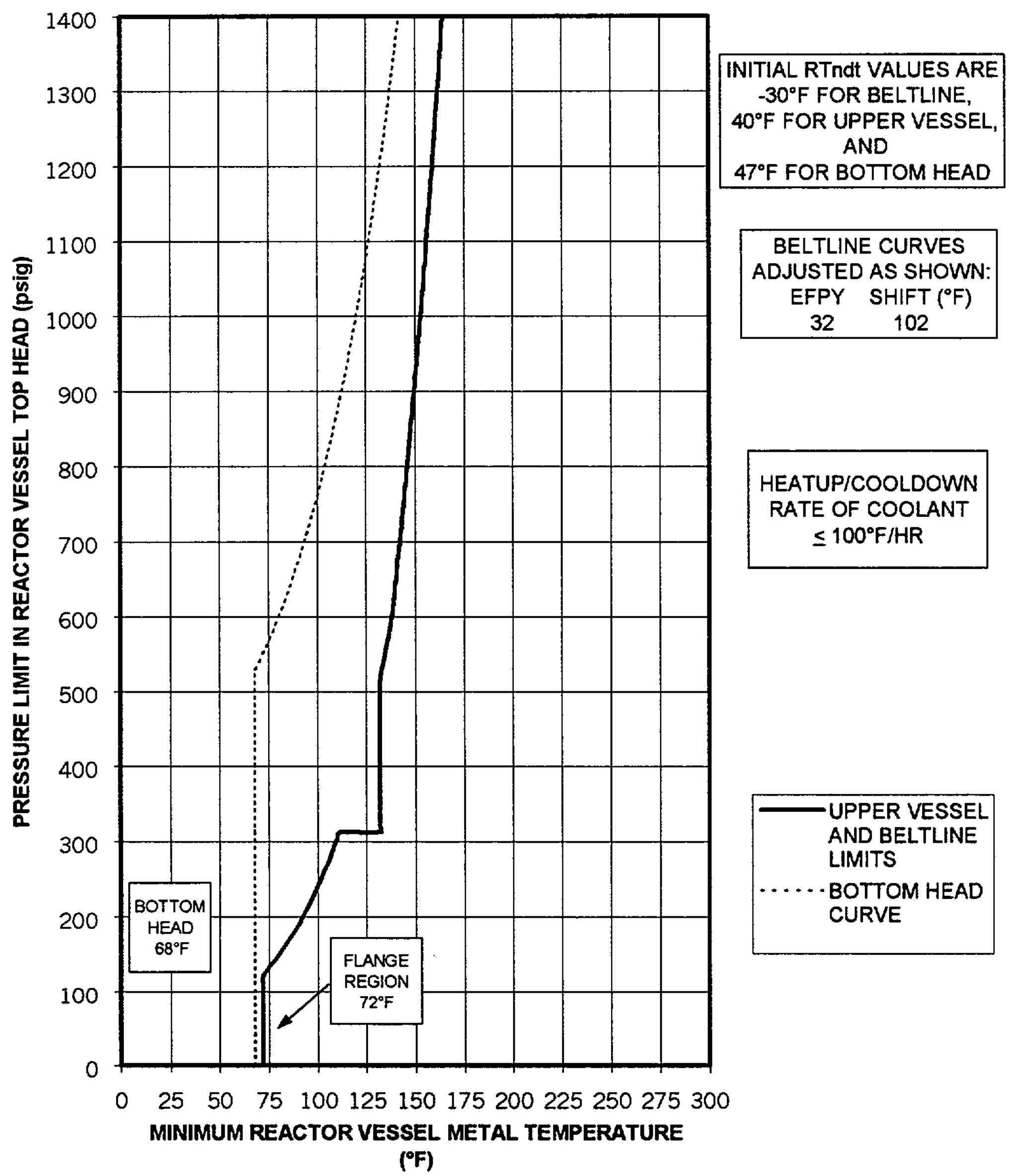
SURVEILLANCE		FREQUENCY
SR 3.4.11.6	------------------NOTE------------------ Not required to be performed until 30 minutes after RCS temperature $\leq$ 77°F for Unit 1 and $\leq$ 91°F for Unit 2 in MODE 4. -- Verify reactor vessel flange and head flange temperatures are $\geq$ 72°F for Unit 1 and $\geq$ 86°F for Unit 2.	30 minutes
SR 3.4.11.7	------------------NOTE------------------ Not required to be performed until 12 hours after RCS temperature $\leq$ 92°F for Unit 1 and $\leq$ 106°F for Unit 2 in MODE 4. -- Verify reactor vessel flange and head flange temperatures are $\geq$ 72°F for Unit 1 and $\geq$ 86°F for Unit 2.	12 hours



P-T Curves for Hydrostatic or Leak Testing

Figure 3.4.11-1 (Page 1 of 1)

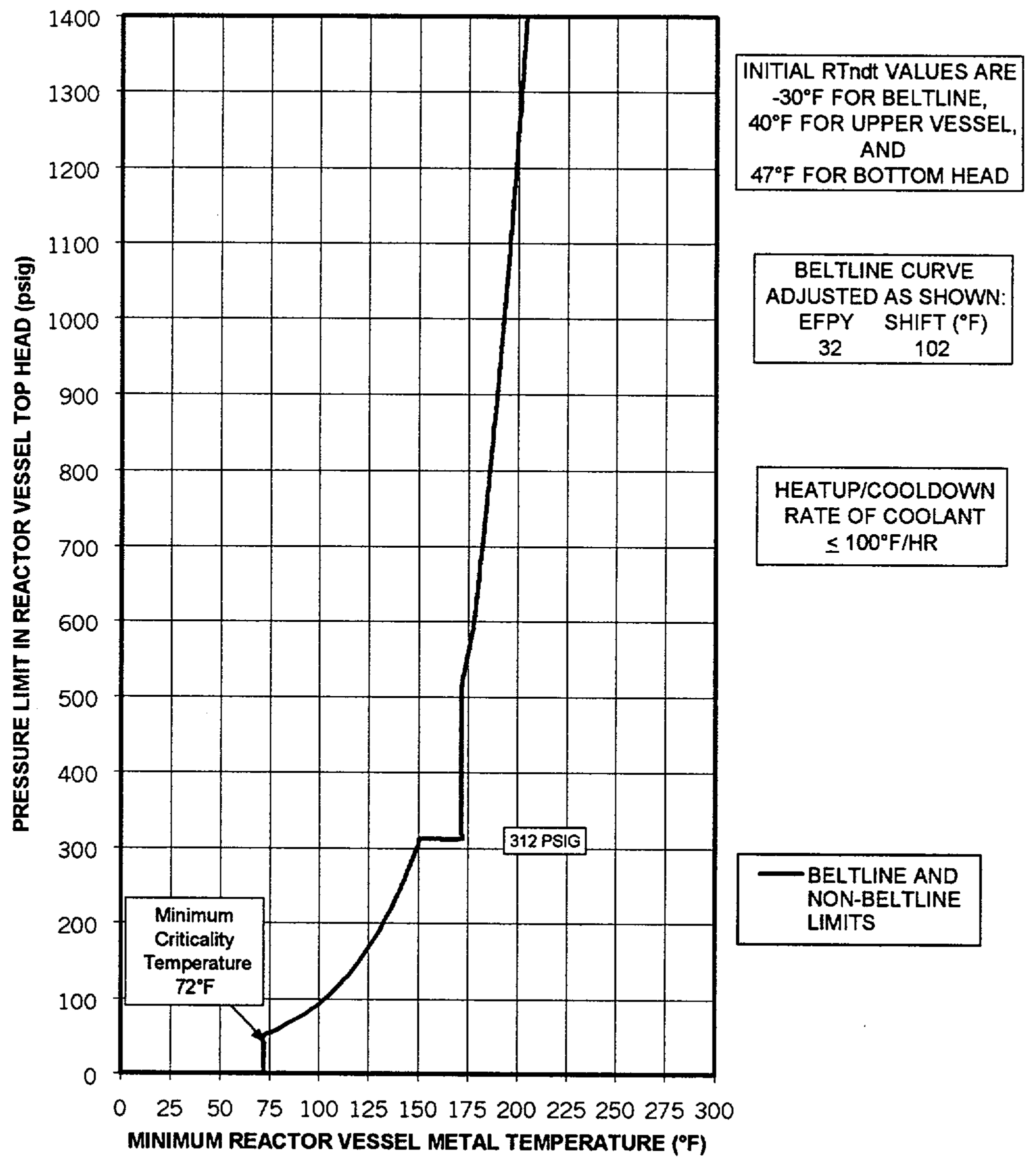
Unit 1
Minimum Reactor Vessel Metal Temperature vs.
Reactor Vessel Pressure (Valid up to 32 EFPY)



P-T Curves for Heatup by Non-Nuclear Means, Cooldown Following A Nuclear Shutdown and Low Power Physics Testing

Figure 3.4.11-2 (Page 1 of 1)

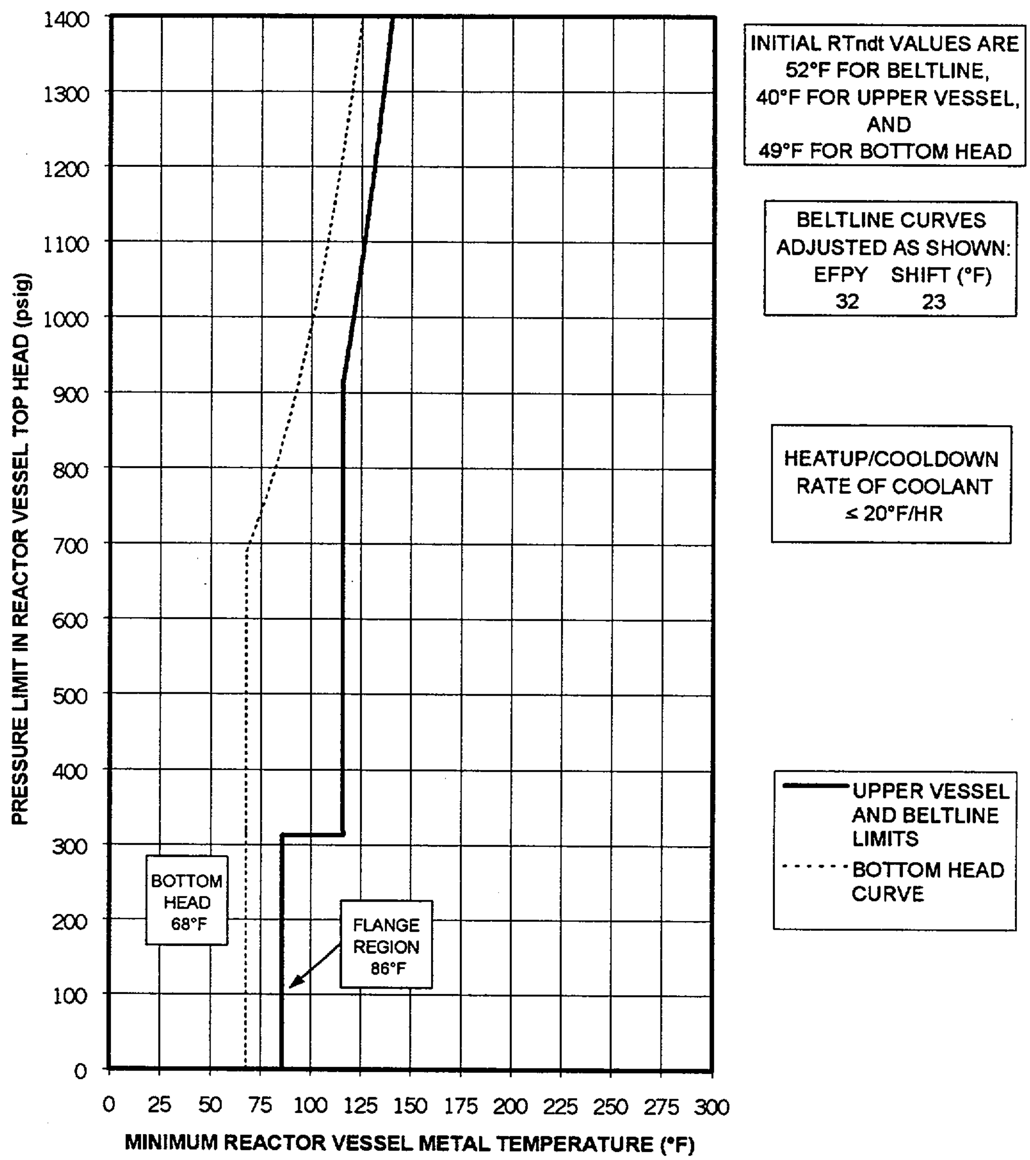
Unit 1
Minimum Reactor Vessel Metal Temperature vs.
Reactor Vessel Pressure (Valid up to 32 EFPY)



P-T Curves for Operation with a Core Critical
other than Low Power Physics Testing

Figure 3.4.11-3 (Page 1 of 1)

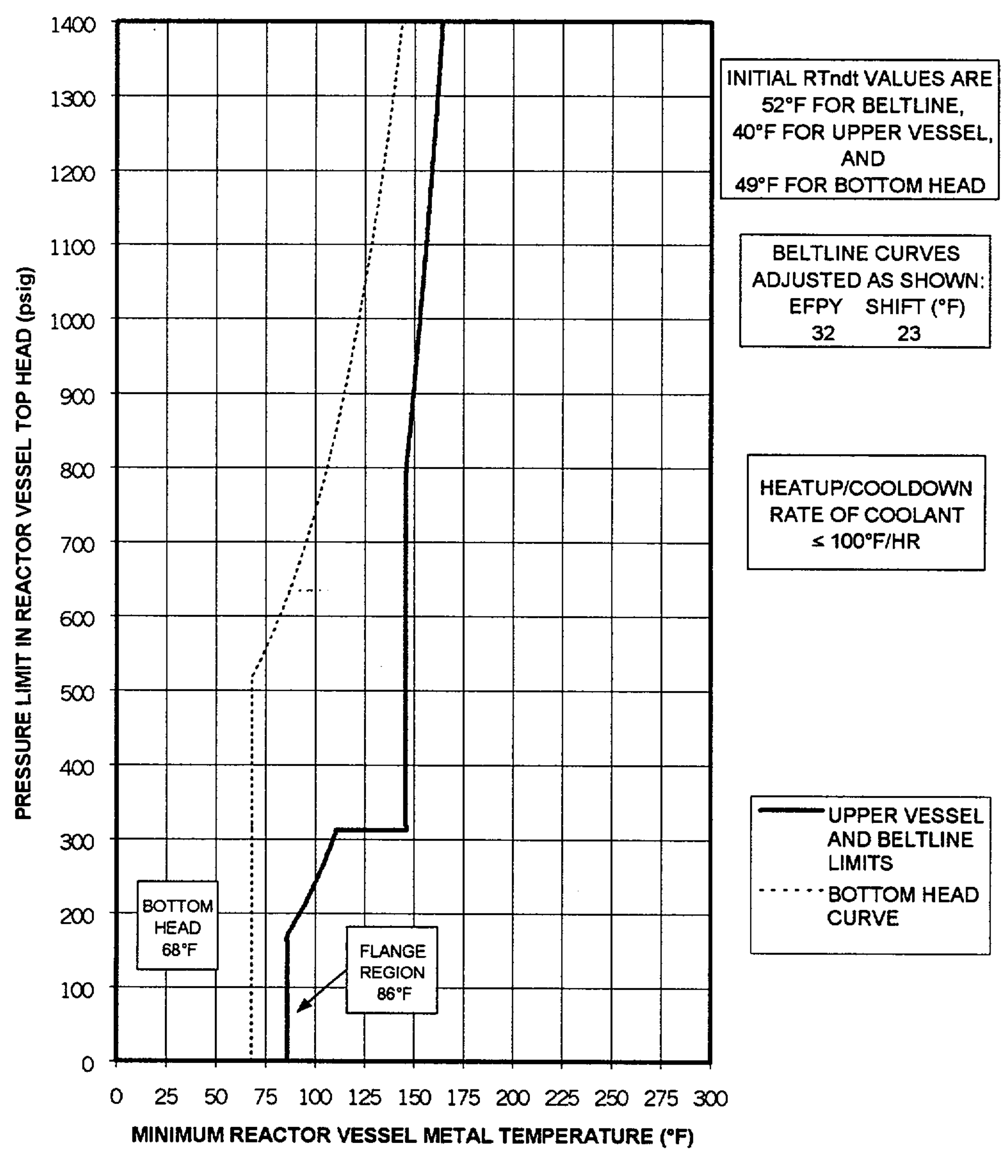
Unit 1
Minimum Reactor Vessel Metal Temperature vs.
Reactor Vessel Pressure (Valid up to 32 EFPY)



P-T Curves for Hydrostatic or Leak Testing

Figure 3.4.11-4 (Page 1 of 1)

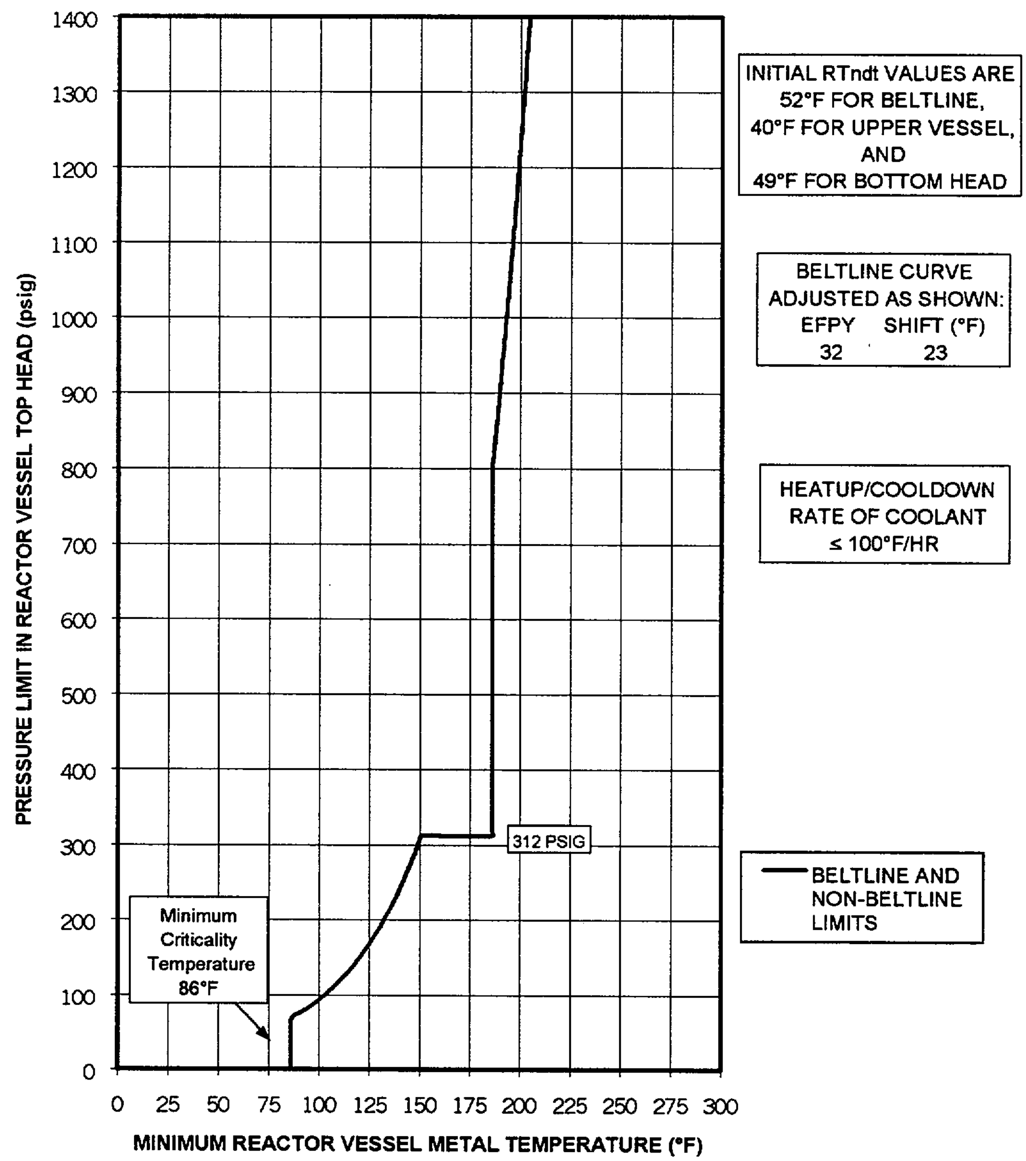
Unit 2
Minimum Reactor Vessel Metal Temperature vs.
Reactor Vessel Pressure (Valid up to 32 EFPY)



P-T Curves for Heatup by Non-Nuclear Means, Cooldown Following
A Nuclear Shutdown and Low Power Physics Testing

Figure 3.4.11-5 (Page 1 of 1)

Unit 2
Minimum Reactor Vessel Metal Temperature vs.
Reactor Vessel Pressure (Valid up to 32 EFPY)



P-T Curves for Operation with a Core Critical
other than Low Power Physics Testing

Figure 3.4.11-6 (Page 1 of 1)

Unit 2
Minimum Reactor Vessel Metal Temperature vs.
Reactor Vessel Pressure (Valid up to 32 EFPY)

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.12 Reactor Steam Dome Pressure

LCO 3.4.12 The reactor steam dome pressure shall be $\leq$ 1020 psig.

APPLICABILITY: MODES 1 and 2.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Reactor steam dome pressure not within limit.	A.1 Restore reactor steam dome pressure to within limit.	15 minutes
B. Required Action and associated Completion Time not met.	B.1 Be in MODE 3.	12 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
SR 3.4.12.1 Verify reactor steam dome pressure is $\leq$ 1020 psig.	12 hours

B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.1 Recirculation Loops Operating

BASES

BACKGROUND

The Reactor Recirculation System is designed to provide a forced coolant flow through the core to remove heat from the fuel. The forced coolant flow removes heat at a faster rate from the fuel than would be possible with just natural circulation. The forced flow, therefore, allows operation at significantly higher power than would otherwise be possible. The recirculation system also controls reactivity over a wide span of reactor power by varying the recirculation flow rate to control the void content of the moderator. The Reactor 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. Each external loop contains a two speed motor driven recirculation pump, a flow control valve, associated piping, jet pumps, valves, and instrumentation. The recirculation loops are part of the reactor coolant pressure boundary and are located inside the drywell structure. The jet pumps are reactor vessel internals.

The recirculated coolant consists of saturated water from the steam separators and dryers that has been subcooled by incoming feedwater. This water passes down the annulus between the reactor vessel wall and the core shroud. A portion of the coolant flows from the vessel, through the two external recirculation loops, and becomes the driving flow for the jet pumps. Each of the two external recirculation loops discharges high pressure flow into an external manifold, from which individual recirculation inlet lines are routed to the jet pump risers within the reactor vessel. The remaining portion of the coolant mixture in the annulus becomes the suction flow for the jet pumps. This flow enters the jet pump at suction inlets and is accelerated by the driving flow. The drive flow and suction flow are mixed in the jet pump throat section and result in partial pressure recovery. The total flow then passes through the jet pump diffuser section into the area below the core (lower plenum), gaining sufficient head in the process to drive the required flow upward through the core.

(continued)

BASES

BACKGROUND (continued)

The subcooled water enters the bottom of the fuel channels and contacts the fuel cladding, where heat is transferred to the coolant. As it rises, the coolant begins to boil, creating steam voids within the fuel channel that continue until the coolant exits the core. Because of reduced moderation, the steam voiding introduces negative reactivity that must be compensated for to maintain or to increase reactor power. The recirculation flow control allows operators to increase recirculation flow and sweep some of the voids from the fuel channel, overcoming the negative reactivity void effect. Thus, the reason for having variable recirculation flow is to compensate for reactivity effects of boiling over a wide range of power generation (i.e., approximately 65 to 100% RTP) without having to move control rods and disturb desirable flux patterns. In addition, the combination of core flow and THERMAL POWER is normally maintained such that core thermal-hydraulic oscillations do not occur. These oscillations can occur during two recirculation loop operation, single recirculation loop, and no recirculation loop operation. Plant procedures include requirements of this LCO as well as other vendor and NRC recommended requirements and actions to minimize the potential of core thermal-hydraulic oscillations.

Each recirculation loop is manually started from the control room. The recirculation flow control valves provide regulation of individual recirculation loop drive flows. The flow in each loop can be manually or automatically controlled.

APPLICABLE SAFETY ANALYSES

The operation of the Reactor Recirculation System is an initial condition assumed in the design basis loss of coolant accident (LOCA) (Ref. 1). 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 initial core flow decrease is rapid because the recirculation pump in the broken loop ceases to pump reactor coolant to the vessel almost immediately. The pump in the intact loop coasts down relatively slowly. This pump coastdown governs the core flow response for the next several seconds until the jet pump suction is uncovered (Ref. 2). The analyses assume that both loops are operating at the same flow prior to the accident. However, the LOCA

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

analysis was reviewed for the case with a flow mismatch between the two loops, with the pipe break assumed to be in the loop with the higher flow. While the flow coastdown and core response are potentially more severe in this assumed case (since the intact loop starts at a lower flow rate and the core response is the same as if both loops were operating at a lower flow rate), a small mismatch has been determined to be acceptable based on engineering judgement.

The recirculation system is also assumed to have sufficient flow coastdown characteristics to maintain fuel thermal margins during abnormal operational transients (Ref. 2), which are analyzed in Chapter 15 of the UFSAR.

A plant specific LOCA 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 Emergency Core Cooling System response will provide adequate core cooling, provided the APLHGR requirements are modified accordingly (Ref. 3).

The transient analyses in Chapter 15 of the UFSAR have also been performed for single recirculation loop operation (Ref. 3) and demonstrate sufficient flow coastdown characteristics to maintain fuel thermal margins during the abnormal operational transients analyzed provided the MCPR requirements are modified. During single recirculation loop operation, modification to the Reactor Protection System average power range monitor (APRM) and the Rod Block Monitor (RBM) Allowable Values is also required to account for the different relationships between recirculation drive flow and reactor core flow. The APLHGR and MCPR limits for single loop operation are specified in the COLR. The APRM Flow Biased Simulated Thermal Power-Upscale Allowable Value is in LCO 3.3.1.1, "Reactor Protection System (RPS) Instrumentation." The Rod Block Monitor-Upscale Allowable Value is specified in the COLR. Safety analyses performed in References 1, 2, and 3 implicitly assume core conditions are stable. However, during operation at the high power/low flow region of the operating domain, a small probability of limit cycle neutron flux oscillations exists depending on combinations of operating conditions (e.g., power shape, bundle power, and bundle flow).

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

General Electric (GE) Service Information Letter (SIL) No. 380 (Ref. 4) addressed boiling instability and made several recommendations. In this SIL, the power/flow operating map was divided into several regions of varying concern. It also discussed the objectives and philosophy of "detect and suppress."

NRC Generic Letter 86-02 (Ref. 5) discussed both the GE and Siemens stability methodology and stated that due to uncertainties, 10 CFR 50, Appendix A, General Design Criteria (GDC) 10 and 12 could not be met using available analytical procedures on a BWR. The Generic Letter discussed SIL 380 and stated that GDC 10 and 12 could be met by imposing SIL 380 recommendations in operating regions of potential instability. The NRC concluded that regions of potential instability constituted decay ratios of 0.8 and greater by the GE methodology and 0.75 by the Siemens methodology. Figure 3.4.1-1 was generated as an interim solution to provide an increased margin of safety until the investigation is completed (Ref. 6)

Recirculation loops operating satisfies Criterion 2 of 10 CFR 50.36(c)(2)(ii).

LCO

Two recirculation loops are normally required to be in operation with their flows matched within the limits specified in SR 3.4.1.1 to ensure that during a LOCA caused by a break of the piping of one recirculation loop the assumptions of the LOCA analysis are satisfied. With the limits specified in SR 3.4.1.1 not met, the recirculation loop with the lower flow must be considered not in operation. With only one recirculation loop in operation, modifications to the required APLHGR limits (LCO 3.2.1, "AVERAGE PLANAR LINEAR HEAT GENERATION RATE (APLHGR)"), MCPR limits (LCO 3.2.2, "MINIMUM CRITICAL POWER RATIO (MCPR)"), APRM Flow Biased Simulated Thermal Power — Upscale Allowable Value (LCO 3.3.1.1), and the Rod Block Monitor — Upscale Allowable Value (LCO 3.3.2.1) must be applied to allow continued operation consistent with the assumptions of Reference 3. In addition, during two-loop and single-loop

(continued)

BASES

LCO (continued)

operation, the combination of core flow and THERMAL POWER must be in Region III of Figure 3.4.1-1 to ensure core thermal-hydraulic oscillations do not occur.

APPLICABILITY

In MODES 1 and 2, requirements for operation of the Reactor Recirculation System are necessary since there is considerable energy in the reactor core and the limiting design basis transients and accidents are assumed to occur.

In MODES 3, 4, and 5, the consequences of an accident are reduced and the coastdown characteristics of the recirculation loops are not important.

ACTIONS

A.1, A.2, and A.3

With one or two recirculation loops in operation in Region II of Figure 3.4.1-1, the plant is operating in a region where the potential for thermal-hydraulic oscillations exists. To ensure oscillations are not occurring, APRM and LPRM neutron flux noise levels must be verified to be less than or equal to the larger of either 3 times the baseline noise levels or 10% peak-to-peak (Required Action A.1 and A.2) when Region II is entered. For the LPRM neutron flux noise verification, detector levels A and C of one LPRM string per core octant plus detector levels A and C of one LPRM string in the center region of the core should be monitored. Prompt action to monitor APRM and LPRM neutron flux noise levels should be taken to ensure oscillations are not occurring.

The 45 minute Completion Time of Required Actions A.1 and A.2 provides a reasonable time to stabilize operation in Region II and verify the neutron flux noise levels are within limits. A verification of the APRM and LPRM neutron flux noise levels once per 12 hours following the initial verification provides frequent periodic information of neutron flux noise levels to verify stable steady state operation. Also, a verification of neutron flux noise levels after any THERMAL POWER increase of ≥ 5% RTP while in Region II provides indication of operational stability following a potential for change of the thermal-hydraulic properties of the system.

(continued)

BASES

ACTIONS

A.1, A.2, and A.3 (continued)

In addition, a verification that one or both recirculation loops are not operating within Region I of Figure 3.4.1-1 (Required Action A.3) is required to be performed once per 12 hours. The Completion Time of once per 12 hours is reasonable based on operating experience and the operator's knowledge of reactor status, including changes in reactor power and core flow.

B.1

If evidence of approaching reactor instability occurs (i.e., APRM or LPRM neutron flux noise levels exceed the associated limit of Required Actions A.1 or A.2, as applicable) while operating in Region II of Figure 3.4.1-1, prompt action should be taken to restore the APRM or LPRM neutron flux noise levels to within the associated limit or exit Region II of Figure 3.4.1-1. This may be accomplished by either increasing core flow by recirculation loop flow control valve manipulation or reduction of THERMAL POWER by control rod insertion. The 2 hour Completion Time is reasonable to restore plant parameters in an orderly manner and without challenging plant systems.

C.1

With one or both recirculation loops in operation in Region I of Figure 3.4.1-1, the plant is operating in a region where the potential for thermal-hydraulic oscillations is increased and sufficient margin may not be available for operator response to suppress potential thermal-hydraulic oscillations. As a result, prompt action should be taken to exit Region I of Figure 3.4.1-1. This may be accomplished by either increasing core flow by recirculation loop flow control valve manipulation or reduction of THERMAL POWER by control rod insertion. The 2 hour Completion Time is reasonable to restore plant parameters in an orderly manner and without challenging plant systems.

(continued)

BASES

ACTIONS (continued)

D.1, D.2, and D.3

With no recirculation loops in service, the probability of thermal-hydraulic oscillations is greatly increased. Therefore, prompt action should be taken to ensure oscillations are not occurring by verifying APRM and LPRM neutron flux noise levels are $\leq$ 10% peak-to-peak. If neutron flux noise levels are discovered to be > 10% peak-to-peak at anytime while in this Condition, Condition E must be immediately entered.

Also, prompt action should be taken to reduce THERMAL POWER low enough to avoid the region of potential instability in natural circulation (i.e., reduce THERMAL POWER below 36% RTP). The 2 hour Completion Time provides are reasonable time to restore operation to Region III of Figure 3.4.1-1.

In addition, with no recirculation loops in operation, plant operation is not allowed to continue in MODE 1 or 2. Therefore, the unit is required to be brought to a MODE in which the LCO does not apply. The allowed Completion Time of 12 hours is reasonable, based on operating experience, to reach MODE 3 in an orderly manner and without challenging plant systems.

E.1

In the event no recirculation loops are in operation and evidence is indicated of approaching reactor instability (i.e., APRM or LPRM neutron flux noise levels exceed the associated limit) or APRM or LPRM neutron flux noise levels cannot be restored within 2 hours while in Region II of Figure 3.4.1-1, action must be immediately initiated to eliminate the potential for a thermal-hydraulic instability event. As such, the reactor mode switch must be immediately placed in the shutdown position.

(continued)

BASES

ACTIONS (continued)

F.1 and G.1

With both recirculation loops operating but the flows not matched, the flows must be matched within 2 hours. If matched flows are not restored, the recirculation loop with lower flow must be declared "not in operation," as required by Required Action F.1. This Required Action does not require tripping the recirculation pump in the lowest flow loop when the mismatch between total jet pump flows of the two loops is greater than the required limits. However, in cases where large flow mismatches occur, low flow or reverse flow can occur in the low flow loop jet pumps, causing vibration of the jet pumps. If zero or reverse flow is detected, the condition should be alleviated by changing flow control valve position to re-establish forward flow or by tripping the pump.

With the requirements of the LCO not met for reasons other than Conditions A, C, D, and F (e.g., one loop is "not in operation"), compliance with the LCO must be restored within 12 hours. A recirculation loop is considered not in operation when the pump in that loop is idle or when the mismatch between total jet pump flows of the two loops is greater than required limits for greater than 2 hours (i.e., Required Action F.1 has been taken). Should a LOCA occur with one recirculation loop not in operation, the core flow coastdown and resultant core response may not be bounded by the LOCA analyses. Therefore, only a limited time is allowed to restore the inoperable loop to operating status.

Alternatively, if the single loop requirements of the LCO are applied to the APLHGR and MCPR operating limits and RPS and RBM Allowable Values, operation with only one recirculation loop would satisfy the requirements of the LCO and the initial conditions of the accident sequence.

The 2 hour and 12 hour Completion Times are based on the low probability of an accident occurring during this time period, on a reasonable time to complete the Required Action, and on frequent core monitoring by operators allowing abrupt changes in core flow conditions to be quickly detected.

(continued)

BASES

ACTIONS (continued)	H.1 If the Required Action and associated Completion Time of Condition G is not met, the unit is required to be brought to a MODE in which the LCO does not apply. To achieve this status, the plant must be brought to MODE 3 within 12 hours. In this condition, the recirculation loops are not required to be operating because of the reduced severity of DBAs and minimal dependence on the recirculation loop coastdown characteristics. The allowed Completion Time of 12 hours is reasonable, based on operating experience, to reach MODE 3 from full power conditions in an orderly manner and without challenging plant systems.
SURVEILLANCE REQUIREMENTS	SR 3.4.1.1 This SR ensures the recirculation loop flows are within the allowable limits for mismatch. At low core flow (i.e., < 70% of rated core flow), the APLHGR and MCPR requirements provide larger margins to the fuel cladding integrity Safety Limit such that the potential adverse effect of early boiling transition during a LOCA is reduced. A larger flow mismatch can therefore be allowed when core flow is < 70% of rated core flow. The recirculation loop jet pump flow, as used in this Surveillance, is the summation of the flows from all of the jet pumps associated with a single recirculation loop. The mismatch is measured in terms of percent of rated core flow. If the flow mismatch exceeds the specified limits, the loop with the lower flow is considered not in operation. This SR is not required when both loops are not in operation since the mismatch limits are meaningless during single loop or natural circulation operation. The Surveillance must be performed within 24 hours after both loops are in operation. The 24 hour Frequency is consistent with the Frequency for jet pump OPERABILITY verification and has been shown by operating experience to be adequate to detect off normal jet pump loop flows in a timely manner.

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BASES

SURVEILLANCE REQUIREMENTS (continued)	SR 3.4.1.2 The SR ensures the combination of core flow and THERMAL POWER are within the appropriate limits to prevent inadvertent entry into a region of potential thermal-hydraulic instability. At low recirculation loop flow and high reactor power, the reactor exhibits increased susceptibility to thermal-hydraulic instability. Figure 3.4.1-1 is based on guidance provided in References 4 and 5. The 24 hour Frequency is based on operating experience and the operator's knowledge of the reactor status, including significant changes in THERMAL POWER and core flow.

REFERENCES

1. UFSAR, Sections 6.3 and 15.6.5.

2. UFSAR, Appendix G.3.1.2.

3. UFSAR, Section 6.B.

4. GE Service Information Letter (SIL) No. 380, "BWR Core Thermal Hydraulic Stability," Revision 1, February 10, 1984.

5. NRC Generic Letter 86-02, "Technical Resolution of Generic Issue B-19, Thermal Hydraulic Stability," January 22, 1986.

6. NRC Safety Evaluation supporting Amendment No. 60 to Facility Operating License No. 11 and Amendment No. 40 to Facility Operating License No. 18, Commonwealth Edison Company, LaSalle County Station, Units 1 and 2, dated September 7, 1988.

B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.2 Flow Control Valves (FCVs)

BASES

BACKGROUND

The Reactor Recirculation System is described in the Background section of the Bases for LCO 3.4.1, "Recirculation Loops Operating," which discusses the operating characteristics of the system and how this affects the design basis transient and accident analyses. The FCVs are part of the Reactor Recirculation System.

The Recirculation Flow Control System consists of the electronic and hydraulic components necessary for the positioning of the two hydraulically actuated FCVs. The recirculation loop flow rate can be rapidly changed within the expected flow range, in response to rapid changes in system demand. Limits on the system response are required to minimize the impact on core flow response during certain accidents and transients. Solid state control logic will generate an FCV "motion inhibit" signal in response to any one of several hydraulic power unit or analog control circuit failure signals. The "motion inhibit" signal causes hydraulic power unit shutdown and hydraulic isolation such that the FCVs fail "as is."

APPLICABLE SAFETY ANALYSES

The FCV stroke rate is limited to ≤ 11% per second in the opening and closing directions on a control signal failure of maximum demand. This stroke rate is an assumption of the analysis of the recirculation flow control failures on decreasing and increasing flow (Refs. 1 and 2). 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 initial core flow decrease is rapid because the recirculation pump in the broken loop ceases to pump almost immediately since it has lost suction. The pump in the intact loop coasts down relatively slowly. This pump coastdown governs the core flow response for the next several seconds (Ref. 3), because the FCV is assumed to fail "as is" due to a motion inhibit as a result of a high drywell pressure interlock. In addition, the closure of a recirculation FCV concurrent with a loss of coolant accident (LOCA) was analyzed during

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)	initial licensing and found to be acceptable for a maximum closure rate of 11% of stroke per second, since this event involves multiple failures. Flow control valves satisfy Criterion 2 of 10 CFR 50.36(c)(2)(ii).
LCO	An FCV in each operating recirculation loop must be OPERABLE to ensure that the assumptions of the design basis transient and accident analyses are satisfied.
APPLICABILITY	In MODES 1 and 2, the FCVs are required to be OPERABLE, since during these conditions there is considerable energy in the reactor core, and the limiting design basis transients and accidents are assumed to occur. In MODES 3, 4, and 5, the consequences of a transient or accident are reduced and OPERABILITY of the flow control valves is not important.
ACTIONS	A Note has been provided to modify the ACTIONS related to FCVs. Section 1.3, Completion Times, specifies once a Condition has been entered, subsequent divisions, subsystems, components or variables expressed in the Condition, discovered to be inoperable or not within limits, will not result in separate entry into the Condition. Section 1.3 also specifies Required Actions of the Condition continue to apply for each additional failure, with Completion Times based on initial entry into the Condition. However, the Required Actions for inoperable FCVs provide appropriate compensatory measures for separate inoperable FCVs. As such, a Note has been provided that allows separate Condition entry for each inoperable FCV. <u>A.1</u> With one or two required FCVs inoperable, the assumptions of the design basis transient and accident analyses may not be met and the inoperable FCV must be returned to OPERABLE status or hydraulically locked within 4 hours. Opening an FCV faster than the limit could result in a more severe flow runout transient. Closing an FCV faster than

(continued)

BASES

ACTIONS

A.1 (continued)

the limit could result in a more severe coolant flow decrease transient. Both conditions could result in violation of the Safety Limit MCPR. The FCVs are designed to lockup (high drywell pressure interlock) under LOCA conditions. When the FCVs "lock-up", the recirculation flow coastdown is adequate and the resulting calculated clad temperatures are acceptable. In addition, it has been calculated with the FCVs closing at the specified limit, the resulting calculated clad temperatures will also be acceptable. Closing an FCV faster than the limit assumed in the LOCA analysis (Ref. 3) could affect the recirculation flow coastdown, resulting in higher peak clad temperatures. Therefore, if an FCV is inoperable, deactivating the valve will essentially lock the valve in position, which will prohibit the FCV from adversely affecting the DBA and transient analyses. Continued operation is allowed in this Condition.

The 4 hour Completion Time is a reasonable time period to complete the Required Action, while limiting the time of operation with an inoperable FCV.

B.1

If the FCVs are not deactivated ("locked up") within the associated Completion Time, the unit must be brought to a MODE in which the LCO does not apply. To achieve this status, the unit must be brought to at least MODE 3 within 12 hours. This brings the unit to a condition where the flow coastdown characteristics of the recirculation loop are not important. The allowed Completion Time of 12 hours is reasonable, based on operating experience, to reach MODE 3 from full power conditions in an orderly manner and without challenging unit systems.

SURVEILLANCE REQUIREMENTS

SR 3.4.2.1

Hydraulic power unit pilot operated 4-way valves located between the servo valves and the common "open" and "close" lines are required to close in the event of a loss of hydraulic pressure. When closed, these valves inhibit FCV motion by blocking hydraulic pressure from the servo valve

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.4.2.1 (continued)

to the common open and close lines as well as to the alternate subloop. This Surveillance verifies FCV lockup on a loss of hydraulic pressure.

The 24 month Frequency is based on the need to perform this Surveillance under the conditions that apply during a plant outage and the potential for an unplanned transient if the Surveillance were performed with the reactor at power. Operating experience has shown these components usually pass the SR when performed at the 24 month Frequency. Therefore, the Frequency was concluded to be acceptable from a reliability standpoint.

SR 3.4.2.2

This SR ensures the overall average rate of FCV movement at all positions is maintained within the analyzed limits.

The 24 month Frequency is based on the need to perform this Surveillance under the conditions that apply during a plant outage and the potential for an unplanned transient if the Surveillance were performed with the reactor at power. Operating experience has shown these components usually pass the SR when performed at the 24 month Frequency. Therefore, the Frequency was concluded to be acceptable from a reliability standpoint.

REFERENCES

1. UFSAR, Section 15.3.2.
2. UFSAR, Section 15.4.5.
3. UFSAR, Appendix G.

B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.3 Jet Pumps

BASES

BACKGROUND

The Reactor Recirculation System is described in the Background section of the Bases for LCO 3.4.1, "Recirculation Loops Operating," which discusses the operating characteristics of the system and how these characteristics affect the Design Basis Accident (DBA) analyses.

The jet pumps are part of the Reactor Recirculation System and are designed to provide forced circulation through the core to remove heat from the fuel. The jet pumps are located in the annular region between the core shroud and the vessel inner wall. Because the jet pump suction elevation is at two thirds core height, the vessel can be reflooded and coolant level maintained at two thirds core height even with the complete break of the recirculation loop pipe that is located below the jet pump suction elevation.

Each reactor coolant recirculation loop contains 10 jet pumps. Recirculated coolant passes down the annulus between the reactor vessel wall and the core shroud. A portion of the coolant flows from the vessel, through the two external recirculation loops, and becomes the driving flow for the jet pumps. Each of the two external recirculation loops discharges high pressure flow into an external manifold from which individual recirculation inlet lines are routed to the jet pump risers within the reactor vessel. The remaining portion of the coolant mixture in the annulus becomes the suction flow for the jet pumps. This flow enters the jet pump at suction inlets and is accelerated by the drive flow. The drive flow and suction flow are mixed in the jet pump throat section. The total flow then passes through the jet pump diffuser section into the area below the core (lower plenum), gaining sufficient head in the process to drive the required flow upward through the core.

APPLICABLE SAFETY ANALYSES

Jet pump OPERABILITY is an explicit assumption in the design basis loss of coolant accident (LOCA) analysis evaluated in Reference 1.

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

The capability of reflooding the core to two-thirds core height is dependent upon the structural integrity of the jet pumps. If the structural system, including the beam holding a jet pump in place, fails, jet pump displacement and performance degradation could occur, resulting in an increased 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.

Jet pumps satisfy Criterion 3 of 10 CFR 50.36(c)(2)(ii).

LCO

The structural failure of any of the jet pumps could cause significant degradation in the ability of the jet pumps to allow reflooding to two thirds core height during a LOCA. OPERABILITY of all jet pumps is required to ensure that operation of the Reactor Recirculation System will be consistent with the assumptions used in the licensing basis analysis (Ref. 1).

APPLICABILITY

In MODES 1 and 2, the jet pumps are required to be OPERABLE since there is a large amount of energy in the reactor core and since the limiting DBAs are assumed to occur in these MODES. This is consistent with the requirements for operation of the Reactor Recirculation System (LCO 3.4.1).

In MODES 3, 4, and 5, the Reactor Recirculation System is not required to be in operation, and when not in operation sufficient flow is not available to evaluate jet pump OPERABILITY.

ACTIONS

A.1

An inoperable jet pump can increase the blowdown area and reduce the capability to reflood during a design basis LOCA. If one or more of the jet pumps are inoperable, the plant must be brought to a MODE in which the LCO does not apply. To achieve this status, the plant must be brought to MODE 3 within 12 hours. The allowed Completion Time of 12 hours is reasonable, based on operating experience, to reach MODE 3 from full power conditions in an orderly manner and without challenging plant systems.

(continued)

BASES (continued)

SURVEILLANCE REQUIREMENTS

SR 3.4.3.1

This SR is designed to detect significant degradation in jet pump performance that precedes jet pump failure (Ref. 2). This SR is required to be performed only when the loop has forced recirculation flow since surveillance checks and measurements can only be performed during jet pump operation. The jet pump failure of concern is a complete mixer displacement due to jet pump beam failure. Jet pump plugging is also of concern since it adds flow resistance to the recirculation loop. Significant degradation is indicated if any two of the three specified criteria confirm unacceptable deviations from established patterns or relationships. The allowable deviations from the established patterns have been developed based on the variations experienced at plants during normal operation and with jet pump assembly failures (Refs. 2 and 3). Since refueling activities (fuel assembly replacement or shuffle, as well as any modifications 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 re-established 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 baselining new "established patterns", engineering judgement of the daily Surveillance results is used to detect significant abnormalities which could indicate a jet pump failure. In addition, during two recirculation loop operation, the jet pump SR should be performed with balanced recirculation loop drive flows (drive flow mismatch less than 5%) to ensure an accurate indication of jet pump performance.

The recirculation flow control valve (FCV) operating characteristics (loop flow characteristics versus FCV position) are determined by the flow resistance from the loop suction through the jet pump nozzles. A change in the relationship may indicate a flow restriction, loss in pump hydraulic performance, leak, or new flow path between the recirculation pump discharge and jet pump nozzle. For this criterion, the loop flow versus FCV position relationship must be verified. When both recirculation loops are operating, the established FCV position should include the

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.4.3.1 (continued)

loop flow characteristics for two recirculation loop operation. When only one recirculation loop is operating, the established FCV position should include the loop flow characteristics for single loop operation.

Total calculated core flow can be determined from either the established THERMAL POWER-core flow relationship or the core plate differential pressure-core flow relationship. Once this relationship has been established, increased or reduced indicated total core flow from the calculated total core flow may be an indication of failures in one or several jet pumps. When determining calculated total core flow in single recirculation loop operation using the core plate differential pressure-core flow relationship, the calculated total core flow value should be derived using the established core plate differential pressure - core flow relationship for two recirculation loop operation.

Individual jet pumps in a recirculation loop typically do not have the same flow. The unequal flow is due to the drive flow manifold, which does not distribute flow equally to all risers. The jet pump diffuser to lower plenum differential pressure pattern or relationship of one jet pump to the loop average is repeatable. An appreciable change in this relationship is an indication that increased (or reduced) resistance has occurred in one of the jet pumps.

The deviations from normal are considered indicative of a potential problem in the recirculation drive flow or jet pump system (Ref. 2). Normal flow ranges and established jet pump differential pressure patterns are established by plotting historical data as discussed in Reference 2.

The 24 hour Frequency has been shown by operating experience to be adequate to verify jet pump OPERABILITY and is consistent with the Frequency for recirculation loop OPERABILITY verification.

This SR is modified by two Notes. Note 1 allows this Surveillance not to be performed until 4 hours after the associated recirculation loop is in operation, since these

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BASES

SURVEILLANCE REQUIREMENTS	SR 3.4.3.1 (continued) checks can only be performed during jet pump operation. The 4 hours is an acceptable time to establish conditions appropriate for data collection and evaluation. Note 2 allows this SR not to be performed until 24 hours after THERMAL POWER exceeds 25% RTP. During low flow conditions, jet pump noise approaches the threshold response of the associated flow instrumentation and precludes the collection of repeatable and meaningful data. The 24 hours is an acceptable time to establish conditions appropriate to perform this SR.
REFERENCES	1. UFSAR, Section 6.3 and Appendices G.2.2.2 and G.3.2.2.3. 2. GE Service Information Letter No. 330. 3. NUREG/CR-3052, "Closeout of IE Bulletin 80-07: BWR Jet Pump Assembly Failure," November 1984.

B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.4 Safety/Relief Valves (S/RVs)

BASES

BACKGROUND

The American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code (Ref. 1) 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/relief valves (S/RVs) are selected such that peak pressure in the nuclear system will not exceed the ASME Code limits for the reactor coolant pressure boundary (RCPB).

The S/RVs are located on the main steam lines between the reactor vessel and the first isolation valve within the drywell. Each S/RV discharges steam through a discharge line to a point below the minimum water level in the suppression pool.

The S/RVs can actuate by either of two modes: the safety mode or the relief mode (however, for the purpose of this LCO, only the safety mode is required). In the safety mode (or spring mode of operation), the direct action of the steam pressure in the main steam lines will act against a spring loaded disk that will pop open when the valve inlet pressure exceeds the spring force. In the relief mode (or power actuated mode of operation), a pneumatic piston/cylinder and mechanical linkage assembly are used to open the valve by overcoming the spring force, even with the valve inlet pressure equal to 0 psig. The pneumatic operator is arranged so that its malfunction will not prevent the valve disk from lifting if steam inlet pressure reaches the spring lift set pressures. In the relief mode, valves may be opened manually or automatically at the selected preset pressure. Seven of the S/RVs that provide the safety and relief function are part of the Automatic Depressurization System specified in LCO 3.5.1, "ECCS—Operating." The instrumentation associated with the relief valve function for the ADS function is discussed in LCO 3.3.5.1, "Emergency Core Cooling Systems (ECCS) Instrumentation."

(continued)

BASES (continued)

APPLICABLE SAFETY ANALYSES

The overpressure protection system must accommodate the most severe pressure transient. Evaluations have determined that the most severe transient is the closure of all main steam isolation valves (MSIVs) followed by reactor scram on high neutron flux (i.e., failure of the direct scram associated with MSIV position) (Ref. 2). For the purpose of the analyses, 12 of the S/RVs are assumed to operate in the safety mode. The analysis results demonstrate that the design S/RV capacity is capable of maintaining reactor pressure below the ASME Code limit of 110% of vessel design pressure (110% x 1250 psig = 1375 psig). This LCO helps to ensure that the acceptance limit of 1375 psig is met during the design basis event.

From an overpressure standpoint, the design basis events are bounded by the MSIV closure with flux scram event described above. For other pressurization events, such as a turbine trip or generator load rejection with Main Turbine Bypass System failure, the S/RVs are assumed to function. The opening of the valves during the pressurization event mitigates the increase in reactor vessel pressure, which affects the MINIMUM CRITICAL POWER RATIO (MCPR) during these events. The number of S/RVs required to mitigate these events is bounded by the number required to be OPERABLE by the LCO.

S/RVs satisfy Criterion 3 of 10 CFR 50.36(c)(2)(ii).

LCO

The safety function of 17 S/RVs, for Unit 1, and 12 S/RVs, for Unit 2, is required to be OPERABLE. The requirements of this LCO are applicable only to the capability of the S/RVs to mechanically open to relieve excess pressure when the lift setpoint is exceeded (safety mode). In Reference 2, an evaluation was performed to establish the parametric relationship between the peak vessel pressure and the number of OPERABLE S/RVs. The results show that with a minimum of 17 S/RVs, for Unit 1, and 12 S/RVs, for Unit 2, in the safety mode OPERABLE, the ASME Code limit of 1375 psig is not exceeded.

(continued)

BASES

LCO (continued)

The S/RV safety setpoints are established to ensure the ASME Code limit on peak reactor pressure is satisfied. The ASME Code specifications require the lowest safety valve be set at or below vessel design pressure (1250 psig) and the highest safety valve be set so the total accumulated pressure does not exceed 110% of the design pressure for overpressurization conditions. The transient evaluations in Reference 3 involving the safety mode are based on these setpoints, but also include the additional uncertainties of ± 3% of the nominal setpoint to account for potential setpoint drift to provide an added degree of conservatism.

Operation with fewer valves OPERABLE than specified, or with setpoints outside the ASME limits, could result in a more severe reactor response to a transient than predicted, possibly resulting in the ASME Code limit on reactor pressure being exceeded.

The S/RVs are required to be OPERABLE to limit peak pressure in the main steam lines and maintain reactor pressure within acceptable limits during events that cause rapid pressurization, so that MCPR is not exceeded.

APPLICABILITY

In MODES 1, 2, and 3, the specified number of S/RVs must be OPERABLE since there may be considerable energy in the reactor core and the limiting design basis transients are assumed to occur. The S/RVs may be required to provide pressure relief to limit peak reactor pressure.

In MODE 4, decay heat is low enough for the RHR System to provide adequate cooling, and reactor pressure is low enough that the overpressure limit is unlikely to be approached by assumed operational transients or accidents. In MODE 5, the reactor vessel head is unbolted or removed and the reactor is at atmospheric pressure. The S/RV function is not needed during these conditions.

ACTIONS

A.1 and A.2

With less than the minimum number of required S/RVs OPERABLE, a transient may result in the violation of the ASME Code limit on reactor pressure. If one or more required S/RVs are inoperable, the plant must be brought to

(continued)

BASES

ACTIONS

A.1 and A.2 (continued)

a MODE in which the LCO does not apply. To achieve this status, the plant must be brought to at least MODE 3 within 12 hours and to MODE 4 within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.4.4.1

This Surveillance demonstrates that the required S/RVs will open at the pressures assumed in the safety analysis of Reference 2. The demonstration of the S/RV safety function lift settings must be performed during shutdown, since this is a bench test, and in accordance with the Inservice Testing Program. The lift setting pressure shall correspond to ambient conditions of the valves at nominal operating temperatures and pressures. The S/RV setpoint is ± 3% for OPERABILITY; however, the valves are reset to ± 1% during the Surveillance to allow for drift. A Note is provided to allow up to two of the required 17 S/RVs for Unit 1 and 12 S/RVs for Unit 2 to be physically replaced with S/RVs with lower setpoints. This provides operational flexibility which maintains the assumptions in the overpressure protection analysis.

The Frequency is specified in the Inservice Testing Program which requires the valves be subjected to a bench test during refueling outages. The Frequency is acceptable based on industry standards and operating history.

REFERENCES

1. ASME, Boiler and Pressure Vessel Code, Section III.
2. UFSAR, Section 5.2.2.1.3.
3. UFSAR, Chapter 15.

B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.5 RCS Operational LEAKAGE

BASES

BACKGROUND

The RCS includes systems and components that contain or transport the coolant to or from the reactor core. The pressure containing components of the RCS and the portions of connecting systems out to and including the isolation valves define the reactor coolant pressure boundary (RCPB). The joints of the RCPB components are welded or bolted.

During plant life, the joint and valve interfaces can produce varying amounts of reactor coolant LEAKAGE, through either normal operational wear or mechanical deterioration. Limits on RCS operational LEAKAGE are required to ensure appropriate action is taken before the integrity of the RCPB is impaired. This LCO specifies the types and limits of LEAKAGE. This protects the RCS pressure boundary described in 10 CFR 50.2, 10 CFR 50.55a(c), and GDC 55 of 10 CFR 50, Appendix A (Refs. 1, 2, and 3).

The safety significance of leaks from the RCPB varies widely depending on the source, rate, and duration. Therefore, detection of LEAKAGE in the drywell is necessary. Methods for quickly separating the identified LEAKAGE from the unidentified LEAKAGE are necessary to provide the operators quantitative information to permit them to take corrective action should a leak occur detrimental to the safety of the facility or the public.

A limited amount of leakage inside the drywell is expected from auxiliary systems that cannot be made 100% leaktight. Leakage from these systems should be detected and isolated from the drywell atmosphere, if possible, so as not to mask RCS operational LEAKAGE detection.

This LCO deals with protection of the RCPB from degradation and the core from inadequate cooling, in addition to preventing the accident analyses radiation release assumptions from being exceeded. The consequences of violating this LCO include the possibility of a loss of coolant accident.

(continued)

BASES (continued)

APPLICABLE SAFETY ANALYSES

The allowable RCS operational LEAKAGE limits are based on the predicted and experimentally observed behavior of pipe cracks. The normally expected background LEAKAGE due to equipment design and the detection capability of the instrumentation for determining system LEAKAGE were also considered. The evidence from experiments suggests, for LEAKAGE even greater than the specified unidentified LEAKAGE limits, the probability is small that the imperfection or crack associated with such LEAKAGE would grow rapidly.

The unidentified LEAKAGE flow limit allows time for corrective action before the RCPB could be significantly compromised. The 5 gpm limit is a small fraction of the calculated flow from a critical crack in the primary system piping. Crack behavior from experimental programs (Refs. 4 and 5) shows leak rates of hundreds of gallons per minute will precede crack instability (Ref. 6).

The low limit on increase in unidentified LEAKAGE assumes a failure mechanism of intergranular stress corrosion cracking (IGSCC) that produces tight cracks. This flow increase limit is capable of providing an early warning of such deterioration.

No applicable safety analysis assumes the total LEAKAGE limit. The total LEAKAGE limit considers RCS inventory makeup capability and drywell sump capacity.

RCS operational LEAKAGE satisfies Criterion 2 of 10 CFR 50.36(c)(2)(ii).

LCO

RCS operational LEAKAGE shall be limited to:

a. <u>Pressure Boundary LEAKAGE</u>

No pressure boundary LEAKAGE is allowed, being indicative of material degradation. LEAKAGE of this type is unacceptable as the leak itself could cause further deterioration, resulting in higher LEAKAGE. Violation of this LCO could result in continued degradation of the RCPB. LEAKAGE past seals and gaskets is not pressure boundary LEAKAGE.

(continued)

BASES

LCO (continued)

b. Unidentified LEAKAGE

Five gpm of unidentified LEAKAGE is allowed as a reasonable minimum detectable amount that the drywell atmosphere monitoring, drywell sump flow monitoring, and drywell air cooler condensate flow rate monitoring equipment can detect within a reasonable time period. Violation of this LCO could result in continued degradation of the RCPB.

c. Total LEAKAGE

The total LEAKAGE limit is based on a reasonable minimum detectable amount. The limit also accounts for LEAKAGE from known sources (identified LEAKAGE). Violation of this LCO indicates an unexpected amount of LEAKAGE and, therefore, could indicate new or additional degradation in an RCPB component or system.

d. Unidentified LEAKAGE Increase

An unidentified LEAKAGE increase of > 2 gpm within the previous 24 hour period indicates a potential flaw in the RCPB and must be quickly evaluated to determine the source and extent of the LEAKAGE. The increase is measured relative to the steady state value; temporary changes in LEAKAGE rate as a result of transient conditions (e.g., startup) are not considered. As such, the 2 gpm increase limit is only applicable in MODE 1 when operating pressures and temperatures are established. Violation of this LCO could result in continued degradation of the RCPB.

APPLICABILITY

In MODES 1, 2, and 3, the RCS operational LEAKAGE LCO applies because the potential for RCPB LEAKAGE is greatest when the reactor is pressurized.

In MODES 4 and 5, RCS operational LEAKAGE limits are not required since the reactor is not pressurized and stresses in the RCPB materials and potential for LEAKAGE are reduced.

(continued)

BASES (continued)

ACTIONS

A.1

With RCS unidentified or total LEAKAGE greater than the limits, actions must be taken to reduce the leak. Because the LEAKAGE limits are conservatively below the LEAKAGE that would constitute a critical crack size, 4 hours is allowed to reduce the LEAKAGE rates before the reactor must be shut down. If an unidentified LEAKAGE has been identified and quantified, it may be reclassified and considered as identified LEAKAGE. However, the total LEAKAGE limit would remain unchanged.

B.1 and B.2

An unidentified LEAKAGE increase of > 2 gpm within a 24 hour period is an indication of a potential flaw in the RCPB and must be quickly evaluated. Although the increase does not necessarily violate the absolute unidentified LEAKAGE limit, certain susceptible components must be determined not to be the source of the LEAKAGE increase within the required Completion Time. For an unidentified LEAKAGE increase greater than required limits, an alternative to reducing LEAKAGE increase to within limits (i.e., reducing the leakage rate such that the current rate is less than the "2 gpm increase in the previous 24 hours" limit; either by isolating the source or other possible methods) is to identify the source of the unidentified leakage increase is not material susceptible to IGSCC.

The 4 hour Completion Time is needed to properly reduce the LEAKAGE increase or identify the source before the reactor must be shut down.

C.1 and C.2

If any Required Action and associated Completion Time of Condition A or B is not met or if pressure boundary LEAKAGE exists, the plant must be brought to a MODE in which the LCO does not apply. To achieve this status, the plant must be brought to MODE 3 within 12 hours and to MODE 4 within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

(continued)

BASES (continued)

SURVEILLANCE REQUIREMENTS

SR 3.4.5.1

The RCS LEAKAGE is monitored by a variety of instruments designed to provide alarms when LEAKAGE is indicated and to quantify the various types of LEAKAGE. Leakage detection instrumentation is discussed in more detail in the Bases for LCO 3.4.7, "RCS Leakage Detection Instrumentation." Sump level and flow rate are typically monitored to determine actual LEAKAGE rates. However, any method may be used to quantify LEAKAGE provided the method has suitable sensitivity to satisfy the requirements of LCO 3.4.5. In conjunction with alarms and other administrative controls, a 12 hour Frequency for this Surveillance is appropriate for identifying changes in LEAKAGE and for tracking required trends (Ref. 7).

REFERENCES

1. 10 CFR 50.2.
2. 10 CFR 50.55a(c).
3. 10 CFR 50, Appendix A, GDC 55.
4. GEAP-5620, "Failure Behavior in ASTM A106 B Pipes Containing Axial Through-Wall Flaws," April 1968.
5. NUREG-75/067, "Investigation and Evaluation of Cracking in Austenitic Stainless Steel Piping of Boiling Water Reactor Plants," October 1975.
6. UFSAR, Section 5.2.5.5.2.
7. Generic Letter 88-01, Supplement 1, February 1992.

B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.6 RCS Pressure Isolation Valve (PIV) Leakage

BASES

BACKGROUND

The function of RCS PIVs is to separate the high pressure RCS from an attached low pressure system. This protects the RCS pressure boundary described in 10 CFR 50.2, 10 CFR 50.55a(c), and GDC 55 of 10 CFR 50, Appendix A (Refs. 1, 2, and 3). PIVs are designed to meet the requirements of Reference 4. During their lives, these valves can produce varying amounts of reactor coolant leakage through either normal operational wear or mechanical deterioration.

The RCS PIV LCO allows RCS high pressure operation when leakage through these valves exists in amounts that do not compromise safety. The PIV leakage limit applies to each individual valve. Leakage through these valves is not included in any allowable LEAKAGE specified in LCO 3.4.5, "RCS Operational LEAKAGE."

Although this Specification provides a limit on allowable PIV leakage rate, its main purpose is to prevent overpressure failure of the low pressure portions of connecting systems. The leakage limit is an indication that the PIVs between the RCS and the connecting systems are degraded or degrading. PIV leakage could lead to overpressure of the low pressure piping or components. Failure consequences could be a loss of coolant accident (LOCA) outside of containment, an unanalyzed accident which could degrade the ability for low pressure injection.

A study (Ref. 5) evaluated various PIV configurations to determine the probability of intersystem LOCAs. This study concluded that periodic leakage testing of the PIVs can substantially reduce intersystem LOCA probability.

PIVs are provided to isolate the RCS from the following connected systems:

a. Residual Heat Removal (RHR) System;

b. Low Pressure Core Spray System;

(continued)

BASES

BACKGROUND (continued)	c. High Pressure Core Spray System; and d. Reactor Core Isolation Cooling System. The PIVs are listed in the Technical Requirements Manual (Ref. 6).
APPLICABLE SAFETY ANALYSES	Reference 5 evaluated various PIV configurations, leakage testing of the valves, and operational changes to determine the effect on the probability of intersystem LOCAs. This study concluded that periodic leakage testing of the PIVs can substantially reduce the probability of an intersystem LOCA. PIV leakage is not considered in any Design Basis Accident analyses. This Specification provides for monitoring the condition of the reactor coolant pressure boundary (RCPB) to detect PIV degradation that has the potential to cause a LOCA outside of containment. RCS PIV leakage satisfies Criterion 2 of 10 CFR 50.36(c)(2)(ii).
LCO	RCS PIV leakage is leakage into closed systems connected to the RCS. Isolation valve leakage is usually on the order of drops per minute. Leakage that increases significantly suggests that something is operationally wrong and corrective action must be taken. Violation of this LCO could result in continued degradation of a PIV, which could lead to overpressurization of a low pressure system and the loss of the integrity of a fission product barrier. The LCO PIV leakage limit is 0.5 gpm per nominal inch of valve size with a maximum limit of 5 gpm (Ref. 4). Reference 7 permits leakage testing at a lower pressure differential than between the specified maximum RCS pressure and the normal pressure of the connected system during RCS operation (the maximum pressure differential). The observed rate may be adjusted to the maximum pressure differential by assuming leakage is directly proportional to the pressure differential to the one-half power.

(continued)

BASES (continued)

APPLICABILITY

In MODES 1, 2, and 3, this LCO applies because the PIV leakage potential is greatest when the RCS is pressurized. In MODE 3, valves in the RHR flowpath are not required to meet the requirements of this LCO when in, or during transition to or from, the RHR shutdown cooling mode of operation.

In MODES 4 and 5, leakage limits are not provided because the lower reactor coolant pressure results in a reduced potential for leakage and for a LOCA outside the containment. Accordingly, the potential for the consequences of reactor coolant leakage is far lower during these MODES.

ACTIONS

The ACTIONS are modified by two Notes. Note 1 has been provided to modify the ACTIONS related to RCS PIV flow paths. Section 1.3, Completion Times, specifies once a Condition has been entered, subsequent divisions, subsystems, components or variables expressed in the Condition, discovered to be inoperable or not within limits, will not result in separate entry into the Condition. Section 1.3 also specifies Required Actions of the Condition continue to apply for each additional failure, with Completion Times based on initial entry into the Condition. However, the Required Actions for the Condition of RCS PIV leakage limits exceeded provide appropriate compensatory measures for separate, affected RCS PIV flow paths. As such, a Note has been provided that allows separate Condition entry for each affected RCS PIV flow path. Note 2 requires an evaluation of affected systems if a PIV is inoperable. The leakage may have affected system OPERABILITY, or isolation of a leaking flow path with an alternate valve may have degraded the ability of the interconnected system to perform its safety function. As a result, the applicable Conditions and Required Actions for systems made inoperable by PIVs must be entered. This ensures appropriate remedial actions are taken, if necessary, for the affected systems.

A.1 and A.2

If leakage from one or more RCS PIVs is not within limit, the flow path must be isolated by at least one closed

(continued)

BASES

ACTIONS

A.1 and A.2 (continued)

manual, de-activated automatic, or check valve within 4 hours. Required Action A.1 and Required Action A.2 are modified by a Note stating that the valves used for isolation must meet the same leakage requirements as the PIVs and must be on the RCPB or the high pressure portion of the system.

Four hours provides time to reduce leakage in excess of the allowable limit and to isolate the flow path if leakage cannot be reduced while corrective actions to reseat the leaking PIVs are taken. The 4 hours allows time for these actions and restricts the time of operation with leaking valves.

Required Action A.2 specifies that the double isolation barrier of two valves be restored by closing another valve qualified for isolation or restoring one leaking PIV. The 72 hour Completion Time after exceeding the limit considers the time required to complete the Required Action, the low probability of a second valve failing during this time period, and the low probability of a pressure boundary rupture of the low pressure ECCS piping when overpressurized to reactor pressure (Ref. 7).

B.1 and B.2

If leakage cannot be reduced or the system isolated, the plant must be brought to a MODE in which the LCO does not apply. To achieve this status, the plant must be brought to MODE 3 within 12 hours and to MODE 4 within 36 hours. This action may reduce the leakage and also reduces the potential for a LOCA outside the containment. The Completion Times are reasonable, based on operating experience, to achieve the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.4.6.1

Performance of leakage testing on each RCS PIV is required to verify that leakage is below the specified limit and to identify each leaking valve. The leakage limit of 0.5 gpm

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.4.6.1 (continued)

per inch of nominal valve diameter up to 5 gpm maximum applies to each valve. Leakage testing requires a stable pressure condition. As stated in the LCO section of the Bases, the test pressure may be at a lower pressure than the maximum pressure differential (at the maximum pressure of 1050 psig) provided the observed leakage rate is adjusted in accordance with Reference 4. For the two PIVs tested in series, the leakage requirement applies to each valve individually and not to the combined leakage across both valves (i.e., the leakage acceptance criteria is the criteria for one valve to account for the condition where all of the leakage is through one valve). If the PIVs are not individually leakage tested, one valve may have failed completely and not be detected if the other valve in series meets the leakage requirement. In this situation, the protection provided by redundant valves would be lost.

The Frequency required by the Inservice Testing Program is within the ASME Code, Section XI, Frequency requirement and is based on the need to perform this Surveillance under the conditions that apply during an outage and the potential for an unplanned transient if the Surveillance were performed with the reactor at power.

This SR is modified by a Note that states the leakage Surveillance is only required to be performed in MODES 1 and 2. Entry into MODE 3 is permitted for leakage testing at high differential pressures with stable conditions not possible in the lower MODES.

REFERENCES

1. 10 CFR 50.2.

2. 10 CFR 50.55a(c).

3. 10 CFR 50, Appendix A, GDC 55.

4. ASME, Boiler and Pressure Vessel Code, Section XI.

5. NUREG-0677, "The Probability of Intersystem LOCA: Impact Due to Leak Testing and Operational Changes," May 1980.

(continued)

BASES

REFERENCES (continued)

6. Technical Requirements Manual.

7. NEDC-31339, "BWR Owners Group Assessment of Emergency Core Cooling System Pressurization in Boiling Water Reactors," November 1986.

B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.7 RCS Leakage Detection Instrumentation

BASES

BACKGROUND

GDC 30 of 10 CFR 50, Appendix A (Ref. 1), requires means for detecting and, to the extent practical, identifying the location of the source of RCS LEAKAGE. Regulatory Guide 1.45 (Ref. 2) describes acceptable methods for selecting leakage detection systems.

Limits on LEAKAGE from the reactor coolant pressure boundary (RCPB) are required so that appropriate action can be taken before the integrity of the RCPB is impaired (Ref. 2). Leakage detection systems for the RCS are provided to alert the operators when leakage rates above normal background levels are detected and also to supply quantitative measurement of rates. The Bases for LCO 3.4.5, "RCS Operational LEAKAGE," discuss the limits on RCS LEAKAGE rates.

Systems for separating the LEAKAGE of an identified source from an unidentified source are necessary to provide prompt and quantitative information to the operators to permit them to take immediate corrective action.

LEAKAGE from the RCPB inside the drywell is detected by at least one of three independently monitored variables, such as drywell air cooler condensate flow rate, sump flow rate, and drywell gaseous and particulate radioactivity levels. The primary means of quantifying LEAKAGE in the drywell is the drywell floor drain sump flow monitoring system.

The drywell floor drain sump flow monitoring system monitors the LEAKAGE collected in the floor drain sump. This unidentified LEAKAGE consists of LEAKAGE from control rod drives, valve flanges or packings, floor drains, the closed cooling water subsystems, and drywell air cooling unit condensate drains, and any LEAKAGE not collected in the drywell equipment drain sump. The drywell floor drain sump has a weir level transmitter that supplies floor drain sump fill-up rate flow indication in the main control room.

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BASES

BACKGROUND (continued)

The floor drain sump has level switches that start and stop the sump pumps when required. The sump pump which is selected Lead starts on a high level in the sump. The other pump starts, and a control room alarm is annunciated, if the sump level reaches the high-high level. The pumps stop when low level is reached in the sump. A timer starts each time the first sump pump starts. A second timer starts when the pump is stopped. If the pump takes longer than a given time to pump down the sump, or if the pump starts too soon after the previous pumpdown, an alarm is sounded in the control room indicating a higher than normal sump fill-up rate. A flow monitor in the discharge line of the drywell floor drain sump pumps provides flow input to a flow totalizer which is indicated in the control room. This flow totalizer can be used to quantify the amount of sump inputs.

The drywell air monitoring systems continuously monitor the drywell atmosphere for airborne particulate and gaseous radioactivity. A sudden increase of radioactivity, which may be attributed to RCPB steam or reactor water LEAKAGE, is annunciated in the control room. The drywell atmosphere particulate and gaseous radioactivity monitoring systems are not capable of quantifying leakage rates, but are sensitive enough to indicate increased LEAKAGE rates of 1 gpm within 1 hour. Larger changes in LEAKAGE rates are detected in proportionally shorter times (Ref. 3).

Condensate from the drywell coolers is routed to the drywell floor drain sump and is monitored by a flow transmitter that provides indication and alarms in the control room. This drywell air cooler condensate flow rate monitoring system serves as an added indicator, but not quantifier, of RCS unidentified LEAKAGE.

APPLICABLE SAFETY ANALYSES

A threat of significant compromise to the RCPB exists if the barrier contains a crack that is large enough to propagate rapidly. LEAKAGE rate limits are set low enough to detect the LEAKAGE emitted from a single crack in the RCPB (Refs. 4 and 5). Each of the leakage detection systems inside the drywell is designed with the capability of detecting LEAKAGE less than the established LEAKAGE rate limits and providing appropriate alarm of excess LEAKAGE in the control room.

A control room alarm allows the operators to evaluate the significance of the indicated LEAKAGE and, if necessary,

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)	shut down the reactor for further investigation and corrective action. The allowed LEAKAGE rates are well below the rates predicted for critical crack sizes (Ref. 6). Therefore, these actions provide adequate response before a significant break in the RCPB can occur. RCS leakage detection instrumentation satisfies Criterion 1 of 10 CFR 50.36(c)(2)(ii).
LCO	The drywell floor drain sump flow monitoring system is required to quantify the unidentified LEAKAGE from the RCS. Thus, for the system to be considered OPERABLE, the floor drain sump fillup rate monitor portion of the system must be OPERABLE. The other monitoring systems provide early alarms to the operators so closer examination of other detection systems will be made to determine the extent of any corrective action that may be required. With the leakage detection systems inoperable, monitoring for LEAKAGE in the RCPB is degraded.
APPLICABILITY	In MODES 1, 2, and 3, leakage detection systems are required to be OPERABLE to support LCO 3.4.5. This Applicability is consistent with that for LCO 3.4.5.
ACTIONS	A.1 With the drywell floor drain sump flow monitoring system inoperable, no other form of sampling can provide the equivalent information to quantify leakage. However, the drywell atmospheric activity monitor and the drywell air cooler condensate flow rate monitor will provide indications of changes in leakage. With the drywell floor drain sump flow monitoring system inoperable, but with RCS unidentified and total LEAKAGE being determined every 12 hours (SR 3.4.5.1), operation may continue for 30 days. The 30 day Completion Time of Required Action A.1 is acceptable, based on operating experience, considering the multiple forms of leakage detection that are still available. Required Action A.1 is modified by a Note that states that the provisions of LCO 3.0.4 are not applicable. As a result, a MODE change is allowed when the drywell floor drain sump flow monitoring

(continued)

BASES

ACTIONS

A.1 (continued)

system is inoperable. This allowance is provided because other instrumentation is available to monitor RCS leakage.

B.1

With both gaseous and particulate drywell atmospheric monitoring channels inoperable (i.e., the required drywell atmospheric monitoring system), grab samples of the drywell atmosphere shall be taken and analyzed to provide periodic leakage information. Provided a sample is obtained and analyzed every 12 hours, the plant may continue operation since at least one other form of drywell leakage detection (i.e., air cooler condensate flow rate monitor) is available.

The 12 hour interval provides periodic information that is adequate to detect LEAKAGE.

C.1

With the required drywell air cooler condensate flow rate monitoring system inoperable, SR 3.4.7.1 is performed every 8 hours to provide periodic information of activity in the drywell at a more frequent interval than the routine Frequency of SR 3.4.7.1. The 8 hour interval provides periodic information that is adequate to detect LEAKAGE and recognizes that other forms of leakage detection are available. However, this Required Action is modified by a Note that allows this action to be not applicable if the required drywell atmospheric monitoring system is inoperable. Consistent with SR 3.0.1, Surveillances are not required to be performed on inoperable equipment.

D.1 and D.2

With both the gaseous and particulate drywell atmospheric monitor channels and the drywell air cooler condensate flow rate monitor inoperable, the only means of detecting LEAKAGE is the drywell floor drain sump flow monitor. This

(continued)

BASES

ACTIONS

D.1 and D.2 (continued)

Condition does not provide the required diverse means of leakage detection. The Required Action is to restore either of the inoperable monitors to OPERABLE status within 30 days to regain the intended leakage detection diversity. The 30 day Completion Time ensures that the plant will not be operated in a degraded configuration for a lengthy time period. The Required Actions are modified by a Note that states that the provisions of LCO 3.0.4 are not applicable. As a result, a MODE change is allowed when both the gaseous and particulate primary containment atmospheric monitoring channels and air cooler condensate flow rate are inoperable. This allowance is provided because other instrumentation is available to monitor RCS leakage.

E.1 and E.2

If any Required Action of Condition A, B, C, or D cannot be met within the associated Completion Time, the plant must be brought to a MODE in which the LCO does not apply. To achieve this status, the plant must be brought to at least MODE 3 within 12 hours and to MODE 4 within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions in an orderly manner and without challenging plant systems.

F.1

With all required monitors inoperable, no required automatic means of monitoring LEAKAGE are available, and immediate plant shutdown in accordance with LCO 3.0.3 is required.

SURVEILLANCE REQUIREMENTS

The Surveillances are modified by a Note to indicate that when a channel is placed in an inoperable status solely for performance of required Surveillances, entry into associated Conditions and Required Actions may be delayed for up to 6 hours, provided the other required instrumentation (the drywell sump flow monitoring system, drywell atmospheric monitoring channel, or the drywell air cooler condensate flow monitoring system, as applicable) is OPERABLE. Upon

(continued)

BASES

SURVEILLANCE REQUIREMENTS (continued)

completion of the Surveillance, or expiration of the 6 hour allowance, the channel must be returned to OPERABLE status or the applicable Condition entered and Required Actions taken. The 6 hour testing allowance is acceptable since it does not significantly reduce the probability of properly monitoring RCS leakage.

SR 3.4.7.1

This SR requires the performance of a CHANNEL CHECK of the required drywell atmospheric monitoring system. The check gives reasonable confidence that the channel is operating properly. The Frequency of 12 hours is based on instrument reliability and is reasonable for detecting off normal conditions.

SR 3.4.7.2

This SR requires the performance of a CHANNEL FUNCTIONAL TEST of the required RCS leakage detection instrumentation. The test ensures that the monitors can perform their function in the desired manner. The test also verifies the alarm function and relative accuracy of the instrument string. The Frequency of 31 days considers instrument reliability, and operating experience has shown it proper for detecting degradation.

SR 3.4.7.3

This SR requires the performance of a CHANNEL CALIBRATION of the required RCS leakage detection instrumentation channels. The calibration verifies the accuracy of the instrument string, including the instruments located inside the drywell. The Frequency of 24 months is a typical refueling cycle and considers channel reliability. Operating experience has proven this Frequency is acceptable.

REFERENCES

1. 10 CFR 50, Appendix A, GDC 30.
2. Regulatory Guide 1.45, May 1973.
3. UFSAR, Section 5.2.5.1.1.

(continued)

BASES

REFERENCES (continued)

4. GEAP-5620, "Failure Behavior in ASTM A106B Pipes Containing Axial Through-Wall Flaws," April 1968.

5. NUREG-75/067, "Investigation and Evaluation of Cracking in Austenitic Stainless Steel Piping of Boiling Water Reactor Plants," October 1975.

6. UFSAR, Section 5.2.5.5.2.

B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.8 RCS Specific Activity

BASES

BACKGROUND

During circulation, the reactor coolant acquires radioactive materials due to release of fission products from fuel leaks into the coolant and activation of corrosion products in the reactor coolant. These radioactive materials in the coolant can plate out in the RCS, and, at times, an accumulation will break away to spike the normal level of radioactivity. The release of coolant during a Design Basis Accident (DBA) could send radioactive materials into the environment.

Limits on the maximum allowable level of radioactivity in the reactor coolant are established to ensure, in the event of a release of any radioactive material to the environment during a DBA, radiation doses are maintained within the limits of 10 CFR 100 (Ref. 1).

This LCO contains iodine specific activity limits. The iodine isotopic activities per gram of reactor coolant are expressed in terms of a DOSE EQUIVALENT I-131. The allowable levels are intended to limit the 2 hour radiation dose to an individual at the site boundary to a small fraction of the 10 CFR 100 limit.

APPLICABLE SAFETY ANALYSES

Analytical methods and assumptions involving radioactive material in the primary coolant are presented in the UFSAR (Ref. 2). The specific activity in the reactor coolant (the source term) is an initial condition for evaluation of the consequences of an accident due to a main steam line break (MSLB) outside containment. No fuel damage is postulated in the MSLB accident, and the release of radioactive material to the environment is assumed to end when the main steam isolation valves (MSIVs) close completely.

This MSLB release forms the basis for determining offsite doses (Ref. 2). The limits on the specific activity of the primary coolant ensure that the 2 hour thyroid and whole body doses at the site boundary, resulting from an MSLB

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

outside containment during steady state operation, will not exceed 10% of the dose guidelines of 10 CFR 100.

The limit on specific activity is a value from a parametric evaluation of typical site locations. This limit is conservative because the evaluation considered more restrictive parameters than for a specific site, such as the location of the site boundary and the meteorological conditions of the site.

RCS specific activity satisfies Criterion 2 of 10 CFR 50.36(c)(2)(ii).

LCO

The specific iodine activity is limited to ≤ 0.2 μCi/gm DOSE EQUIVALENT I-131. This limit ensures the source term assumed in the safety analysis for the MSLB is not exceeded, so any release of radioactivity to the environment during an MSLB is less than a small fraction of the 10 CFR 100 limits.

APPLICABILITY

In MODE 1, and MODES 2 and 3 with any main steam line not isolated, limits on the primary coolant radioactivity are applicable since there is an escape path for release of radioactive material from the primary coolant to the environment in the event of an MSLB outside of primary containment.

In MODES 2 and 3 with the main steam lines isolated, such limits do not apply since an escape path does not exist. In MODES 4 and 5, no limits are required since the reactor is not pressurized and the potential for leakage is reduced.

ACTIONS

A.1 and A.2

When the reactor coolant specific activity exceeds the LCO DOSE EQUIVALENT I-131 limit, but is ≤ 4.0 μCi/gm, samples must be analyzed for DOSE EQUIVALENT I-131 at least once every 4 hours. In addition, the specific activity must be restored to the LCO limit within 48 hours. The Completion Time of once every 4 hours is based on the time needed to take and analyze a sample. The 48 hour Completion Time to

(continued)

BASES

ACTIONS

A.1 and A.2 (continued)

restore the activity level provides a reasonable time for temporary coolant activity increases (iodine spikes or crud bursts) to be cleaned up with the normal processing systems.

A note to the Required Action of Condition A excludes the MODE change restriction of LCO 3.0.4. This exception allows entry into the applicable MODE(S) while relying on the ACTIONS even though the ACTIONS may eventually require plant shutdown. This exception is acceptable due to the significant conservatism incorporated into the specific activity limit, the low probability of an event which is limiting due to exceeding this limit, and the ability to restore transient specific activity excursions while the plant remains at, or proceeds to power operation.

B.1, B.2.1, B.2.2.1, and B.2.2.2

If the DOSE EQUIVALENT I-131 cannot be restored to $\leq$ 0.2 μCi/gm within 48 hours, or if at any time it is > 4.0 μCi/gm, it must be determined at least every 4 hours and all the main steam lines must be isolated within 12 hours. Isolating the main steam lines precludes the possibility of releasing radioactive material to the environment in an amount that is more than a small fraction of the requirements of 10 CFR 100 during a postulated MSLB accident.

Alternately, the plant can be brought to MODE 3 within 12 hours and to MODE 4 within 36 hours. This option is provided for those instances when isolation of main steam lines is not desired (e.g., due to the decay heat loads). In MODE 4, the requirements of the LCO are no longer applicable.

The Completion Time of once every 4 hours is the time needed to take and analyze a sample. The 12 hour Completion Time is reasonable, based on operating experience, to isolate the main steam lines in an orderly manner and without challenging plant systems. Also, the allowed Completion Times for Required Actions B.2.2.1 and B.2.2.2 for bringing the plant to MODES 3 and 4 are reasonable, based on

(continued)

BASES

ACTIONS

B.1, B.2.1, B.2.2.1, and B.2.2.2 (continued)

operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.4.8.1

This Surveillance is performed to ensure iodine remains within limit during normal operation. The 7 day Frequency is adequate to trend changes in the iodine activity level.

This SR is modified by a Note that requires this Surveillance to be performed only in MODE 1 because the level of fission products generated in other MODES is much less.

REFERENCES

1. 10 CFR 100.11.

2. UFSAR, Section 15.6.4.5.

B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.9 Residual Heat Removal (RHR) Shutdown Cooling System—Hot Shutdown

BASES

BACKGROUND

Irradiated fuel in the shutdown reactor core generates heat during the decay of fission products and increases the temperature of the reactor coolant. This decay heat must be removed to reduce the temperature of the reactor coolant to ≤ 200°F in preparation for performing Refueling or Cold Shutdown maintenance operations, or the decay heat must be removed for maintaining the reactor in the Hot Shutdown condition.

The two redundant, manually controlled shutdown cooling subsystems of the RHR System provide decay heat removal. Each loop consists of a motor driven pump, a heat exchanger, and associated piping and valves. Both loops have a common suction from the same recirculation loop. Each pump discharges the reactor coolant, after circulation through the respective heat exchanger, to the reactor via the associated recirculation loop. The RHR heat exchangers transfer heat to the Residual Heat Removal Service Water System (LCO 3.7.1, "Residual Heat Removal Service Water (RHRSW) System").

APPLICABLE SAFETY ANALYSES

Decay heat removal by the RHR System in the shutdown cooling mode is not required for mitigation of any event or accident evaluated in the safety analyses. Decay heat removal is, however, an important safety function that must be accomplished or core damage could result.

The RHR Shutdown Cooling System meets Criterion 4 of 10 CFR 50.36(c)(2)(ii).

LCO

Two RHR shutdown cooling subsystems are required to be OPERABLE, and, when no recirculation pump is in operation, one shutdown cooling subsystem must be in operation. An OPERABLE RHR shutdown cooling subsystem consists of one OPERABLE RHR pump, one heat exchanger, and the associated piping and valves. Each shutdown cooling subsystem is considered OPERABLE if it can be manually aligned (remote or

(continued)

BASES

LCO (continued)	local) in the shutdown cooling mode for removal of decay heat. In MODE 3, one RHR shutdown cooling subsystem can provide the required cooling, but two subsystems are required to be OPERABLE to provide redundancy. Operation of one subsystem can maintain or reduce the reactor coolant temperature as required. To ensure adequate core flow to allow for accurate average reactor coolant temperature monitoring, nearly continuous operation is required. Note 1 permits both RHR shutdown cooling subsystems and recirculation pumps to not be in operation for a period of 2 hours in an 8 hour period. Note 2 allows one RHR shutdown cooling subsystem to be inoperable for up to 2 hours for performance of surveillance tests. These tests may be on the affected RHR System or on some other plant system or component that necessitates placing the RHR System in an inoperable status during the performance. This is permitted because the core heat generation can be low enough and the heatup rate slow enough to allow some changes to the RHR subsystems or other operations requiring RHR flow interruption and loss of redundancy.
APPLICABILITY	In MODE 3 with reactor vessel pressure below the RHR cut in permissive pressure (i.e., the actual pressure at which the interlock resets) the RHR Shutdown Cooling System must be OPERABLE and one RHR shutdown cooling subsystem shall be operated in the shutdown cooling mode to remove decay heat to reduce or maintain coolant temperature. With an RHR shutdown cooling subsystem not in operation, a recirculation pump is required to be in operation. In MODES 1 and 2, and in MODE 3 with reactor vessel pressure greater than or equal to the RHR cut-in permissive pressure, this LCO is not applicable. Operation of the RHR System in the shutdown cooling mode is not allowed above this pressure because the RCS pressure may exceed the design pressure of the shutdown cooling piping. Decay heat removal at reactor pressures greater than or equal to the RHR cut-in permissive pressure is typically accomplished by condensing the steam in the main condenser. Additionally, in MODE 2, the OPERABILITY requirements for the Emergency Core Cooling Systems (ECCS) (LCO 3.5.1, "ECCS—Operating") do not allow placing the RHR shutdown cooling subsystem into operation.

(continued)

BASES

APPLICABILITY (continued)

The requirements for decay heat removal in MODES 4 and 5 are discussed in LCO 3.4.10, "Residual Heat Removal (RHR) Shutdown Cooling System — Cold Shutdown"; LCO 3.9.8, "Residual Heat Removal (RHR) — High Water Level"; and LCO 3.9.9, "Residual Heat Removal (RHR) — Low Water Level."

ACTIONS

A Note to the ACTIONS excludes the MODE change restriction of LCO 3.0.4. This exception allows entry into the applicable MODE while relying on the ACTIONS even though the ACTIONS may eventually require plant shutdown. This exception is acceptable due to the redundancy of the OPERABLE subsystems, the low pressure at which the plant is operating, the low probability of an event occurring during operation in this condition, and the availability of alternate methods of decay heat removal capability.

A second Note has been provided to modify the ACTIONS related to RHR shutdown cooling subsystems. Section 1.3, Completion Times, specifies once a Condition has been entered, subsequent divisions, subsystems, components or variables expressed in the Condition, discovered to be inoperable or not within limits, will not result in separate entry into the Condition. Section 1.3 also specifies Required Actions of the Condition continue to apply for each additional failure, with Completion Times based on initial entry into the Condition. However, the Required Actions for inoperable shutdown cooling subsystems provide appropriate compensatory measures for separate inoperable shutdown cooling subsystems. As such, a Note has been provided that allows separate Condition entry for each inoperable RHR shutdown cooling subsystem.

<u>A.1, A.2, and A.3</u>

With one RHR shutdown cooling subsystem inoperable for decay heat removal, except as permitted by LCO Note 2, the inoperable subsystem must be restored to OPERABLE status without delay. In this condition, the remaining OPERABLE subsystem can provide the necessary decay heat removal. The overall reliability is reduced, however, because a single failure in the OPERABLE subsystem could result in reduced RHR shutdown cooling capability. Therefore an alternate method of decay heat removal must be provided.

(continued)

BASES

ACTIONS

A.1, A.2, and A.3 (continued)

With both RHR shutdown cooling subsystems inoperable, an alternate method of decay heat removal must be provided in addition to that provided for the initial RHR shutdown cooling subsystem inoperability. This re-establishes backup decay heat removal capabilities, similar to the requirements of the LCO. The 1 hour Completion Time is based on the decay heat removal function and the probability of a loss of the available decay heat removal capabilities.

The required cooling capacity of the alternate method should be ensured by verifying (by calculation or demonstration) its capability to maintain or reduce temperature. Decay heat removal by ambient losses can be considered as, or contributing to, the alternate method capability. Alternate methods that can be used include (but are not limited to) the Condensate/Feed and Main Steam Systems or the Reactor Water Cleanup System (by itself or using feed and bleed in combination with the Control Rod Drive System or Condensate/Feed System), and a combination of an ECCS pump and S/RVs.

However, due to the potentially reduced reliability of the alternate methods of decay heat removal, it is also required to reduce the reactor coolant temperature to the point where MODE 4 is entered.

B.1, B.2, and B.3

With no RHR shutdown cooling subsystem and no recirculation pump in operation, except as is permitted by LCO Note 1, reactor coolant circulation by the RHR shutdown cooling subsystem or one recirculation pump must be restored without delay.

Until RHR or recirculation pump operation is re-established, an alternate method of reactor coolant circulation must be placed into service. This will provide the necessary circulation for monitoring coolant temperature. The 1 hour Completion Time is based on the coolant circulation function and is modified such that the 1 hour is applicable

(continued)

BASES

ACTIONS

B.1, B.2, and B.3 (continued)

separately for each occurrence involving a loss of coolant circulation. Furthermore, verification of the functioning of the alternate method must be reconfirmed every 12 hours thereafter. This will provide assurance of continued temperature monitoring capability.

During the period when the reactor coolant is being circulated by an alternate method (other than by the required RHR shutdown cooling subsystem or recirculation pump), the reactor coolant temperature and pressure must be periodically monitored to ensure proper function of the alternate method. The once per hour Completion Time is deemed appropriate.

SURVEILLANCE REQUIREMENTS

SR 3.4.9.1

This Surveillance verifies that one RHR shutdown cooling subsystem or recirculation pump is in operation and circulating reactor coolant. The required flow rate is determined by the flow rate necessary to provide sufficient decay heat removal capability. The Frequency of 12 hours is sufficient in view of other visual and audible indications available to the operator for monitoring the RHR subsystem in the control room.

This Surveillance is modified by a Note allowing sufficient time to align the RHR System for shutdown cooling operation after clearing the pressure interlock that isolates the system, or for placing a recirculation pump in operation. The Note takes exception to the requirements of the Surveillance being met (i.e., forced coolant circulation is not required for this initial 2 hour period), which also allows entry into the Applicability of this Specification in accordance with SR 3.0.4 since the Surveillance will not be "not met" at the time of entry into the Applicability.

REFERENCES

None.

B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.10 Residual Heat Removal (RHR) Shutdown Cooling System—Cold Shutdown

BASES

BACKGROUND

Irradiated fuel in the shutdown reactor core generates heat during the decay of fission products and increases the temperature of the reactor coolant. This decay heat must be removed to maintain the temperature of the reactor coolant at ≤ 200°F in preparation for performing refueling maintenance operations, or the decay heat must be removed for maintaining the reactor in the Cold Shutdown condition.

The two redundant, manually controlled shutdown cooling subsystems of the RHR System provide decay heat removal. Each loop consists of a motor driven pump, a heat exchanger, and associated piping and valves. Both loops have a common suction from the same recirculation loop. Each pump discharges the reactor coolant, after circulation through the respective heat exchanger, to the reactor via separate feedwater lines or to the reactor via the associated recirculation loop. The RHR heat exchangers transfer heat to the Residual Heat Removal Service Water (RHRSW) System.

APPLICABLE SAFETY ANALYSES

Decay heat removal by the RHR System in the shutdown cooling mode is not required for mitigation of any event or accident evaluated in the safety analyses. Decay heat removal is, however, an important safety function that must be accomplished or core damage could result.

The RHR Shutdown Cooling System meets Criterion 4 of 10 CFR 50.36(c)(2)(ii).

LCO

Two RHR shutdown cooling subsystems are required to be OPERABLE, and, when no recirculation pump is in operation, one RHR shutdown cooling subsystem must be in operation. An OPERABLE RHR shutdown cooling subsystem consists of one OPERABLE RHR pump, one heat exchanger, the necessary portions of the RHRSW System and Ultimate Heat Sink capable of providing cooling to the heat exchanger, and the associated piping and valves. Each shutdown cooling

(continued)

BASES

LCO (continued)	subsystem is considered OPERABLE if it can be manually aligned (remote or local) in the shutdown cooling mode for removal of decay heat. In MODE 4, one RHR shutdown cooling subsystem can provide the required cooling, but two subsystems are required to be OPERABLE to provide redundancy. Operation of one subsystem can maintain and reduce the reactor coolant temperature as required. To ensure adequate core flow to allow for accurate average reactor coolant temperature monitoring, nearly continuous operation is required. Note 1 allows both RHR shutdown cooling subsystems to be inoperable during hydrostatic testing. This is necessary since the RHR Shutdown Cooling System is not designed to operate at the Reactor Coolant System pressures achieved during hydrostatic testing. This is acceptable since adequate reactor coolant circulation will be achieved by operation of a reactor recirculation pump and since systems are available to control reactor coolant temperature. Note 2 permits both RHR shutdown cooling subsystems and recirculation pumps to not be in operation for a period of 2 hours in an 8 hour period. Note 3 allows one RHR shutdown cooling subsystem to be inoperable for up to 2 hours for performance of surveillance tests. These tests may be on the affected RHR System or on some other plant system or component that necessitates placing the RHR System in an inoperable status during the performance. This is permitted because the core heat generation can be low enough and the heatup rate slow enough to allow some changes to the RHR subsystems or other operations requiring RHR flow interruption and loss of redundancy.
APPLICABILITY	In MODE 4, the RHR Shutdown Cooling System must be OPERABLE and one RHR shutdown cooling subsystem shall be operated in the shutdown cooling mode to remove decay heat to maintain coolant temperature below 200°F. With an RHR shutdown cooling subsystem not in operation, a recirculation pump is required to be in operation. In MODES 1 and 2, and in MODE 3 with reactor vessel pressure greater than or equal to the RHR cut-in permissive pressure, this LCO is not applicable. Operation of the RHR System in the shutdown cooling mode is not allowed above this

(continued)

BASES

APPLICABILITY (continued)

pressure because the RCS pressure may exceed the design pressure of the shutdown cooling piping. Decay heat removal at reactor pressures greater than or equal to the RHR cut-in permissive pressure is typically accomplished by condensing the steam in the main condenser. Additionally, in MODE 2 below this pressure, the OPERABILITY requirements for the Emergency Core Cooling Systems (ECCS) (LCO 3.5.1, "ECCS — Operating") do not allow placing the RHR shutdown cooling subsystem into operation.

The requirements for decay heat removal in MODE 3 below the cut-in permissive pressure and in MODE 5 are discussed in LCO 3.4.9, "Residual Heat Removal (RHR) Shutdown Cooling System — Hot Shutdown"; LCO 3.9.8, "Residual Heat Removal (RHR) — High Water Level"; and LCO 3.9.9, "Residual Heat Removal (RHR) — Low Water Level."

ACTIONS

A Note has been provided to modify the ACTIONS related to RHR shutdown cooling subsystems. Section 1.3, Completion Times, specifies once a Condition has been entered, subsequent divisions, subsystems, components or variables expressed in the Condition, discovered to be inoperable or not within limits, will not result in separate entry into the Condition. Section 1.3 also specifies Required Actions of the Condition continue to apply for each additional failure, with Completion Times based on initial entry into the Condition. However, the Required Actions for inoperable shutdown cooling subsystems provided appropriate compensatory measures for separate inoperable shutdown cooling subsystems. As such, a Note has been provided that allows separate Condition entry for each inoperable RHR shutdown cooling subsystem.

A.1

With one of the two RHR shutdown cooling subsystems inoperable, except as permitted by LCO Notes 1 and 3, the remaining subsystem is capable of providing the required decay heat removal. However, the overall reliability is reduced. Therefore, an alternate method of decay heat removal must be provided. With both RHR shutdown cooling subsystems inoperable, an alternate method of decay heat

(continued)

BASES

ACTIONS

A.1 (continued)

removal must be provided in addition to that provided for the initial RHR shutdown cooling subsystem inoperability. This re-establishes backup decay heat removal capabilities, similar to the requirements of the LCO. The 1 hour Completion Time is based on the decay heat removal function and the probability of a loss of the available decay heat removal capabilities. Furthermore, verification of the functional availability of these alternate method(s) must be reconfirmed every 24 hours thereafter. This will provide assurance of continued heat removal capability.

The required cooling capacity of the alternate method should be ensured by verifying (by calculation or demonstration) its capability to maintain or reduce temperature. Decay heat removal by ambient losses can be considered as, or contributing to, the alternate method capability. Alternate methods that can be used include (but are not limited to) the Condensate/Feed and Main Steam Systems, the Reactor Water Cleanup System (by itself or using feed and bleed in combination with the Control Rod Drive System or Condensate/Feed System) and a combination of an ECCS pump and S/RVs.

B.1 and B.2

With no RHR shutdown cooling subsystem and no recirculation pump in operation, except as is permitted by LCO Notes 1 and 2, and until RHR or recirculation pump operation is re-established, an alternate method of reactor coolant circulation must be placed into service. This will provide the necessary circulation for monitoring coolant temperature. The 1 hour Completion Time is based on the coolant circulation function and is modified such that the 1 hour is applicable separately for each occurrence involving a loss of coolant circulation. Furthermore, verification of the functioning of the alternate method must be reconfirmed every 12 hours thereafter. This will provide assurance of continued temperature monitoring capability.

(continued)

BASES

ACTIONS

B.1 and B.2 (continued)

During the period when the reactor coolant is being circulated by an alternate method (other than by the required RHR shutdown cooling subsystem or recirculation pump), the reactor coolant temperature and pressure must be periodically monitored to ensure proper function of the alternate method. The once per hour Completion Time is deemed appropriate.

SURVEILLANCE REQUIREMENTS

SR 3.4.10.1

This Surveillance verifies that one RHR shutdown cooling subsystem or recirculation pump is in operation and circulating reactor coolant. The required flow rate is determined by the flow rate necessary to provide sufficient decay heat removal capability. The Frequency of 12 hours is sufficient in view of other visual and audible indications available to the operator for monitoring the RHR subsystem in the control room.

REFERENCES

None.

B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.11 RCS Pressure and Temperature (P/T) Limits

BASES

BACKGROUND

All components of the RCS are designed to withstand effects of cyclic loads due to system pressure and temperature changes. These loads are introduced by startup (heatup) and shutdown (cooldown) operations, power transients, and reactor trips. This LCO limits the pressure and temperature changes during RCS heatup and cooldown, within the design assumptions and the stress limits for cyclic operation.

The Specification contains P/T limit curves for heatup, cooldown, inservice leak and hydrostatic testing, and criticality and also limits the maximum rate of change of reactor coolant temperature. The P/T limit curves are applicable for 32 effective full power years.

Each P/T limit curve defines an acceptable region for normal operation. The usual use of the curves is operational guidance during heatup or cooldown maneuvering, when pressure and temperature indications are monitored and compared to the applicable curve to determine that operation is within the allowable region.

The LCO establishes operating limits that provide a margin to brittle failure of the reactor vessel and piping of the reactor coolant pressure boundary (RCPB). The vessel is the component most subject to brittle failure. Therefore, the LCO limits apply mainly to the vessel.

10 CFR 50, Appendix G (Ref. 1), requires the establishment of P/T limits for material fracture toughness requirements of the RCPB materials. Reference 1 requires an adequate margin to brittle failure during normal operation, anticipated operational occurrences, and system hydrostatic tests. It mandates the use of the American Society of Mechanical Engineers (ASME) Code, Section III, Appendix G (Ref. 2).

The actual shift in the RT_{NDT} of the vessel material will be established periodically by removing and evaluating the irradiated reactor vessel material specimens, in accordance with ASTM E 185 (Ref. 3) and 10 CFR 50, Appendix H (Ref. 4). The operating P/T limit curves will be adjusted,

(continued)

BASES

BACKGROUND (continued)

as necessary, based on the evaluation findings and the recommendations of Reference 5.

The P/T limit curves are composite curves established by superimposing limits derived from stress analyses of those portions of the reactor vessel and head that are the most restrictive. At any specific pressure, temperature, and temperature rate of change, one location within the reactor vessel will dictate the most restrictive limit. Across the span of the P/T limit curves, different locations are more restrictive, and, thus, the curves are composites of the most restrictive regions.

The non-nuclear heatup and cooldown curve 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.

The P/T criticality limits include the Reference 1 requirement that they be at least 40°F above the non-critical heatup curve or the cooldown curve and not lower than the minimum permissible temperature for the inservice leak and hydrostatic testing.

The consequence of violating the LCO limits is that the RCS has been operated under conditions that can result in brittle failure of the RCPB, possibly leading to a nonisolable leak or loss of coolant accident. In the event these limits are exceeded, an evaluation must be performed to determine the effect on the structural integrity of the RCPB components. The ASME Code, Section XI, Appendix E (Ref. 6), provides a recommended methodology for evaluating an operating event that causes an excursion outside the limits.

APPLICABLE SAFETY ANALYSES

The P/T limits are not derived from Design Basis Accident (DBA) analyses. They are prescribed during normal operation to avoid encountering pressure, temperature, and temperature rate of change conditions that might cause undetected flaws to propagate and cause nonductile failure of the RCPB, a condition that is unanalyzed. Reference 7 approved the curves and limits required by this Specification. Since the

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

P/T limits are not derived from any DBA, there are no acceptance limits related to the P/T limits. Rather, the P/T limits are acceptance limits themselves since they preclude operation in an unanalyzed condition.

RCS P/T limits satisfy Criterion 2 of 10 CFR 50.36(c)(2)(ii).

LCO

The elements of this LCO are:

a. RCS pressure and temperature are within the limits specified in Figures 3.4.11-1, 3.4.11-2, 3.4.11-3, 3.4.11-4, 3.4.11-5, and 3.4.11-6 heatup and cooldown rates are ≤ 100°F in any 1 hour period during RCS heatup, cooldown, and inservice leak and hydrostatic testing, and the RCS temperature change during system leakage and hydrostatic testing is ≤ 20°F in any 1 hour period when the RCS temperature and pressure are not within the limits of Figure 3.4.11-2 and 3.4.11-5 as applicable;

b. The temperature difference between the reactor vessel bottom head coolant and the reactor pressure vessel (RPV) coolant is ≤ 145°F during recirculation pump startup in MODES 1, 2, 3, and 4;

c. The temperature difference between the reactor coolant in the respective recirculation loop and in the reactor vessel is ≤ 50°F during recirculation pump startup in MODES 1, 2, 3, and 4;

d. RCS pressure and temperature are within the applicable criticality limits specified in Figures 3.4.11-3 and 3.4.11-6, prior to achieving criticality; and

e. The reactor vessel flange and the head flange temperatures are ≥ 72°F for Unit 1 and ≥ 86°F for Unit 2 when tensioning the reactor vessel head bolting studs and when the reactor head is tensioned.

These limits define allowable operating regions and permit a large number of operating cycles while also providing a wide margin to nonductile failure.

(continued)

BASES

LCO (continued)

The rate of change of temperature limits control the thermal gradient through the vessel wall and are used as inputs for calculating the heatup, cooldown, and inservice leak and hydrostatic testing P/T limit curves. Thus, the LCO for the rate of change of temperature restricts stresses caused by thermal gradients and also ensures the validity of the P/T limit curves.

Violation of the limits places the reactor vessel outside of the bounds of the stress analyses and can increase stresses in other RCS components. The consequences depend on several factors, as follows:

a. The severity of the departure from the allowable operating pressure temperature regime or the severity of the rate of change of temperature;

b. The length of time the limits were violated (longer violations allow the temperature gradient in the thick vessel walls to become more pronounced); and

c. The existence, size, and orientation of flaws in the vessel material.

APPLICABILITY

The potential for violating a P/T limit exists at all times. For example, P/T limit violations could result from ambient temperature conditions that result in the reactor vessel metal temperature being less than the minimum allowed temperature for boltup. Therefore, this LCO is applicable even when fuel is not loaded in the core.

ACTIONS

A.1 and A.2

Operation outside the P/T limits while in MODE 1, 2, or 3 must be corrected so that the RCPB is returned to a condition that has been verified by stress analyses.

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.

(continued)

BASES

ACTIONS

A.1 and A.2 (continued)

Besides restoring operation within limits, an engineering evaluation is required to determine if RCS operation can continue. The evaluation must verify the RCPB 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. ASME Code, Section XI, Appendix E (Ref. 6), may be used to support the evaluation. However, its use is restricted to evaluation of the vessel beltline.

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.

Condition A is modified by a Note requiring Required Action A.2 be completed whenever the Condition is entered. The Note emphasizes the need to perform the evaluation of the effects of the excursion outside the allowable limits. Restoration alone per Required Action A.1 is insufficient because higher than analyzed stresses may have occurred and may have affected the RCPB integrity.

B.1 and B.2

If a Required Action and associated Completion Time of Condition A are not met, the plant must be brought to a lower MODE because either the RCS remained in an unacceptable P/T region for an extended period of increased stress, or a sufficiently severe event caused entry into an unacceptable region. Either possibility indicates a need for more careful examination of the event, best accomplished with the RCS at reduced pressure and temperature. With the reduced pressure and temperature conditions, the possibility of propagation of undetected flaws is decreased.

Pressure and temperature are reduced by bringing the plant to at least MODE 3 within 12 hours and to MODE 4 within 36 hours. The allowed Completion Times are reasonable,

(continued)

BASES

ACTIONS

B.1 and B.2 (continued)

based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

C.1 and C.2

Operation outside the P/T limits in other than MODES 1, 2, and 3 (including defueled conditions) must be corrected so that the RCPB is returned to a condition that has been verified by stress analyses. The Required Action must be initiated without delay and continued until the limits are restored.

Besides restoring the P/T limit parameters to within limits, an engineering evaluation is required to determine if RCS operation is allowed. This evaluation must verify that the RCPB integrity is acceptable and must be completed before approaching criticality or heating up to > 200°F. Several methods may be used, including comparison with pre-analyzed transients, new analyses, or inspection of the components. ASME Section XI, Appendix E (Ref. 6), may be used to support the evaluation; however, its use is restricted to evaluation of the beltline.

Condition C is modified by a Note requiring Required Action C.2 be completed whenever the Condition is entered. The Note emphasizes the need to perform the evaluation of the effects of the excursion outside the allowable limits. Restoration alone per Required Action C.1 is insufficient because higher than analyzed stresses may have occurred and may have affected the RCPB integrity.

SURVEILLANCE REQUIREMENTS

SR 3.4.11.1

Verification that operation is within limits is required every 30 minutes when RCS pressure and temperature conditions are undergoing planned changes. This Frequency is considered reasonable in view of the control room indication available to monitor RCS status. Also, since temperature rate of change limits are specified in hourly increments, 30 minutes permits assessment and correction of

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.4.11.1 (continued)

minor deviations. The limits of Figures 3.4.11-1, 3.4.11-2, 3.4.11-3, 3.4.11-4, 3.4.11-5, and 3.4.11-6 are met when operation is to the right of the applicable curve.

Surveillance for heatup, cooldown, or inservice leak and hydrostatic testing may be discontinued when the criteria given in the relevant plant procedure for ending the activity are satisfied.

This SR has been modified by a Note that requires this Surveillance to be performed only during system heatup and cooldown operations and inservice leak and hydrostatic testing.

SR 3.4.11.2

A separate limit is used when the reactor is approaching criticality. Consequently, the RCS pressure and temperature must be verified within the appropriate limits before withdrawing control rods that will make the reactor critical. The limits of Figures 3.4.11-3 and 3.4.11-6 are met when operation is to the right of the applicable curve.

Performing the Surveillance within 15 minutes before control rod withdrawal for the purpose of achieving criticality provides adequate assurance that the limits will not be exceeded between the time of the Surveillance and the time of the control rod withdrawal.

SR 3.4.11.3 and SR 3.4.11.4

Differential temperatures within the applicable limits ensure that thermal stresses resulting from the startup of an idle recirculation pump will not exceed design allowances. In addition, compliance with these limits ensures that the assumptions of the analysis for the startup of an idle recirculation loop (Ref. 8) are satisfied.

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.4.11.3 and SR 3.4.11.4 (continued)

Performing the Surveillance within 15 minutes before starting the idle recirculation pump provides adequate assurance that the limits will not be exceeded between the time of the Surveillance and the time of the idle pump start.

An acceptable means of demonstrating compliance with the temperature differential requirement in SR 3.4.11.3 is to compare temperatures of the reactor pressure vessel steam space coolant and the bottom head drain line coolant.

An acceptable means of demonstrating compliance with the temperature differential requirement in SR 3.4.11.4 is to compare the temperatures of the operating recirculation loop and the idle loop.

SR 3.4.11.3 and SR 3.4.11.4 have been modified by a Note that requires the Surveillance to be met only in MODES 1, 2, 3, and 4 during a recirculation pump startup since this is when the stresses occur. In MODE 5, the overall stress on limiting components is lower; therefore, ΔT limits are not required.

SR 3.4.11.5, SR 3.4.11.6, and SR 3.4.11.7

Limits on the reactor vessel flange and head flange temperatures are generally bounded by the other P/T limits during system heatup and cooldown. However, operations approaching MODE 4 from MODE 5 and in MODE 4 with RCS temperature less than or equal to certain specified values require assurance that these temperatures meet the LCO limits.

The flange temperatures must be verified to be above the limits within 30 minutes before and every 30 minutes thereafter while tensioning the vessel head bolting studs to ensure that once the head is tensioned the limits are satisfied. When in MODE 4 with RCS temperature ≤ 77°F for Unit 1 and ≤ 91°F for Unit 2, 30 minute checks of the flange temperatures are required because of the reduced margin to the limits. When in MODE 4 with RCS temperature ≤ 92°F for

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.4.11.5, SR 3.4.11.6, and SR 3.4.11.7 (continued)

Unit 1 and ≤ 106°F for Unit 2, monitoring of the flange temperature is required every 12 hours to ensure the temperatures are within the specified limits.

The 30 minute Frequency reflects the urgency of maintaining the temperatures within limits, and also limits the time that the temperature limits could be exceeded. The 12 hour Frequency is reasonable based on the rate of temperature change possible at these temperatures.

SR 3.4.11.5 is modified by a Note that requires the Surveillance to be performed only when tensioning the reactor vessel head bolting studs. SR 3.4.11.6 is modified by a Note that requires the Surveillance to be initiated 30 minutes after RCS temperature ≤ 77°F for Unit 1 and ≤ 91°F for Unit 2 in MODE 4, SR 3.4.11.7 is modified by a Note that requires the Surveillance to be initiated 12 hours after RCS temperature ≤ 92°F for Unit 1 and ≤ 106°F for Unit 2 in MODE 4. The Notes contained in these SRs are necessary to specify when the reactor vessel flange and head flange temperatures are required to be verified to be within the specified limits.

REFERENCES

1. 10 CFR 50, Appendix G.
2. ASME, Boiler and Pressure Vessel Code, Section III, Appendix G.
3. ASTM E 185.
4. 10 CFR 50, Appendix H.
5. Regulatory Guide 1.99, Revision 2, May 1988.
6. ASME, Boiler and Pressure Vessel Code, Section XI, Appendix E.
[7. NRC Safety Evaluation supporting Amendment No. 71 to Facility Operating License No. NPF-11 and Amendment No. 55 to Facility Operating License No. NPF-18 – LaSalle County Station, Units 1 and 2, dated January 16, 1990.]
8. UFSAR, Section 15.4.4.

B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.12 Reactor Steam Dome Pressure

BASES

BACKGROUND	The reactor steam dome pressure is an assumed value in the determination of compliance with reactor pressure vessel overpressure protection criteria and is also an assumed initial condition of Design Basis Accidents (DBAs) and transients.
APPLICABLE SAFETY ANALYSES	The reactor steam dome pressure of $\leq$ 1020 psig is an initial condition of the vessel overpressure protection analysis of Reference 1. This analysis assumes an initial maximum reactor steam dome pressure and evaluates the response of the pressure relief system, primarily the safety/relief 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. Reference 2 also assumes an initial reactor steam dome pressure for the analysis of DBAs and transients used to determine the limits for fuel cladding integrity MCPR (see Bases for LCO 3.2.2, "MINIMUM CRITICAL POWER RATIO (MCPR)") and 1% cladding plastic strain (see Bases for LCO 3.2.1, "AVERAGE PLANAR LINEAR HEAT GENERATION RATE (APLHGR)" and LCO 3.2.3, "LINEAR HEAT GENERATION RATE (LHGR)"). The nominal reactor operating pressure is approximately 1005 psig. Transient analyses typically use the nominal or a design dome pressure as input to the analysis. Small deviations (5 to 10 psi) from the nominal pressure are not expected to change most of the transient analyses results. However, sensitivity studies for fast pressurization events (main turbine generator load rejection without bypass, turbine trip without bypass, and feedwater controller failure) indicate that the delta-CPR may increase for lower initial pressures. Therefore, the fast pressurization events have considered a bounding initial pressure based on a typical operating range to assure a conservative delta-CPR and operating limit. Reactor steam dome pressure satisfies the requirements of Criterion 2 of 10 CFR 50.36(c)(2)(ii).

(continued)

BASES (continued)

LCO	The specified reactor steam dome pressure limit of $\leq$ 1020 psig ensures the plant is operated within the assumptions of the reactor overpressure analysis. Operation above the limit may result in a transient response more severe than analyzed.
APPLICABILITY	In MODES 1 and 2, the reactor steam dome pressure is required to be less than or equal to the limit. In these MODES, the reactor may be generating significant steam, and events that may challenge the overpressure limits are possible. In MODES 3, 4, and 5, the limit is not applicable because the reactor is shut down. In these MODES, the reactor pressure is well below the required limit, and no anticipated events will challenge the overpressure limits.
ACTIONS	<u>A.1</u> With the reactor steam dome pressure greater than the limit, prompt action should be taken to reduce pressure to below the limit and return the reactor to operation within the bounds of the analyses. The 15 minute Completion Time is reasonable considering the importance of maintaining the pressure within limits. This Completion Time also ensures that the probability of an accident while pressure is greater than the limit is minimal. <u>B.1</u> If the reactor steam dome pressure cannot be restored to within the limit within the associated Completion Time, the plant must be brought to a MODE in which the LCO does not apply. To achieve this status, the plant must be brought to at least MODE 3 within 12 hours. The allowed Completion Time of 12 hours is reasonable, based on operating experience, to reach MODE 3 from full power conditions in an orderly manner and without challenging plant systems.

(continued)

BASES (continued)

SURVEILLANCE REQUIREMENTS	<u>SR 3.4.12.1</u> Verification that reactor steam dome pressure is $\leq$ 1020 psig ensures that the initial condition of the vessel overpressure protection analysis is met. Operating experience has shown the 12 hour Frequency to be sufficient for identifying trends and verifying operation within safety analyses assumptions.
REFERENCES	1. UFSAR, Section 5.2.2.2.1. 2. UFSAR, Chapter 15.

3/4.4 REACTOR COOLANT SYSTEM A.1

3/4.4.1 RECIRCULATION SYSTEM

RECIRCULATION LOOPS A.2

LIMITING CONDITION FOR OPERATION

LCO 3.4.1 3.4.1.1 Two reactor coolant system recirculation loops shall be in operation.

APPLICABILITY: OPERATIONAL CONDITIONS 1 and 2

ACTION

within Region III of Figure 3.4.1-1 A3

LCO 3.4.1 a. With only one (1) reactor coolant system recirculation loop in operation, ~~comply with Specification 3.4.1.5~~ and: L.1

ACTION G 1. Within ~~four (4)~~ 12 hours: satisfy the requirements of the LCO A.2

a) ~~Place the recirculation flow control system in the Master Manual mode or lower, and~~ LA.1

b) ~~Increase the MINIMUM CRITICAL POWER RATIO (MCPR) Safety Limit by 0.01 per Specification 2.1.2, and~~ A.4

LCO 3.4.1 c) Increase the MINIMUM CRITICAL POWER RATIO (MCPR) Limiting Condition for Operation ~~by 0.01~~ as specified in the COLR for Single Loop Operation per Specification 3.2.3, and, LA.6

d) Reduce the Average Power Range Monitor (APRM) Scram ~~and Rod Block~~ and Rod Block Monitor ~~Trip Setpoints~~ and Allowable Values to those applicable to single recirculation loop operation per Specifications 2.2.1 and 3.3.6. A.5

e) Reduce the AVERAGE PLANAR LINEAR HEAT GENERATION RATE (APLHGR) Limiting Condition for Operation by the applicable Single Loop Operation (SLO) factor specified in the CORE OPERATING LIMITS REPORT.

ACTION H 2. Otherwise, be in at least HOT SHUTDOWN within the next twelve (12) hours.

ACTION D b. With no reactor coolant recirculation loops in operation:

~~1. Take the ACTION required by Specification 3.4.1.5 and~~ A.3

2. Be in at least HOT SHUTDOWN within ~~the next six (6)~~ 12 hours. L.2

REACTOR COOLANT SYSTEM

RECIRCULATION LOOP FLOW

A.1

LIMITING CONDITION FOR OPERATION

jet pump A.6

LCO 3.4.1 3.4.1.3 Recirculation loop flow mismatch shall be maintained within:

SR 3.4.1.1

a. 5% of rated recirculation flow with core flow greater than or equal to 70% of rated core flow.

b. 10% of rated recirculation flow with core flow less than 70% of rated core flow.

APPLICABILITY: OPERATIONAL CONDITIONS 1 and 2 during two recirculation loop operation.

ACTION:

ACTION F

With recirculation loop flows different by more than the specified limits, either:

A.7

a. ~~Restore the~~ recirculation loop flows ~~to~~ within ~~the~~ specified ~~limit~~ within 2 hours, or

b. Declare the recirculation loop with the lower flow not in operation. ~~and take the ACTION require by Specification 3.4.1.1.~~ A.14

SURVEILLANCE REQUIREMENTS

Add proposed SR 3.4.1.1 Note L.3

SR 3.4.1.1 4.4.1.3 Recirculation loop flows shall be verified to be within the limits at least once per 24 hours.

jet pump A.6

3/4.4 REACTOR COOLANT SYSTEM

3/4.4.1 RECIRCULATION SYSTEM

THERMAL HYDRAULIC STABILITY

A.1

LIMITING CONDITION FOR OPERATION

LCO 3.4.1

3.4.1.5 Forced core circulation shall be maintained with:

a. ~~Total core flow greater than or equal to 45% of rated core flow,~~ or — A.8

b. THERMAL POWER within Region III of Figure 3.4.1.5-1, or

Action A

c. THERMAL POWER within Region II of Figure 3.4.1.5-1 AND APRM and LPRM noise levels not exceeding the larger of: i) Three (3) times the established baseline noise levels or, ii) 10% peak-to-peak indicated noise level. — A.9

APPLICABILITY: OPERATIONAL CONDITION 1 and 2 — A.13

ACTION

Action C

a. In Region I of Figure 3.4.1.5-1:

1. With at least 1 reactor coolant recirculation loop in operation ~~immediately initiate action to:~~

a) ~~Decrease THERMAL POWER by control rod insertion, completing the power decrease~~ within two (2) hours ~~to~~ exit Region I or,

b) ~~Increase core flow with the operating Recirculation Loop(s), to~~ exit Region I within two (2) hours. — LA.2

Action D

2. With no reactor coolant recirculation loops in operation: — LA.2

a) ~~Immediately~~ reduce CORE THERMAL POWER ~~by inserting control rods,~~ observing the indicated APRM and LPRM noise levels, and complete power reduction to below 36% of RATED CORE THERMAL POWER within two (2) hours, and — A.10

Action E

b) If indicated LPRM or APRM noise levels exceed 10% peak-to-peak, immediately place the reactor mode switch in the SHUTDOWN position. — A.3

Required Action D.3

c) ~~Comply with Specification 3.4.1.1 ACTION b.2~~ — Be in MODE 3 in 12 hours — L.2

REACTOR COOLANT SYSTEM

A.1

ACTION (Continued)

Action B

b. In Region II of Figure 3.4.1.5-1, with APRM or LPRM neutron flux noise levels exceeding the larger of: i) Three (3) times the established baseline noise levels, or ii) 10% peak-to-peak noise indication.

LA.3

1. ~~Immediately initiate corrective action by inserting control rods or increasing core flow to restore the noise levels to within the required limit within 2 hours, otherwise.~~

LA.3

2. ~~Insert control rods to reduce THERMAL POWER and/or increase core flow to~~ enter Region III of Figure 3.4.1.5-1 within the ~~next~~ 2 hours.

M.2

Add Action E

M.3

SURVEILLANCE REQUIREMENTS

Action A

4.4.1.5 When operating within Region II of Figure 3.4.1.5-1, verify:

1. That the APRM and LPRM neutron flux noise levels do not exceed the larger of: i) Three (3) times the established baseline levels or, ii) 10% peak-to-peak indicated noise level:

   a. At least once per 12 hours, and

   LA.4

   b. ~~Initiate the surveillance within 15 minutes~~ after entering the region or completing an increase of at least 5% of RATED THERMAL POWER, ~~completing the surveillance~~ within ~~the next 30~~ 45 minutes.

   A.12

2. That ~~core flow is greater than or equal to 39% of rated core flow~~ operation is not within Region I of Figure 3.4.1-1 at least once per 12 hours.

A.9

Add proposed SR 3.4.1.2

M.4

LA.5

~~#Detector levels A and C of one LPRM string per core octant plus detector levels A and C of one LPRM string in the center region of the core should be monitored.~~

Figure 3.4.1-1 (page 1 of 1)
Power versus Flow

A.1

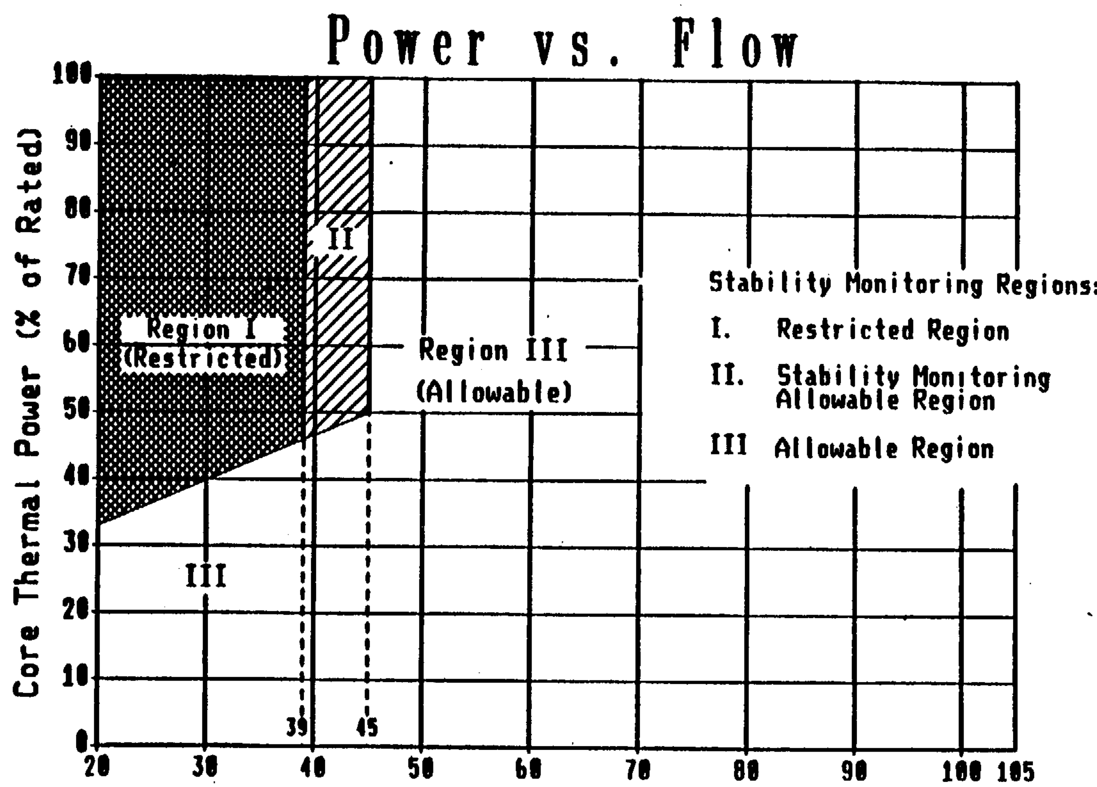

Figure 3.4.1.5-1

ITS 3.4.1

3/4.4 REACTOR COOLANT SYSTEM

A.1

3/4.4.1 RECIRCULATION SYSTEM

RECIRCULATION LOOPS

A.2

LIMITING CONDITION FOR OPERATION

LCO 3.4.1 **3.4.1.1** Two reactor coolant system recirculation loops shall be in operation.

APPLICABILITY: OPERATIONAL CONDITIONS 1 and 2 within Region III of Figure 3.4.1-1 A.3

ACTION

LCO 3.4.1 a. With only one (1) reactor coolant system recirculation loop in operation, ~~comply with Specification 3.4.1.5~~ and: L.1

ACTION G 1. Within ~~four (4)~~ 12 hours: satisfy the requirements of the LCO A.2

~~a) Place the recirculation flow control system in the Master Manual mode or lower, and~~ LA.1

~~b) Increase the MINIMUM CRITICAL POWER RATIO (MCPR) Safety Limit by 0.01 per Specification 2.1.2, and~~ A.4

LCO 3.4.1 as specified in the COLR for Single Loop Operation

c) Increase the MINIMUM CRITICAL POWER RATIO (MCPR) Limiting Condition for Operation ~~by 0.01~~ per Specification 3.2.3, and, LA.6

d) Reduce the Average Power Range Monitor (APRM) Scram ~~and Rod Block~~ and Rod Block Monitor ~~Trip Setpoints~~ and Allowable Values to those applicable to single recirculation loop operation per Specifications 2.2.1 and 3.3.6. A.5

e) Reduce the AVERAGE PLANAR LINEAR HEAT GENERATION RATE (APLHGR) Limiting Condition for Operation by the applicable Single Loop Operation (SLO) factor specified in the CORE OPERATING LIMITS REPORT.

ACTION H 2. Otherwise, be in at least HOT SHUTDOWN within the next twelve (12) hours.

ACTION D b. With no reactor coolant recirculation loops in operation:

~~1. Take the ACTION required by Specification 3.4.1.5, and~~ A.3

2. Be in at least HOT SHUTDOWN within ~~the next six (6)~~ 12 hours. L.2

REACTOR COOLANT SYSTEM

A.1

RECIRCULATION LOOP FLOW

LIMITING CONDITION FOR OPERATION

jet pump A.6

LCO 3.4.1 3.4.1.3 Recirculation loop flow mismatch shall be maintained within:

SR 3.4.1.1

a. 5% of rated recirculation flow with core flow greater than or equal to 70% of rated core flow.

b. 10% of rated recirculation flow with core flow less than 70% of rated core flow.

APPLICABILITY: OPERATIONAL CONDITIONS 1 and 2 during two recirculation loop operation.

ACTION:

Action F

With recirculation loop flows different by more than the specified limits, either:

a. ~~Restore the recirculation loop flows to within the specified limit within 2 hours, or~~ A.7

b. Declare the recirculation loop with the lower flow not in operation. ~~and take the ACTION required by Specification 3.4.1.1.~~ A.14

SURVEILLANCE REQUIREMENTS

Add proposed SR 3.4.1.1 Note L.3

SR 3.4.1.1 4.4.1.3 Recirculation loop flows shall be verified to be within the limits at least once per 24 hours.

jet pump A.6

REACTOR COOLANT SYSTEM

A.1

3/4.4 REACTOR COOLANT SYSTEM

3/4.4.1 RECIRCULATION SYSTEM

THERMAL HYDRAULIC STABILITY

LIMITING CONDITION FOR OPERATION

LCO 3.4.1

3.4.1.5 Forced core circulation shall be maintained with:

a. ~~Total core flow greater than or equal to 45% of rated core flow, or~~ A.8

b. THERMAL POWER within Region III of Figure 3.4.1.5-1, or

Action A

c. THERMAL POWER within Region II of Figure 3.4.1.5-1 AND APRM and LPRM noise levels not exceeding the larger of: i) Three (3) times the established baseline noise levels or, ii) 10% peak-to-peak indicated noise level. A.9

APPLICABILITY: OPERATIONAL CONDITION 1 and 2 A.13

ACTION

Action C

a. In Region I of Figure 3.4.1.5-1:

1. With at least 1 reactor coolant recirculation loop in operation ~~immediately initiate action to:~~

   a) ~~Decrease THERMAL POWER by control rod insertion, completing the power decrease~~ within two (2) hours ~~to~~ exit Region I or,

   b) ~~Increase core flow with the operating Recirculation Loop(s), to~~ exit Region I within two (2) hours.

LA.2

2. With no reactor coolant recirculation loops in operation: LA.2

Action D

   a) ~~Immediately~~ reduce CORE THERMAL POWER ~~by inserting control rods, observing the~~ indicated APRM and LPRM noise levels, and complete power reduction to below 36% of RATED CORE THERMAL POWER within two (2) hours, and A.10

Action E

   b) If indicated LPRM or APRM noise levels exceed 10% peak-to-peak, immediately place the reactor mode switch in the SHUTDOWN position.

Required Action D.3

   c) ~~Comply with Specification 3.4.1.1 ACTION b.2~~ A.3

Be in MODE 3 in 12 hours

L.2

REACTOR COOLANT SYSTEM

A.1

ACTION (Continued)

Action B

b. In Region II of Figure 3.4.1.5-1, with APRM or LPRM neutron flux noise levels exceeding the larger of: i) Three (3) times the established baseline noise levels, or ii) 10% peak-to-peak noise indication.

LA.3

1. ~~Immediately initiate corrective action by inserting control rods or increasing core flow to restore the noise levels to within the required limit within 2 hours, otherwise.~~

2. ~~Insert control rods to reduce THERMAL POWER and/or increase core flow to~~ enter Region III of Figure 3.4.1.5-1 within the ~~next~~ 2 hours.

LA.3

M.2

Add ACTION E

M.3

SURVEILLANCE REQUIREMENTS

Action A

4.4.1.5 When operating within Region II of Figure 3.4.1.5-1, verify:

1. That the APRM and LPRM neutron flux noise levels do not exceed the larger of: i) Three (3) times the established baseline levels or, ii) 10% peak-to-peak indicated noise level:

a. At least once per 12 hours, and

b. ~~Initiate the surveillance within 15 minutes~~ after entering the region or completing an increase of at least 5% of RATED THERMAL POWER, ~~completing the surveillance~~ within ~~the next 30~~ 45 minutes.

LA.4

A.12

2. That ~~core flow is greater than or equal to 39% of rated core flow~~ operation is not within Region I of Figure 3.4.1-1 at least once per 12 hours.

A.9

Add proposed SR 3.4.1.2

M.4

~~#Detector levels A and C of one LPRM string per core octant plus detector levels A and C of one LPRM string in the center region of the core should be monitored.~~

LA.5

Figure 3.4.1-1 (Page 1 of 1)
Power versus Flow

A.1

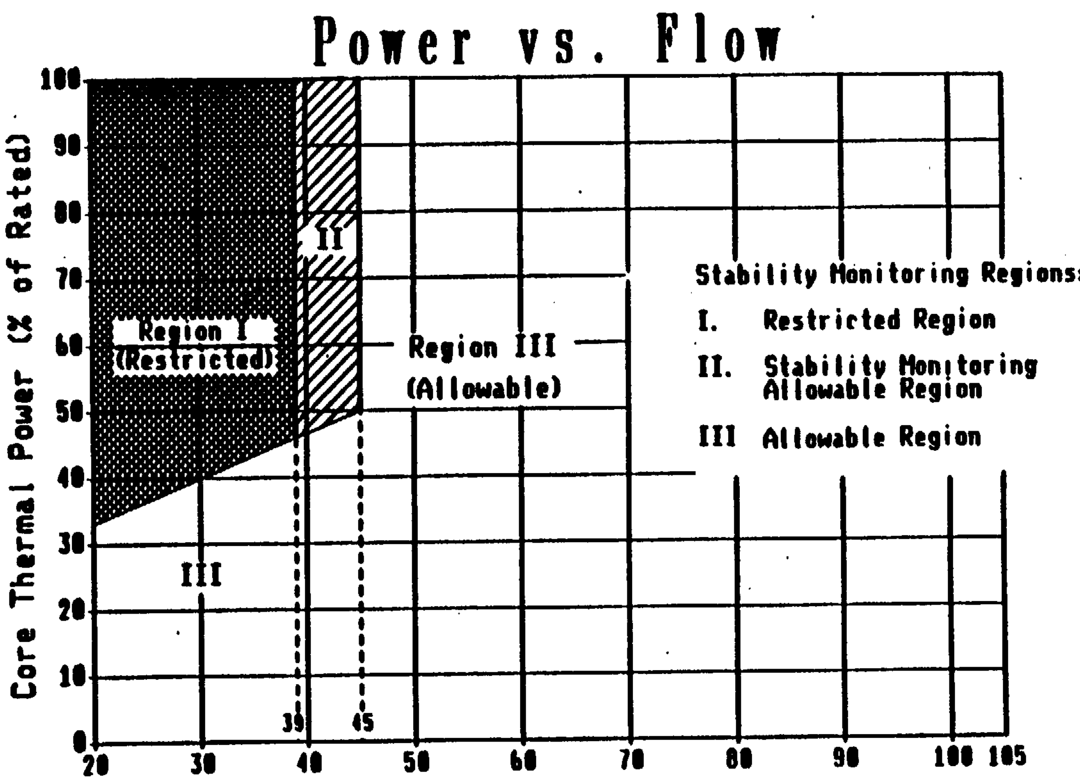

Figure 3.4.1.5-1

ITS 3.4.1

ADMINISTRATIVE

A.1 In the conversion of the LaSalle 1 and 2 current Technical Specifications (CTS) to the proposed plant specific Improved Technical Specifications (ITS), certain wording preferences or conventions are adopted that do not result in technical changes (either actual or interpretational). Editorial changes, reformatting, and revised numbering are adopted to make the ITS consistent with the BWR Standard Technical Specifications, NUREG-1434, Rev. 1 (i.e., the Improved Standard Technical Specifications (ISTS)).

A.2 CTS 3.4.1.1 requires both recirculation loops to be in operation. When one loop is inoperable, CTS 3.4.1.1 Action a provides requirements that allow continued operation with only one recirculation loop in operation. CTS 3.4.1.1 has been rewritten into two distinct options in ITS 3.4.1, with the first option of ITS 3.4.1 requiring two recirculation loops and the second option of ITS 3.4.1 only requiring one recirculation loop with the added requirements of CTS 3.4.1.1 ACTIONS a.1.c), a.1.d and a.1.e). Since these specific requirements are now part of the LCO, CTS 3.4.1.1 Action a.1 (ITS 3.4.1 ACTION G) has been modified to require compliance with the requirements of the LCO. This change is for ease of use and understanding only, and thus is administrative.

A.3 CTS 3.4.1.1 Action a requires compliance with Specification 3.4.1.5 when only one reactor coolant recirculation loop is in operation. Compliance with Specification 3.4.1.5 is already required by CTS 3.0.1 (ITS LCO 3.0.1). In addition, CTS 3.4.1.1 Action b.1 requires performing the Actions of Specification 3.4.1.5 when no reactor coolant recirculation loops are in operation. CTS 3.4.1.5 Action a.2 is already required when CTS 3.4.1.5 is not met (i.e., when forced circulation is not maintained). Also, when no recirculation loops are in operation, CTS 3.4.1.5 Action a.2.c) requires performing Specification 3.4.1.1 Action b.2. This action is already required when no recirculation loops are in service per CTS 3.4.1.1. Since the compliance with other Technical Specifications and performance of other Technical Specifications Actions are already required, the actions to require compliance with another Specification and its associated Actions are redundant and unnecessary. Therefore, the proposed change is considered administrative in nature.

ADMINISTRATIVE (continued)

A.4 CTS 3.4.1.1 Action a.1.b) requires an increase of the MCPR safety limit per CTS 2.1.2 when only one recirculation loop is in operation. The requirement is not included in ITS 3.4.1 since the Safety Limit requirement (CTS 2.1.2) is currently specified as the single loop limit; thus, when the plant is in single loop, the limit applies immediately, not in 4 hours as allowed by CTS 3.4.1.1 Action a.1.b). The ITS maintains the single loop MCPR safety limit in ITS 2.1.1.2.

A.5 The requirements in CTS 3.4.1.1 Action a.1.d) to reduce the Average Power Range Monitor (APRM) Rod Block Setpoints has been deleted since this function has been relocated to the Technical Requirements Manual (see Discussion of Changes for ITS 3.3.2.1). In addition, reference to APRM Scram and RBM Trip Setpoints is deleted since the trip setpoints are an operational detail that is not directly related to the OPERABILITY of the instrumentation. Reference to the Trip setpoints has been eliminated in the referenced Specifications 2.2.1 and 3.3.6 (ITS 3.3.1.1 and ITS 3.3.2.1), therefore, this change is considered administrative.

A.6 CTS 3.4.1.3 and 4.4.1.3 are proposed to be revised in ITS SR 3.4.1.1 to specify jet pump flow mismatch rather than recirculation loop flow mismatch. The flow in the recirculation loop and jet pump loop is proportional, and the measurement of jet pump loop flow versus recirculation loop flow is consistent with the assumptions of the LOCA analysis as cited in UFSAR, Appendix G, Section G.2.2.2. Therefore, the change is considered a presentation preference and is administrative.

A.7 CTS 3.4.1.3 Action a requires restoration of the recirculation loop flow to within the limits if they are not within the limits. The revised presentation of ITS ACTIONS (based on the BWR Standard Technical Specifications, NUREG-1434, Rev. 1) does not explicitly detail options to "restore...to within the specified limit" when an alternate ACTION is provided that allows continued operation. This action is always an option, and is implied in all ACTIONS. Since CTS 3.4.1.3 Action b (ITS 3.4.1 ACTION F) provides an alternate action that allows continued operation, deleting CTS 3.4.1.3 Action a is purely editorial.

A.8 CTS 3.4.1.5.a requires the total core flow to be $\geq$ 45% of rated core flow during forced core circulation operation and CTS 3.4.1.5.b requires THERMAL POWER to be within Region III of CTS Figure 3.4.1.5-1 (ITS Figure 3.4.1-1). Only one of these two requirements has to be met. The CTS 3.4.1.5.a restriction has been deleted since it is sufficient to make reference to be in Region III of ITS Figure 3.4.1-1. Since this change deletes a duplicate requirement, the change is administrative.

ADMINISTRATIVE (continued)

A.9 CTS 3.4.1.5 requires APRM and LRPM noise levels to be within limits when in Region II of Figure 3.4.1.5-1 (ITS Figure 3.4.1-1). Proposed ITS LCO 3.4.1 requires recirculation loop operation within Region III of Figure 3.4.1-1. When recirculation loop operation is within Region II of Figure 3.4.1-1 (CTS Figure 3.4.1.5-1), ITS 3.4.1 ACTION A provides requirements to verify APRM and LPRM noise levels are within limits. Also, CTS 4.4.1.5.2 requires verification that core flow is greater than 39% rated core flow with recirculation loop operation is within Region II of Figure 3.4.1.5-1. 39% rated core flow is the lowest flow limit of Region II. Operation less than 39% rated core flow when in Region II results in entry into Region I of Figure 3.4.1.5-1. As such, ITS 3.4.1 Required Action A.3 requires verification that operation is not within Region I of Figure 3.4.1-1. The proposed changes are only changes in presentation preference and do not alter the intent of the existing requirements. Therefore, the changes are administrative.

A.10 CTS 3.4.1.5 Action a.2.a) requires reducing core thermal power to below 36% rated thermal power by inserting control rods when no recirculation loops are in operation. ITS 3.4.1 Required Action D.2 does not specify inserting of the control rods to reduce thermal power below 36% rated thermal power. When no recirculation loops are in operation, the only acceptable operational method of reducing thermal power to below 36% rated thermal power is by the insertion of the control rods. Therefore, it is not necessary to specify the method of reducing thermal power with no recirculation loops in operation. Since this change does not change the intent of the existing requirements and reduction of power will still be accomplished by the insertion of control rods when no recirculation loops are in operation, this change is administrative.

A.11 Not used.

A.12 CTS 4.4.1.5.1.b requires initiation of the surveillance within 15 minutes and completing the surveillance within the next 30 minutes. The ISTS philosophy is to specify the Completion Time from the discovery of the Condition. Therefore, 45 minutes (15 minutes to initiate plus the next 30 minutes to complete) is proposed to perform the APRM and LPRM noise level verification when the recirculation loops are operating within Region II of CTS Figure 3.4.1.5-1 (ITS Figure 3.4.1-1). Initiation of the surveillance within 15 minutes is discussed in Discussion of Change LA.4. Since this change is a presentation preference and does not alter the intent of the current requirements, this change is administrative.

ADMINISTRATIVE (continued)

A.13 CTS 3.4.1.5, which requires forced circulation to be maintained with flow within the limits of CTS Figure 3.4.1.5-1, is only applicable in Operational Condition 1 (ITS MODE 1). This proposed change will also require forced circulation to be maintained with flow within the limits of ITS Figure 3.4.1-1 (CTS Figure 3.4.1.5-1) in MODE 2. Maintaining flow within the limits of ITS Figure 3.4.1-1 ensures core thermal-hydraulic oscillations do not occur. The region of instability is > 30% Rated Thermal Power. Therefore, it is not operationally possible to be in the region of instability in MODE 2. While the plant is operating in MODE 2, ITS LCO 3.4.1 requires two recirculation loops to be operating in Region III of ITS Figure 3.4.1-1. This proposed change is made for easier readability and interpretation of the Specification. Since this change is a presentation preference and does not alter the intent of the current requirements, this change is administrative.

A.14 CTS 3.4.1.3 Action b requires action to be taken per CTS 3.4.1.1 when a recirculation loop is declared not in operation. The format of the ITS does not include providing "cross references." CTS 3.4.1.1 (ITS 3.4.1) adequately prescribes the necessary conditions for compliance without such references. Therefore, the existing reference to "take the ACTION required by Specification 3.4.1.1" in CTS 3.4.1.3 Action b serves no functional purpose, and its removal is purely an administrative difference in presentation.

TECHNICAL CHANGES - MORE RESTRICTIVE

M.1 Not used.

M.2 CTS 3.4.1.5 Action b.2 requires entry into Region III of Figure 3.4.1.5-1 within 2 hours following attempts to restore APRM and LPRM flux noise levels to within limits in 2 hours. This allows a total of 4 hours (2 hours to restore APRM and LPRM noise levels and 2 hours to enter Region III of Figure 3.4.1.5-1) to operate in Region II with APRM and LPRM flux noise levels outside established limits. ITS 3.4.1 Required Action B.1 allows only 2 hours to satisfy the requirements of the LCO (i.e., enter Region III of ITS Figure 3.4.1-1). The Completion Time of 2 hours is reasonable, based on operating experience, to restore plant parameters in an orderly manner without challenging plant systems. Since this change reduces the time to enter Region III of CTS Figure 3.4.1.5-1 (ITS Figure 3.4.1-1) from 4 to 2 hours, the proposed change is considered to be more restrictive.

TECHNICAL CHANGES - MORE RESTRICTIVE (continued)

M.3 Currently, if CTS 3.4.1.5 Action b.2 cannot be performed, CTS 3.0.3 would require a plant shutdown (i.e., initiate action within 1 hour and be in Startup within the next 6 hours and Hot Shutdown in the following 6 hours). ITS 3.4.1 Required Action E.1 requires the reactor mode switch to be immediately placed in the shutdown position. This action (i.e., place the reactor mode switch in the shutdown position) is consistent with the requirement when no recirculation loops are operating and APRM or LPRM neutron flux noise levels not within limit per CTS 3.4.1.5 Action a.2.b). This change is necessary since neutron flux noise levels outside the associated limit may indicate approaching reactor instability. Therefore, action must be immediately initiated to terminate the potential for a thermal-hydraulic instability event. This proposed change is more restrictive on plant operation.

M.4 CTS 3.4.1.5.b requires the THERMAL POWER to be in Region III of Figure 3.4.1.5-1. However, there is no Surveillance Requirement that verifies this requirement on a periodic basis. ITS SR 3.4.1.2 has been added to verify operation is in Region III of ITS Figure 3.4.1-1 every 24 hours. This will ensure that entry into a region where potential instabilities can occur will not go undetected. Therefore, this change is more restrictive on plant operations.

TECHNICAL CHANGES - LESS RESTRICTIVE

"Generic"

LA.1 The details of CTS 3.4.1.1 Action a.1.a) relating to operational controls during single recirculation loop operation are proposed to be relocated to the UFSAR. Operation of the flow control system in the local manual mode is the normal manner in which flow is controlled when in two loop operation. Thus, the flow control system is normally already in the proper mode for single loop operation; there is no need to place it in the proper mode since it is already in the proper mode. It also is not related to the ability of the system to perform its safety function. Therefore, the relocated details are not required to be in the ITS to provide adequate protection of the public health and safety. Changes to the UFSAR are controlled by the provisions of 10 CFR 50.59.

TECHNICAL CHANGES - LESS RESTRICTIVE (continued)

LA.2 The CTS 3.4.1.5 Action a.1 requirement to "immediately initiate action to" reduce THERMAL POWER to exit Region I when one or both recirculation loops are in operation is relocated to the Bases. In addition, the CTS 3.4.1.5 Action a.2.a) requirement to "immediately" reduce CORE THERMAL POWER to reduce power below 36% of Rated Thermal Power when no recirculation loops are in operation is relocated to the Bases. These relocations to the Bases will be in the form of a discussion that "prompt action should be taken" to exit Region I of CTS Figure 3.4.1.5-1 (ITS Figure 3.4.1-1) or reduce THERMAL POWER to below 36% RTP, as applicable. Immediate action may not always be the conservative method to assure safety. The 2 hour Completion Time of ITS 3.4.1 Required Action C.1 to exit Region I of ITS Figure 3.4.1-1 and the 2 hour Completion Time of ITS 3.4.1 Required Action D.2 to reduce THERMAL POWER to < 36% RTP allows appropriate actions to be evaluated by the operator and completed in a timely manner. Also, the requirements of CTS 3.4.1.5 Actions a.1.a) and a.1.b) that provide the methods of exiting Region I of CTS Figure 3.4.1.5-1 are proposed to be relocated to the Bases. The requirement of ITS Required Action C.1 to exit Region I of ITS Figure 3.4.1-1 is adequate to ensure the region of thermal-hydraulic instability is exited. Therefore, the relocated requirements are not required to be in the ITS to provide adequate protection of the public health and safety. Changes to the Bases will be controlled by the provisions of the proposed Bases Control Program described in Chapter 5 of the ITS.

LA.3 The details on how to restore APRM and LPRM noise levels within limits in CTS 3.4.1.5 Actions b.1 and b.2 are proposed to be relocated to the Bases. Methods to comply with ACTIONS (e.g., how to decrease THERMAL POWER and increase recirculation flow) are more appropriately maintained in the Bases. This is consistent with the philosophy of the BWR Standard Technical Specifications, NUREG-1434, Rev. 1, which is to not be overly prescriptive in the Technical Specifications. ITS 3.4.1 Required Action B.1 which requires restoring APRM and LPRM noise levels to within limits within a limited period of time, is adequate for protection of the public health and safety. In addition, the CTS 3.4.1.5 Action b.1 requirement to "immediately initiate corrective action" to restore APRM and LPRM flux noise levels to within limits is relocated to the Bases in the form of the discussion that "prompt action should be taken" to satisfy the requirements of the LCO. Immediate action may not always be the conservative method to assure safety. The 2 hour Completion Time of ITS 3.4.1 Required Action B.1 allows appropriate actions to be evaluated by the operator and completed in a timely manner. Therefore, the relocated details are not required to be in the ITS to provide adequate protection of the public health and safety. Changes to the Bases will be controlled by the provisions of the proposed Bases Control Program described in Chapter 5 of the ITS.

TECHNICAL CHANGES - LESS RESTRICTIVE (continued)

LA.4 The CTS 4.4.1.5.1.b requirement to "initiate the surveillance within 15 minutes" after entering Region II of CTS Figure 3.4.1.5-1 or completing an increase of at least 5% of RATED THERMAL POWER is relocated to the Bases in the form of the discussion that "prompt action should be taken" to verify APRM and LPRM flux noise levels. When Region II of ITS Figure 3.4.1-1 (CTS Figure 3.4.1.5-1) is entered inadvertently due to a plant transient, operator attention should be focused on stabilizing the plant. As such, specifying a time to initiate the performance of a surveillance may not always be the conservative method to assure safety. The 45 minute Completion Time of ITS 3.4.1 Required Actions A.1 and A.2 allows appropriate actions to be evaluated by the operator and completed in a timely manner. Therefore, the relocated requirement is not required to be in ITS to provide adequate protection of the public health and safety. Changes to the Bases will be controlled by the provisions of the proposed Bases Control Program described in Chapter 5 of the ITS.

LA.5 The details of CTS 4.4.1.5 footnote # concerning which LPRM detectors to monitor (i.e., detector levels A and C of one LPRM string per core octant plus detectors A and C of one LPRM string in the center region of the core should be monitored), are proposed to be relocated to the Bases. These details are not necessary to be included in the Technical Specifications to ensure the Surveillance is adequately performed. ITS 3.4.1 ACTION A will continue to require the LPRM neutron flux noise levels to be monitored. Therefore, the relocated details are not required to be in the ITS to provide adequate protection of the public health and safety. Changes to the Bases will be controlled by the provisions of the proposed Bases Control Program described in Chapter 5 of the ITS.

LA.6 The detail of the actual MCPR correction factor for the MCPR operating limit for single loop operation ("0.01") in CTS 3.4.1.1 Action a.1.c is proposed to be relocated to the COLR. The requirement in proposed LCO 3.4.1 to apply the LCO 3.2.2, "MINIMUM CRITICAL POWER RATIO (MCPR)," single loop operation limits specified in the COLR during operation with one recirculation loop and the requirement in proposed ITS 3.4.1 Action G to satisfy the requirements of the LCO within 12 hours are adequate to ensure the current requirement is performed during single loop operation. Since all the requirements of CTS 3.4.1.1 Action a.1.c (except for the actual limit) are maintained in the proposed Specification, the proposed changes are considered adequate. As such, the relocated details are not required to be in the ITS to provide adequate protection of the public health and safety. Changes to the COLR will be controlled by the provisions of the COLR change control process described in Chapter 5 of the ITS.

TECHNICAL CHANGES - LESS RESTRICTIVE (continued)

"Specific"

L.1 The time to adjust power distribution limits and Reactor Protection System and Control Rod Block instrumentation Allowable Values for single loop operation in CTS 3.4.1.1 Action a.1 is proposed to be increased from 4 hours to 12 hours (proposed Required Action G.1 Completion Time). The increased Completion Time to perform the power distribution limits and instrument adjustments for single recirculation loop operation, considering the time required to secure the necessary resources (e.g., notifying appropriate personnel, obtaining equipment needed to perform the adjustments, and performing appropriate per-job briefing for the RPS and Control Rod Block instrumentation adjustments), is reasonable to avoid unnecessary transients on the plant. The 12 hour Completion Time to adjust the power distribution limits and instrumentation for single recirculation loop operation is considered acceptable based on the low probability of an accident occurring during this period and frequent core monitoring by operations allowing abrupt changes in core flow conditions to be quickly detected. These proposed changes are offset by the benefit of not hastily adjusting the instrumentation for single loop operation which could increase the probability of a plant transient.

L.2 In the event no recirculation loops are in operation, the time required to shutdown in ITS 3.4.1 (Required Action D.3) is 12 hours versus the 6 hour time period allowed by CTS 3.4.1.1 Action b.2 and CTS 3.4.1.5 Action a.2.c). In this degraded condition with no recirculation loops in operation, a Completion Time of 12 hours to be in MODE 3 (Hot Shutdown) provides a reasonable time period to place the unit in a MODE in which the LCO does not apply and is consistent with the Completion Time of similar Technical Specification required plant shutdowns. This change is considered acceptable since in a natural circulation condition, the severity of a DBA is reduced, and there is minimal dependence on the recirculation loop coastdown characteristics. Allowing 12 hours to reach MODE 3 is an acceptable exchange in risk; the risk of a DBA or instability during the additional period to reach MODE 3, versus the potential risk of a unit upset that could challenge safety systems resulting from a rapid plant shutdown.

L.3 CTS 4.4.1.3 requires the recirculation loop flow mismatch to be verified within the limits once per 24 hours when in Operational Condition 1 and 2 during two recirculation loop operation. CTS 4.0.4 requires the Surveillances to be met prior to entry into the applicable Mode or other specified conditions.
CTS 4.4.1.3 cannot be performed prior to its Applicability if shifting from single loop to two loop operation while in MODE 1 or 2. Therefore, a note has been

TECHNICAL CHANGES - LESS RESTRICTIVE

L.3 (cont'd) added (proposed SR 3.4.1.1 Note) providing an allowance for time to initiate the Frequency to avoid intentional entry into the ACTIONS each time the second recirculation pump is started. The time allowed is consistent with the current frequency of the Surveillance (24 hours), and is therefore considered acceptable.

RELOCATED SPECIFICATIONS

None

A.1 ITS 3.4.2

Add proposed LCO 3.4.2, Applicability, and ACTIONS A.2

SURVEILLANCE REQUIREMENTS

4.4.1.1 Each reactor coolant system recirculation loop flow control valve shall be demonstrated OPERABLE at least once per 18 months by: 24 LD.1

SR 3.4.2.1 a. Verifying that the control valve fails "as is" on loss of hydraulic pressure at the hydraulic power units, and

SR 3.4.2.2 b. Verifying that the average rate of control valve movement is:

1. Less than or equal to 11% of stroke per second opening, and

2. Less than or equal to 11% of stroke per second closing.

A.1 ITS 3.4.2

REACTOR COOLANT SYSTEM

Add proposed LCO 3.4.2, Applicability, and ACTIONS A.2

SURVEILLANCE REQUIREMENTS

4.4.1.1 Each reactor coolant system recirculation loop flow control valve shall be demonstrated OPERABLE at least once per ~~18~~ 24 months by: LD.1

SR 3.4.2.1 a. Verifying that the control valve fails "as is" on loss of hydraulic pressure at the hydraulic power units, and

SR 3.4.2.2 b. Verifying that the average rate of control valve movement is:

1. Less than or equal to 11% of stroke per second opening, and

2. Less than or equal to 11% of stroke per second closing.

ADMINISTRATIVE

A.1 In the conversion of the LaSalle 1 and 2 current Technical Specifications (CTS) to the proposed plant specific Improved Technical Specifications (ITS), certain wording preferences or conventions are adopted that do not result in technical changes (either actual or interpretational). Editorial changes, reformatting, and revised numbering are adopted to make the ITS consistent with the BWR Standard Technical Specifications, NUREG-1434, Rev. 1 (i.e., the Improved Standard Technical Specifications (ISTS)).

A.2 CTS 4.4.1.1 provides Surveillance Requirements for the flow control valves (FCVs). Since CTS 4.4.1.1 is part of the Recirculation Loop Technical Specification, CTS 3/4.4.1.1, it is covered by the LCO of CTS 3.4.1.1 and the Applicability of CTS 3.4.1.1. The ITS provides a separate LCO for the FCVs, thus a new LCO statement and Applicability statement are needed. However, since they continue to require FCV OPERABILITY in the same MODES as CTS 3/4.4.1.1, the addition of the new LCO and Applicability are administrative. ITS 3.4.2 ACTION A allows 4 hours to lock up the flow control valve if it is inoperable. This time is consistent with the time in CTS 3.4.1.1 Action a when a loop is not in operation. The actual proposed action (lock up the FCV) is the acceptance criteria to which the FCV is tested by the current Surveillance (CTS 4.4.1.1.a). Thus placing the FCV in this position performs the safety function of the FCV. The proposed change will provide only additional clarification of the current requirements, and is therefore considered administrative.

TECHNICAL CHANGES - MORE RESTRICTIVE

None

TECHNICAL CHANGES - LESS RESTRICTIVE

"Generic"

LD.1 The Frequencies for performing CTS 4.4.1.1.a and 4.4.1.1.b (proposed SRs 3.4.2.1 and 3.4.2.2) have been extended from 18 months to 24 months. These SRs ensure that FCVs fail "as is" on loss of hydraulic pressure at the hydraulic control unit and that the average rate of FCV movement is within the specific limit ($\leq$ 11%/sec). The proposed change will allow these Surveillances to extend their Surveillance Frequency from the current 18 month Surveillance Frequency (i.e., a maximum of 22.5 months accounting for the allowable grace period

TECHNICAL CHANGES - LESS RESTRICTIVE

LD.1 (cont'd) specified in CTS 4.0.2 and proposed SR 3.0.2) to a 24 month Surveillance Frequency (i.e., a maximum of 30 months accounting for the allowable grace period specified in CTS 4.0.2 and proposed SR 3.0.2). This proposed change was evaluated in accordance with the guidance provided in NRC Generic Letter No. 91-04, "Changes in Technical Specification Surveillance Intervals to Accommodate a 24-Month Fuel Cycle," dated April 2, 1991. Reviews of historical maintenance and surveillance data have shown that these tests normally pass their Surveillances at the current Frequency. An evaluation has been performed using this data, and it has been determined that the effect on safety is small.

During normal operation, the FCVs are slowly positioned to obtain the required core flow and power conditions. If an actual or false signal is present requiring a Motion Inhibit (position setpoint demand signal exceed limits, large velocity controller deviation for more than a preset time, and high drywell pressure), the associated FCV should lock up. Therefore, during normal plant operations, the system is utilized and major deviations will not go unnoticed. If any inconsistencies are observed during FCV movement, the flow control system or the hydraulic control units may be taken out of service to perform the required maintenance. After repair, the system may be tested during plant operation to ensure the FCVs function properly.

If a DBA LOCA were to occur, drywell pressure will increase. Drywell pressure sensors will detect this pressurization and immediately drop hydraulic pressure to the pilot lines of check valves on the FCV actuators. With loss of pilot pressure, the check valves will close and lock up the FCV. Until these interlocks are cleared, no control system signal (intentional or inadvertent) can cause FCV position to change. Failure Modes and Effects Analysis have shown that, given a LOCA event, no single failure in the electronic/hydraulic controls can cause the FCV to close while in the normal manual control mode. As a result of these considerations, FCV closure in the unbroken loop is not expected to occur during the LOCA event.

Even if the FCVs were signaled to close for some unlikely reason (LOCA plus two failures: failure of drywell high pressure signal such that FCV lockup does not occur, and failure of FCV controls), backup electronic velocity limiters are included in the recirculation control system to limit FCV velocity to 11%/sec. Additional multiple specific component failures in these limiters must occur to cause full closure of the FCV at velocities in excess of this value. Accordingly, the electronically limited rate of less than or equal to 11% of FCV actuator stroke rate is considered a realistic yet conservative closure rate.

TECHNICAL CHANGES - LESS RESTRICTIVE

LD.1 (cont'd) The velocity limiters are also available to minimize the consequences of the Reactor Recirculation Flow Runout and Flow Control Failure (decreasing flow) transients ensuring the FCVs either open or close at a rate less than or equal to 11%/sec which is assumed in the analysis. In these transients the analysis assumes the FCVs both move at a velocity of ± 11%/sec. The probability of this type of failure is very small since LaSalle normally positions the flow controllers in manual and in this condition the control signal of each loop is independent of each other. Now in the case of transients involving the failure of one FCV, the analysis assumes an FCV moves at a velocity of 30%/sec in the opening direction and 60%/sec in the closing direction. In these transients, the velocity limiters are available to limit the FCV velocity to ± 11%/sec and in addition the hydraulic system is designed to limit the FCV velocity 30%/sec in both the open and close direction, which is within the values assumed in the transient analysis.

Based on the Reactor Recirculation System design and the ability to detect deviations during operation, it is shown that the impact, if any, on system availability is minimal as a result of the change.

The review of historical surveillance data also demonstrated that there are no failures that would invalidate the conclusion that the impact, if any, on system availability is minimal from a change to CTS 4.4.1.1.a and 4.4.1.1.b as implemented in SRs 3.4.2.1 and 3.4.2.2. In addition, the proposed 24-month Surveillance Frequency, if performed at the maximum interval by proposed SR 3.0.2 (30 months), does not invalidate any assumptions in the plant licensing basis.

"Specific"

None

RELOCATED SPECIFICATIONS

None

A.1 ITS 3.4.3

REACTOR COOLANT SYSTEM

JET PUMPS

LIMITING CONDITION FOR OPERATION

LCO 3.4.3 3.4.1.2 All jet pumps shall be OPERABLE.

APPLICABILITY: OPERATIONAL CONDITIONS 1 and 2.

ACTION:

ACTION A With one or more jet pumps inoperable, be in at least HOT SHUTDOWN within 12 hours.

SURVEILLANCE REQUIREMENTS

L.1 Add proposed SR 3.4.3.1 Note 1

SR 3.4.3.1 4.4.1.2.1 Each of the above required jet pumps shall ~~be demonstrated~~ OPERABLE ~~prior to the THERMAL POWER exceeding 25% of RATED THERMAL POWER and~~ at least once per 24 hours by measuring and recording each of the below specified parameters and verifying that no two of the following conditions occur ~~when both recirculation loops are operating with balanced flow~~. LA.1

Add proposed SR 3.4.3.1 Note 2 L.2

a. The indicated recirculation loop drive flow differs by more than 10% from the established flow control valve position ~~loop flow characteristics for two recirculation loop operation~~. A.2

b. The indicated total core flow differs by more than 10% from the established calculated total core flow value ~~derived from either the:~~ LA.1 LA.2

1. ~~Established THERMAL POWER-core flow relationship, or~~

2. ~~Established core plate differential pressure-core flow relationship for two recirculation loop operation.~~

c. The indicated diffuser-to-lower plenum differential pressure of any individual jet pump differs from established two recirculation loop operation patterns by more than ~~10%~~ 20%. L.4 A.3

SR 3.4.3.1 4.4.1.2.2 During single recirculation loop operation, each of the above ~~required~~ jet pumps shall be demonstrated OPERABLE at least once per 24 hours by verifying that no two of the following conditions occur: L.1 A.2

Add proposed SR 3.4.3.1 Note 1

Add proposed SR 3.4.3.1 Note 2 L.2

a. The indicated recirculation loop drive flow in the operating loop differs by more than 10% from the ~~established single~~ recirculation flow control valve position ~~loop flow characteristics~~. LA.1

in the operating loop

b. The indicated total core flow differs by more than 10% from the established calculated total core flow value ~~from single recirculation loop flow measurements derived from either the:~~ LA.2

1. ~~Established THERMAL POWER-core flow relationship, or~~

2. ~~Established core plate differential pressure-core flow relationship for two recirculation loop operation.~~

c. The indicated diffuser-to-lower plenum differential pressure of any individual jet pump differs from established single recirculation loop operational patterns by more than ~~10%~~ 20%. L.4 A.3

L.3

REACTOR COOLANT SYSTEM

JET PUMPS

LIMITING CONDITION FOR OPERATION

LCO 3.4.3 3.4.1.2 All jet pumps shall be OPERABLE.

APPLICABILITY: OPERATIONAL CONDITIONS 1 and 2.

ACTION:

ACTION A With one or more jet pumps inoperable, be in at least HOT SHUTDOWN within 12 hours.

SURVEILLANCE REQUIREMENTS

L.1 add proposed NOTE 1 to SR 3.4.3.1

SR 3.4.3.1 4.4.1.2.1 Each of the above required jet pumps shall be demonstrated OPERABLE ~~prior to THERMAL POWER exceeding 25% of RATED THERMAL POWER and~~ at least once per 24 hours by measuring and recording each of the below specified parameters and verifying that no two of the following conditions occur ~~when both recirculation loops are operating with balanced flow.~~ LA.1 add proposed NOTE 2 to SR 3.4.3.1 L.2

drive A.2

a. The indicated recirculation loop drive flow differs by more than 10% from the established flow control valve position ~~-loop flow characteristics for two recirculation loop operation.~~ LA.1

b. The indicated total core flow differs by more than 10% from the established calculated total core flow value ~~derived from either the:~~ A.3

~~1. Established THERMAL POWER-core flow relationship, or~~

~~2. Established core plate differential pressure-core flow relationship for two recirculation loop operation.~~ LA.2 L.4

c. The indicated diffuser-to-lower plenum differential pressure of any individual jet pump differs from established two recirculation loop operation patterns by more than ~~10%~~ 20%. L.1 add proposed NOTE 1 to SR 3.4.3.1

SR 3.4.3.1 4.4.1.2.2 During single recirculation loop operation, each of the above ~~required~~ in the operating loop jet pumps shall be demonstrated OPERABLE at least once per 24 hours by verifying that no two of the following conditions occur: add proposed NOTE 2 to SR 3.4.3.1 L.2

drive A.2

a. The indicated recirculation loop drive flow in the operating loop differs by more than 10% from the established single recirculation flow control valve position ~~-loop flow characteristics.~~ LA.1

b. The indicated total core flow differs by more than 10% from the established calculated total core flow value ~~from single recirculation loop flow measurements derived from either the:~~ A.3

~~1. Established THERMAL POWER-core flow relationship, or~~

~~2. Established core plate differential pressure-core flow relationship for two recirculation loop operation.~~ LA.2

c. The indicated diffuser-to-lower plenum differential pressure of any individual jet pump differs from established single recirculation loop by more than ~~10%~~ 20%. L.4

L.3

ADMINISTRATIVE

A.1 In the conversion of the LaSalle 1 and 2 current Technical Specifications (CTS) to the proposed plant specific Improved Technical Specifications (ITS), certain wording preferences or conventions are adopted that do not result in technical changes (either actual or interpretational). Editorial changes, reformatting, and revised numbering are adopted to make the ITS consistent with the BWR Standard Technical Specifications, NUREG-1434, Rev. 1 (i.e., the Improved Standard Technical Specifications (ISTS)).

A.2 The term “loop flow” in CTS 4.4.1.2.1.a and CTS 4.4.1.2.2.a has been modified in ITS (SR 3.4.3.1.a) to specify “loop drive flow” to be consistent with NUREG-1434. This change only clarifies the intent and does not alter the meaning or intent of the Surveillance Requirement since recirculation loop flow is recirculation loop drive flow. Because this change is an enhanced presentation of existing intent, the change is considered administrative.

A.3 CTS 4.4.1.2.1.b and CTS 4.4.1.2.2.b compare the indicated total core flow with an established total core flow value. This value is derived by two methods specified in CTS 4.4.1.2.1.b.1 and b.2; and CTS 4.4.1.2.2.b.1 and b.2. ITS relocates these methods of calculating the total core flow value to the ITS Bases (See Discussion of Change LA.2). As such, the word “calculated” is being added to CTS 4.4.1.2.1.b and CTS 4.4.1.2.2.b (ITS SR 3.4.3.1.b) to differentiate between the indicated total core flow and the calculated total core flow. Since this change is only an enhanced presentation of existing intent, this change is considered administrative.

TECHNICAL CHANGES - MORE RESTRICTIVE

None

TECHNICAL CHANGES - LESS RESTRICTIVE

"Generic"

LA.1 CTS 4.4.1.2.1, 4.4.1.2.1.a, and 4.4.1.2.2.a provide specific details of the loop flow characteristics required when determining established flow control valve position. Details of the methods for determining established flow control valve position (determining the loop flow characteristics of the flow control valve for two recirculation loop operation versus single recirculation loop operation and ensuring balanced recirculation loop flows when both recirculation loops are

TECHNICAL CHANGES - LESS RESTRICTIVE

LA.1 (cont'd) operating) are to be relocated to the Bases. The requirements of ITS SR 3.4.3.1 are adequate to determine jet pump OPERABILITY. As a result, the relocated details do not need to be included in the Technical Specifications to provide adequate protection of the public health and safety. Changes to the Bases will be controlled by the provisions of the Bases Control Program described in Chapter 5 of the ITS.

LA.2 CTS 4.4.1.2.1.b and 4.4.1.2.2.b, which provide methods of deriving total core flow values, are to be relocated to the Bases. These details are not necessary to ensure that the jet pumps are maintained OPERABLE. The requirements of ITS 3.4.3 and SR 3.4.3.1.b are adequate to ensure the total core flow is compared to the proper established patterns and the jet pumps are maintained OPERABLE. Therefore, the relocated detail is not required to be in the ITS to provide adequate protection of the public health and safety. Changes to the Bases will be controlled by the provisions of the proposed Bases Control Program described in Chapter 5 of the ITS.

"Specific"

L.1 CTS 4.4.1.2.1 requires the jet pump surveillance to be performed on the required jet pumps prior to THERMAL POWER exceeding 25% RATED THERMAL POWER and at least once per 24 hours. CTS 4.4.1.2.2 requires the jet pump surveillance to be performed on the required jet pumps once per 24 hours. This change adds a Note to CTS 4.4.1.2.1 and CTS 4.4.1.2.2 (proposed SR 3.4.3.1 Note 1) to allow a 4-hour delay in performance of the Surveillance after the associated recirculation loop is restored to operation. The Note allows 4 hours to perform the Surveillance after the associated recirculation loop is in operation, because these checks can only be performed during jet pump operation (i.e., when the loop is in operation). The four hours is considered an acceptable time to establish conditions appropriate for data collection and evaluation.

L.2 CTS 4.4.1.2.1 requires the jet pump surveillance to be performed on the required jet pumps prior to THERMAL POWER exceeding 25% RATED THERMAL POWER and at least once per 24 hours. CTS 4.4.1.2.2 requires the jet pump surveillance to be performed on the required jet pumps at least once per 24 hours. During low jet pump flow conditions, which occur when THERMAL POWER is < 25% RTP, jet pump noise approaches the threshold response of the associated instrumentation and precludes the collection of repeatable and meaningful data. The requirements of ITS 3.4.3, the associated Surveillance Requirement, and the requirement of SR 3.0.1 (SRs shall be met during the

TECHNICAL CHANGES - LESS RESTRICTIVE

L.2 (cont'd) MODES or other specified conditions in the Applicability for individual LCOs) are adequate for maintaining the jet pumps OPERABLE. Since the data collected during performance of the surveillance < 25% RTP is not meaningful, it is unnecessary to perform the jet pump surveillance during these low flow conditions. In addition, the most common outcome of the performance of Surveillances is the successful demonstration the acceptance criteria are satisfied and OPERABILITY verified. Delaying the performance of this Surveillance until appropriate conditions are established is considered acceptable. As a result, a Note is added to CTS 4.4.1.2.1 and CTS 4.4.1.2.2 (ITS SR 3.4.3.1 Note 2) that allows 24 hours to perform the surveillance after THERMAL POWER exceeds 25% RTP. The 24 hours is considered an acceptable time to establish conditions appropriate for data collection and evaluation.

L.3 CTS 4.4.1.2.2 requires that jet pump operability be demonstrated for each of the "above required" pumps. At LaSalle, the phrase "above required" pumps refers to those in associated LCO 3.4.1.2, which states that "All jet pumps shall be OPERABLE." In the past this CTS requirement has resulted in evaluating the operability of jet pumps that are not in operation, while in single loop operation. This assessment was performed based on the expected flow indication (reversed differential pressures). The proposed ITS SR 3.4.3.1 eliminates the requirement to assess operability of jet pumps that are not in operation, limiting testing to those jet pumps in the operating loop, when in single loop operation. The reverse direction and low flow rates through the jet pumps that are not in operation do not provide meaningful indication of the operability of the jet pumps that are not in operation. This change is consistent with NUREG-1434.

L.4 CTS 4.4.1.2.1.c and 4.4.1.2.2.c require the indicated diffuser-to-lower plenum differential pressure of any individual jet pump to not differ from established patterns by more than 10%. The proposed change adjusts the surveillance acceptance criteria from 10% to 20% for individual jet pump diffuser-to-lower plenum differential pressure variations from the established pattern. This acceptance criteria is located in the Surveillance that verifies the OPERABILITY of the jet pumps. This change corrects an error in Technical Specifications. This change is consistent with the recommendations of SIL-330 (GE Service Information Letter number 330) and NUREG/CR-3052 (Closeout of IE Bulletin 80-07: BWR Jet Pump Assembly Failure). SIL-330 specifies a 10% criteria for individual jet pump flow distribution. When measured by jet pump diffuser-to-lower plenum differential pressure, the equivalent limit is 20% because of the relationship between flow and delta-P. Since LaSalle Units 1 and 2 utilize the diffuser-to-lower plenum differential pressure measurement, the variance allowed

TECHNICAL CHANGES - LESS RESTRICTIVE

L.4 (cont'd) should have been 20% as was recommended in SIL-330 and NUREG/CR-3052. Since the value is being changed from 10% to 20%, it is considered a relaxation from existing requirements although the change corrects an error. Therefore, this change constitutes a less restrictive change. This change is consistent with BWR ISTS, NUREG-1434, Rev. 1.

RELOCATED SPECIFICATIONS

None

A.1 ITS 3.4.4

REACTOR COOLANT SYSTEM

3/4.4.2 SAFETY/RELIEF VALVES

LIMITING CONDITION FOR OPERATION A.2

LCO 3.4.4 3.4.2 The safety valve function of 17 of the below listed 18 reactor coolant system safety/relief valves shall be OPERABLE with the specified code safety valve function lift setting*#; ~~all installed valves shall be closed~~ with OPERABLE position indication. L.1

Moved to ITS 3.3.3.1

SR 3.4.4.1
a. 4 safety/relief valves @1205 psig ±3%
b. 4 safety/relief valves @1195 psig ±3%
c. 4 safety/relief valves @1185 psig ±3%
d. 4 safety/relief valves @1175 psig ±3%
e. 2 safety/relief valves @1150 psig ±3%

A.2

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2, and 3.

ACTION:

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

b. With one or more of the above required safety/relief valve stem position indicators inoperable, restore the inoperable stem position indicators 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. A.2

SURVEILLANCE REQUIREMENTS

4.4.2.1 The safety/relief valve stem position indicators of each safety/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.**

Moved to ITS 3.3.3.1

A.2

4.4.2.2 ~~The low-low set function shall be demonstrated not to interfere with the OPERABILITY of the safety/relief valves~~ or the ADS by performance of a CHANNEL CALIBRATION at least once per 18 months. L.2 A.3

Moved to ITS 3.3.5.1 LA.1

*The lift ~~setting pressure~~ shall correspond to ~~ambient conditions of the valves at nominal operating temperatures and pressures.~~ Following testing, lift settings shall be within ±1%. A.4 L.3

SR 3.4.4.1

Note to SR 3.4.4.1
#Up to two inoperable valves may be replaced with spare OPERABLE valves with lower setpoints until ~~the next refueling outage.~~

SR 3.4.4.1
**The provisions of Specification 4.0.4 are not applicable provided the surveillance is performed within 12 hours after reactor steam pressure is adequate to perform the test. A.2

Moved to ITS 3.3.3.1

Page 1 of 2

A.1

REACTOR COOLANT SYSTEM

3/4.4.2 SAFETY/RELIEF VALVES

LIMITING CONDITION FOR OPERATION

LCO 3.4.4 3.4.2 The safety valve function of 12 of the below listed 13 reactor coolant system safety/relief valves shall be OPERABLE with the specified code safety valve function lift setting*#; all installed valves shall be closed with OPERABLE position indication. L.1 A.2 Moved to ITS 3.3.3.1

a. 2 safety/relief valves @1205 psig ±3%
b. 3 safety/relief valves @1195 psig ±3%
c. 2 safety/relief valves @1185 psig ±3%
d. 4 safety/relief valves @1175 psig ±3%
e. 2 safety/relief valves @1150 psig ±3%

L.1

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2, and 3.

ACTION:

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

b. With one or more of the above required safety/relief valve stem position indicators inoperable, restore the inoperable stem position indicators 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. A.2

SURVEILLANCE REQUIREMENTS

4.4.2.1 ~~The safety/relief~~ valve stem position indicators of each safety/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.**

Moved to ITS 3.3.3.1 A.2

4.4.2.2 The low low set function shall be demonstrated not to interfere with the OPERABILITY of the safety/relief valves or the ADS by performance of a CHANNEL CALIBRATION at least once per 18 months. L.2 A.3 Moved to ITS 3.3.5.1

SR 3.4.4.1 *The lift setting pressure shall correspond to ambient conditions of the valves at nominal operating temperatures and pressures. Following testing, lift settings shall be within ±1%. LA.1 A.4

Note to SR 3.4.4.1 #Up to two inoperable valves may be replaced with spare OPERABLE valves with lower setpoints until the next refueling outage. L.3

**The provisions of Specification 4.0.4 are not applicable provided the surveillance is performed within 12 hours after reactor steam pressure is adequate to perform the test. A.2 Moved to ITS 3.3.3.1

ADMINISTRATIVE

A.1 In the conversion of the LaSalle 1 and 2 current Technical Specifications (CTS) to the proposed plant specific Improved Technical Specifications (ITS), certain wording preferences or conventions are adopted that do not result in technical changes (either actual or interpretational). Editorial changes, reformatting, and revised numbering are adopted to make the ITS consistent with the BWR Standard Technical Specifications, NUREG-1434, Rev. 1 (i.e., the Improved Standard Technical Specifications (ISTS)).

A.2 The CTS 3.4.2, CTS 3.4.2 Action b, and CTS 4.4.2.1 requirements associated with safety valve position indication are moved to Section 3.3 of the ITS in accordance with the format of the ISTS. any technical changes to these requirements are addressed in the Discussion of Changes for ITS 3.3.3.1.

A.3 The requirements associated with the ADS function instrumentation in CTS 4.4.2.2 do not verify the ability of the S/RVs to perform their related safety function of limiting vessel pressure less than the vessel pressure safety limit during an transient. Therefore, the requirements associated with the ADS function instrumentation are moved to Section 3.3 of the ITS. Any technical changes to these requirements are addressed in the Discussion of Changes for ITS 3.3.5.1.

A.4 CTS 3/4.4.2 does not have a specific Surveillance that requires the S/RV lift setpoints to be periodically verified. Proposed SR 3.4.4.1 has been added to verify the proper lift setpoints of the required S/RVs are within limits in accordance with the Inservice Testing Program. Since CTS 4.0.5 currently requires this type of testing, this addition is considered administrative.

TECHNICAL CHANGES - MORE RESTRICTIVE

None

TECHNICAL CHANGES - LESS RESTRICTIVE

"Generic"

LA.1 The details of CTS 3.4.2 footnote *, relating to lift setting pressure of the safety/relief valves (the lift setting pressure shall correspond to ambient conditions of the valves at nominal operating temperatures and pressures), are proposed to be relocated to the Bases. The requirements of proposed SR 3.4.4.1 are adequate to ensure safety/relief valve lift setpoints are within required

TECHNICAL CHANGES - LESS RESTRICTIVE

LA.1 (cont'd) settings. As a result, the details relocated to the Bases are not necessary for ensuring safety/relief valve setpoints are maintained within required settings and do not need to be in the ITS to provide adequate protection of the public health and safety. Changes to the Bases will be controlled by the provisions of the proposed Bases Control Program described in Chapter 5 of the ITS.

"Specific"

L.1 CTS 3.4.2 requires all installed S/RVs to be closed when in Operational Conditions 1, 2, and 3. The Bases of CTS 3/4.4.2 states that the safety function of the S/RVs is to operate to prevent the Reactor Coolant System from being pressurized above the pressure Safety Limit. The requirement for the S/RVs to be closed in CTS 3.4.2 is not required to meet that safety function. The Bases of CTS 3/4.4.2 also states that the S/RVs are required to be closed to ensure that the integrity of the primary coolant boundary is known to exist. However, there are other Technical Specification requirements that provide adequate assurance that the primary coolant boundary is maintained and that the S/RVs remain closed in Operational Conditions 1, 2, and 3 (ITS MODES 1, 2, and 3). The Reactor Coolant System Operational LEAKAGE and Reactor Coolant System Leakage Detection Instrumentation Technical Specifications specifically provide requirements to detect primary coolant boundary leakage. In addition, the requirements of the Suppression Pool Average Temperature and Suppression Pool Level Technical Specifications provide requirements that would detect an open or leaking S/RV and Emergency Operating Procedures provide appropriate operator actions to mitigate the effects of an open S/RV on the reactor pressure vessel and the primary containment. Therefore, the CTS 3.4.2 requirement that all installed S/RVs be closed is unnecessary and is deleted.

L.2 CTS 4.4.2.2, in part, verifies that the low-low set function does not interfere with the OPERABILITY of the S/RVs by a CHANNEL CALIBRATION. Any failure associated with the low-low set instrumentation logic does not impact the ability of the S/RVs to mechanically open on overpressure. As a result, the low-low set function cannot impact the safety function of the S/RVs since the safety function only requires the S/RV to mechanically open to mitigate a vessel overpressurization transient. Therefore, the requirement of CTS 4.4.2.2, which verifies that the low-low set function does not interfere with the OPERABILITY of the S/RVs by a CHANNEL CALIBRATION, is not necessary to ensure the OPERABILITY of the safety function of the S/RVs and is deleted.

TECHNICAL CHANGES - LESS RESTRICTIVE (continued)

L.3 The # Note to CTS 3.4.2 allows up to two S/RVs to be replaced with two spare OPERABLE S/RVs with lower setpoint values until the next refueling outage. This Note is modified in ITS (Note to SR 3.4.4.1) to allow one or two required S/RVs to be replaced with S/RVs with lower setpoint values indefinitely. this change is considered acceptable since the safety function of the S/RVs to mitigate a vessel overpressurization transient can still be performed. This change, which allows two required S/RVs to have lower setpoint values indefinitely, does not impact the assumptions of the overpressure protection analysis or the containment loading analysis. This change is consistent with proposed ISTS Generic Change TSTF-298.

RELOCATED SPECIFICATIONS

None

REACTOR COOLANT SYSTEM

OPERATIONAL LEAKAGE

LIMITING CONDITION FOR OPERATION

3.4.3.2 Reactor coolant system leakage shall be limited to:

a. No PRESSURE BOUNDARY LEAKAGE.

b. 5 gpm UNIDENTIFIED LEAKAGE.

c. 25 gpm total leakage averaged over any the previous A.2 24 hour period.

A.3 moved to LCO 3.4.6

d. 1 gpm leakage at a reactor coolant system pressure at 1000 ± 50 psig from any reactor coolant system pressure isolation valve specified in Table 3.4.3.2-1.

e. 2 gpm increase in UNIDENTIFIED LEAKAGE within any the previous A.2 24 hour period. in Mode 1 L.2

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2 and 3.

ACTION:

ACTION C a. With any PRESSURE BOUNDARY LEAKAGE, be in at least HOT SHUTDOWN within 12 hours and in COLD SHUTDOWN within the next 24 hours.

ACTION A b. With any reactor coolant system leakage greater than the limits in b and/or c, above, reduce the leakage rate to within the limits within 4 hours or be in at least HOT SHUTDOWN within the next 12 hours and

ACTION C in COLD SHUTDOWN within the following 24 hours.

A.3 Moved to LCO 3.4.6

c. With any reactor coolant system pressure isolation valve leakage greater than the above limit, isolate the high pressure portion of the affected system from the low pressure portion within 4 hours by use of at least two closed valves, or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

d. With one or more high/low pressure interface valve leakage pressure monitors inoperable, restore the inoperable monitor(s) to OPERABLE status within 7 days or verify the pressure to be less than the alarm setpoint at least once per 12 hours by local indication; restore the inoperable monitor(s) 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 12 hours.

ACTION B e. With any reactor coolant system leakage greater than the limit in e above, identify the source of leakage within 4 hours or be in at

ACTION C least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

is not IGSCC susceptible material M.1

or reduce leakage to within limit A.4

A.1

REACTOR COOLANT SYSTEM

SURVEILLANCE REQUIREMENTS

SR 3.4.5.1 4.4.3.2.1 The reactor coolant system leakage shall be demonstrated to be within each of the above limits ~~on average~~ once per ~~8 hours not to exceed~~ 12 hours.* L.1

4.4.3.2.2 Each reactor coolant system pressure isolation valve specified in Table 3.4.3.2-1 shall be demonstrated OPERABLE:

a. Pursuant to Specification 4.0.5, except that in lieu of any leakage testing required by Specification 4.0.5, each valve shall be demonstrated OPERABLE by verifying leakage to be within its limit:

1. At least once per 18 months, and

2. Prior to returning the valve to service following maintenance, repair or replacement work on the valve which could affect its leakage rate.

The provisions of Specification 4.0.4 are not applicable for entry into OPERATIONAL CONDITION 3.

b. By demonstrating OPERABILITY of the high/low pressure interface valve leakage pressure monitors by performance of a:

1. CHANNEL FUNCTIONAL TEST at least once per 31 days, and

2. CHANNEL CALIBRATION at least once per 18 months,

With the alarm setpoint for the:

1. HPCS system ≤ 100 psig.

2. LPCS system ≤ 500 psig.

3. LPCI/shutdown cooling system ≤ 400 psig. A.3

4. RHR shutdown cooling ≤ 190 psig.

5. RCIC ≤ 90 psig.

<moved to LCO 3.4.6>

~~*Technical Specification 4.0.2 does not apply.~~ L.1

REACTOR COOLANT SYSTEM

OPERATIONAL LEAKAGE A.1 ITS 3.4.5

LIMITING CONDITION FOR OPERATION

3.4.3.2 Reactor coolant system leakage shall be limited to:

LCO 3.4.5

a. No PRESSURE BOUNDARY LEAKAGE.

b. 5 gpm UNIDENTIFIED LEAKAGE.

c. 25 gpm total leakage averaged over any the previous A.2 24 hour period.

d. 1 gpm leakage at a reactor coolant system pressure at 1000 ± 50 psig from any reactor coolant system pressure isolation valve specified in Table 3.4.3.2-1. A.3 <Moved to LCO 3.4.6>

e. 2 gpm increase in UNIDENTIFIED LEAKAGE within any the previous A.2 24 hour period. in Mode 1 L.2

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2 and 3.

ACTION:

ACTION C a. With any PRESSURE BOUNDARY LEAKAGE, be in at least HOT SHUTDOWN within 12 hours and in COLD SHUTDOWN within the next 24 hours.

ACTION A b. With any reactor coolant system leakage greater than the limits in b and/or c, above, reduce the leakage rate to within the limits within 4 hours or be in at least HOT SHUTDOWN within the next 12 hours and
ACTION C in COLD SHUTDOWN within the following 24 hours.

c. With any reactor coolant system pressure isolation valve leakage greater than the above limits, isolate the high pressure portion of the affected system from the low pressure portion within 4 hours by use of at least two closed valves, or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours. A.3 <Moved to LCO 3.4.6>

d. With one or more high/low pressure interface valve leakage pressure monitors inoperable, restore the inoperable monitor(s) to OPERABLE status within 7 days or verify the pressure to be less than the alarm setpoint at least once per 12 hours by local indication; restore the inoperable monitor(s) 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 12 hours.

ACTION B e. With any reactor coolant system leakage greater than the limit in e, above, identify the source of leakage within 4 hours or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN
ACTION C within the following 24 hours.

is not IGSCC susceptible material M.1

or reduce leakage to within limit A.4

 A.1 ITS 3.4.5

SURVEILLANCE REQUIREMENTS

SR3.4.5.1 4.4.3.2.1 The reactor coolant system leakage shall ~~be demonstrated to be~~ within each of the above limits ~~on average~~ once per ~~8 hours not to exceed~~ 12 hours.* L.1

4.4.3.2.2 Each reactor coolant system pressure isolation valve specified in Table 3.4.3.2-1 shall be demonstrated OPERABLE:

a. Pursuant to Specification 4.0.5, except that in lieu of any leakage testing required by Specification 4.0.5, each valve shall be demonstrated OPERABLE by verifying leakage to be within its limit:

1. At least once per 18 months, and

2. Prior to returning the valve to service following maintenance, repair or replacement work on the valve which could affect its leakage rate.

The provisions of Specification 4.0.4 are not applicable for entry into OPERATIONAL CONDITION 3.

b. By demonstrating OPERABILITY of the high/low pressure interface valve leakage pressure monitors by performance of a:

1. CHANNEL FUNCTIONAL TEST at least once per 31 days, and

2. CHANNEL CALIBRATION at least once per 18 months,

With the alarm setpoint for the:

1. HPCS system ≤ 100 psig.

2. LPCS system ≤ 500 psig.

3. LPCI/shutdown cooling system ≤ 400 psig.

4. RHR shutdown cooling ≤ 190 psig.

5. RCIC ≤ 90 psig.

A.3

<Moved to LCO 3.4.6>

~~*Technical Specification 4.0.2 does not apply.~~ L.1

ADMINISTRATIVE

A.1 In the conversion of the LaSalle 1 and 2 current Technical Specifications (CTS) to the proposed plant specific Improved Technical Specifications (ITS), certain wording preferences or conventions are adopted that do not result in technical changes (either actual or interpretational). Editorial changes, reformatting, and revised numbering are adopted to make the ITS consistent with the BWR Standard Technical Specifications, NUREG-1434, Rev. 1 (i.e., the Improved Standard Technical Specifications (ISTS)).

A.2 In CTS 3.4.3.2.c and CTS 3.4.3.2.e, the LEAKAGE limits apply at any moment, to the previous 24 hours (not any future or past 24 hour period). This results in a "rolling average" covering "any 24-hour period." Therefore, changing "any 24-hour period" to "the previous 24-hour period" in ITS 3.4.4.c and 3.4.4.d does not change the intent of the requirement. This change is editorial, and as such, is considered administrative only.

A.3 CTS 3.4.3.2.d, 3.4.3.2 Action c, 3.4.3.2 Action d, and 4.4.3.2.2 are being moved to ITS 3.4.6 in accordance with the format of the BWR Standard Technical Specifications, NUREG-1434, Rev. 1. Any technical changes to this requirement will be addressed in the Discussion of Changes for ITS: 3.4.6.

A.4 CTS 4.4.3.2 Action e requires identification of the source of the leakage within 4 hours if the limit of CTS 3.4.3.2.e is not met. ITS 3.4.5 Required Action B.1 has been added to provide an option to reduce the leakage to within the limit in lieu of identifying the source. This change is considered administrative since restoring compliance with the LCO is always an option (per CTS 3.0.2), whether or not it is specifically stated in the Actions.

TECHNICAL CHANGES - MORE RESTRICTIVE

M.1 CTS 3.4.3.2 Action e requires, with unidentified LEAKAGE increase greater than the limit, that the source of leakage be identified within 4 hours. ITS 3.4.5 Required Action B.2 requires the source of leakage to be identified within 4 hours and that it not be from intergranular stress corrosion cracking (IGSCC) susceptible material. The limit on unidentified LEAKAGE increase is provided to address the failure mechanism associated with IGSCC. Therefore, to allow continued plant operation, it is necessary to also identify that the source of unidentified LEAKAGE increase is not from IGSCC susceptible material. This change represents an additional restriction on plant operation necessary to ensure that degradation of IGSCC susceptible material is identified and mitigated.

TECHNICAL CHANGES - LESS RESTRICTIVE

"Generic"

None

"Specific"

L.1 CTS 4.4.3.2.1 requires the RCS leakage (RCS unidentified and total LEAKAGE and unidentified LEAKAGE increase) to be determined on average once per 8 hours, not to exceed 12 hours. The Surveillance Frequency for CTS 4.4.3.2.1 has been changed to 12 hours in ITS SR 3.4.5.1. This change essentially allows the 25% extension specified in CTS 4.0.2 (proposed SR 3.0.2) to be applied to the current 12 hour surveillance interval. As such, the maximum interval has been extended from 12 hours to 15 hours. NRC guidance is provided in Generic Letter 88-01, Supplement 1, allowing a Surveillance Frequency of once per shift not to exceed 12 hours. The proposed Surveillance Frequency is consistent with the NRC guidance since the normal Frequency remains equivalent to a 12 hour shift. The proposed 3 hour extension to the surveillance interval is considered acceptable since the probability of a pipe break occurring in the primary containment during the extension period is small and the vast majority of the surveillances are completed with no indication of excessive RCS operational LEAKAGE. Furthermore, the leak detection instrumentation will remain available during the extension period such that excessive RCS operational LEAKAGE will continue to be alarmed in the main control room and a change in sump flow will continue to be indicated on the control room leak rate recorder.

L.2 The unidentified LEAKAGE rate increase limit is proposed to be applicable only in MODE 1, instead of the current MODES 1, 2, and 3. As a plant starts up and increases pressure, leakage will occur due to the increased pressure. Thus, an increase is detected, and if greater than the limit, could require a unit shutdown, even though there is no safety problem. This proposed change will not require the limit to be applied until MODE 1 is achieved, which is when reactor pressure has effectively stabilized at normal operating pressure. The overall 5 gpm unidentified LEAKAGE limit will still be maintained. This limit is much below the expected flow from a critical crack in the primary system. This change is consistent with the latest licensed BWRs as well as the BWR ISTS, NUREG-1434, Rev. 1.

RELOCATED SPECIFICATIONS

None

A.1

ITS 3.4.6

REACTOR COOLANT SYSTEM

OPERATIONAL LEAKAGE

LIMITING CONDITION FOR OPERATION

0.5 gpm leakage per nominal inch of valve size up to a maximum leakage of 5 gpm for each PIV

LCO 3.4.6 3.4.3.2 Reactor coolant system leakage shall be limited to:

a. No PRESSURE BOUNDARY LEAKAGE.

b. 5 gpm UNIDENTIFIED LEAKAGE.

c. 25 gpm total leakage averaged over any 24 hour period.

L.4

SR 3.4.6.1, LCO 3.4.6 — d. ~~1 gpm leakage~~ at a reactor coolant system pressure at 1000 ± 50 psig from any reactor coolant system pressure isolation valve ~~specified in Table 3.4.3.2-1~~.

LA.1

e. 2 gpm increase in UNIDENTIFIED LEAKAGE within any 24 hour period.

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2 and 3~~*~~. add proposed MODE 3 RHR allowance L.1

ACTION: add proposed ACTIONS NOTES 1 and 2 A.2

a. With any PRESSURE BOUNDARY LEAKAGE, be in at least HOT SHUTDOWN within 12 hours and in COLD SHUTDOWN within the next 24 hours.

b. With any reactor coolant system leakage greater than the limits in b and/or c, above, reduce the leakage rate to within the limits within 4 hours or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

ACTION A — c. With any reactor coolant system pressure isolation valve leakage greater than the above limit, isolate the high pressure portion of the affected system ~~from the low pressure portion~~ within 4 hours by use of at least two closed valves, ~~or be~~ in at least HOT SHUTDOWN L.2

ACTION B — within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

~~d. With one or more high/low pressure interface valve leakage pressure monitors inoperable, restore the inoperable monitor(s) to OPERABLE status within 7 days or verify the pressure to be less than the alarm setpoint at least once per 12 hours by local indication; restore the inoperable monitor(s) 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 12 hours.~~ M.1 LC.1

e. With any reactor coolant system leakage greater than the limit in e above, identify the source of leakage within 4 hours or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

add Required Actions A.1 and A.2 Note

See ITS 3.4.5

REACTOR COOLANT SYSTEM

<See ITS 3.4.5>

SURVEILLANCE REQUIREMENTS

4.4.3.2.1 The reactor coolant system leakage shall be demonstrated to be within each of the above limits on average once per 8 hours not to exceed 12 hours.*

SR 3.4.6.1 4.4.3.2.2 Each reactor coolant system pressure isolation valve ~~specified in Table 3.4.3.2-1~~ shall be demonstrated OPERABLE: LA.1

a. Pursuant to Specification 4.0.5, except that in lieu of any leakage testing required by Specification 4.0.5, each valve shall be demonstrated OPERABLE by verifying leakage to be within its limit:

1. ~~At least once per 18 months, and~~ LA.2

2. ~~Prior to returning the valve to service following maintenance, repair or replacement work on the valve which could affect its leakage rate.~~ L.3

NOTE to SR 3.4.6.1 The provisions of Specification 4.0.4 are not applicable for entry into OPERATIONAL CONDITION 3.

~~b. By demonstrating OPERABILITY of the high/low pressure interface valve leakage pressure monitors by performance of a:~~

~~1. CHANNEL FUNCTIONAL TEST at least once per 31 days, and~~

~~2. CHANNEL CALIBRATION at least once per 18 months,~~

~~With the alarm setpoint for the:~~

~~1. HPCS system ≤ 100 psig.~~

~~2. LPCS system ≤ 500 psig.~~

~~3. LPCI/shutdown cooling system ≤ 400 psig.~~

~~4. RHR shutdown cooling ≤ 190 psig.~~

~~5. RCIC ≤ 90 psig.~~

LC.1

*Technical Specification 4.0.2 does not apply. <See ITS 3.4.5>

TABLE 3.4.3.2-1

REACTOR COOLANT SYSTEM PRESSURE ISOLATION VALVES

LA.1

SYSTEM	VALVE NUMBER	FUNCTION
a. LPCS	E21-F006	LPCS Injection
	E-21-F005*	LPCS Injection
b. HPCS	E22-F005	HPCS Injection
	E22-F004	HPCS Injection
c. RHR	E12-F041A	LPCI Injection
	E12-F041B	LPCI Injection
	E12-F041C	LPCI Injection
	E12-F042A	LPCI Injection
	E12-F042B*	LPCI Injection
	E12-F042C*	LPCI Injection
	E12-F050A	Shutdown Cooling Return
	E12-F050B	Shutdown Cooling Return
	E12-F053A	Shutdown Cooling Return
	E12-F053B*	Shutdown Cooling Return
	E12-F009	Shutdown Cooling Suction
	E12-F008	Shutdown Cooling Suction
d. RDIC	E51-F066	RCIC Head Spray
	E51-F065	RCIC Head Spray

A.3

*The specified 18 month interval may be waived for Cycle 1 provided the surveillance is performed during Refuel 1.

REACTOR COOLANT SYSTEM

OPERATIONAL LEAKAGE

A.1

LIMITING CONDITION FOR OPERATION

L.4

0.5 gpm leakage per nominal inch of valve size up to a maximum leakage of 5 gpm for each PIV

LCO 3.4.6 3.4.3.2 Reactor coolant system leakage shall be limited to:

a. No PRESSURE BOUNDARY LEAKAGE.

b. 5 gpm UNIDENTIFIED LEAKAGE.

c. 25 gpm total leakage averaged over any 24 hour period.

SR 3.4.6.1 LCO 3.4.6 d. ~~1 gpm leakage~~ at a reactor coolant system pressure at 1000 ± 50 psig from any reactor coolant system pressure isolation valve ~~specified in Table 3.4.3.2-1~~. LA.1

e. 2 gpm increase in UNIDENTIFIED LEAKAGE within any 24 hour period.

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2 and 3. ← add proposed MODE 3 RHR Allowance L.1

ACTION: ← add proposed ACTIONs Notes 1 and 2 A.2

a. With any PRESSURE BOUNDARY LEAKAGE, be in at least HOT SHUTDOWN within 12 hours and in COLD SHUTDOWN within the next 24 hours.

b. With any reactor coolant system leakage greater than the limits in b and/or c, above, reduce the leakage rate to within the limits within 4 hours or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

ACTION A

ACTION B

c. With any reactor coolant system pressure isolation valve leakage greater than the above limits, isolate the high pressure portion of the affected system from the low pressure portion within 4 hours by use of at least two closed valves, or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours. L.2

d. ~~With one or more high/low pressure interface valve leakage pressure monitors inoperable, restore the inoperable monitor(s) to OPERABLE status within 7 days or verify the pressure to be less than the alarm setpoint at least once per 12 hours by local indication; restore the inoperable monitor(s) 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 12 hours.~~ LC.1

e. With any reactor coolant system leakage greater than the limit in e, above, identify the source of leakage within 4 hours or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

<See ITS 3.4.5>

M.1

add Required Actions A.1 and A.2 Note

A.1

ITS 3.4.6

SURVEILLANCE REQUIREMENTS

<See ITS 3.4.5>

4.4.3.2.1 The reactor coolant system leakage shall be demonstrated to be within each of the above limits on average once per 8 hours not to exceed 12 hours.*

SR 3.4.6.1 4.4.3.2.2 Each reactor coolant system pressure isolation valve ~~specified in Table 3.4.3.2-1~~ shall be demonstrated OPERABLE:

LA.1

a. Pursuant to Specification 4.0.5, except that in lieu of any leakage testing required by Specification 4.0.5, each valve shall be demonstrated OPERABLE by verifying leakage to be within its limit:

1. At least ~~once per 18 months, and~~ LA.2

~~2. Prior to returning the valve to service following maintenance, repair or replacement work on the valve which could affect its leakage rate.~~ L.3

NOTE to SR 3.4.6.1 The provisions of Specification 4.0.4 are not applicable for entry into OPERATIONAL CONDITION 3.

~~b. By demonstrating OPERABILITY of the high/low pressure interface valve leakage pressure monitors by performance of a:~~

~~1. CHANNEL FUNCTIONAL TEST at least once per 31 days, and~~

~~2. CHANNEL CALIBRATION at least once per 18 months,~~

~~With the alarm setpoint for the:~~

~~1. HPCS system ≤ 100 psig.~~

~~2. LPCS system ≤ 500 psig.~~

~~3. LPCI/shutdown cooling system ≤ 400 psig.~~

~~4. RHR shutdown cooling ≤ 190 psig.~~

~~5. RCIC ≤ 90 psig.~~

LC.1

<See ITS 3.4.5>

*Technical Specification 4.0.2 does not apply.

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A.1

ITS 3.4.6

TABLE 3.4.3.2-1

REACTOR COOLANT SYSTEM PRESSURE ISOLATION VALVES

SYSTEM	VALVE NUMBER	FUNCTION
a. LPCS	E21-F006	LPCS Injection
	E21-F005	LPCS Injection
b. HPCS	E22-F005	HPCS Injection
	E22-F004	HPCS Injection
c. RHR	E12-F041A	LPCI Injection
	E12-F041B	LPCI Injection
	E12-F041C	LPCI Injection
	E12-F042A	LPCI Injection
	E12-F042B	LPCI Injection
	E12-F042C	LPCI Injection
	E12-F050A	Shutdown Cooling Return
	E12-F050B	Shutdown Cooling Return
	E12-F053A	Shutdown Cooling Return
	E12-F053B	Shutdown Cooling Return
	E12-F009	Shutdown Cooling Suction
	E12-F008	Shutdown Cooling Suction
d. RCIC	E51-F066	RCIC Head Spray
	E5I-F065	RCIC Head Spray

LA.1

ADMINISTRATIVE

A.1 In the conversion of the LaSalle 1 and 2 current Technical Specifications (CTS) to the proposed plant specific Improved Technical Specifications (ITS), certain wording preferences or conventions are adopted that do not result in technical changes (either actual or interpretational). Editorial changes, reformatting, and revised numbering are adopted to make the ITS consistent with the BWR Standard Technical Specifications, NUREG-1434, Rev. 1 (i.e., the Improved Standard Technical Specifications (ISTS)).

A.2 The ITS 3.4.6 ACTIONS include two Notes not currently provided in the CTS. The first Note to the ACTIONS ("Separate Condition entry is allowed for each flow path") provides explicit instructions for proper application of the ACTIONS for Technical Specification compliance. In conjunction with the proposed Specification 1.3 - "Completion Times," this Note provides direction consistent with the intent of the existing ACTIONS for inoperable PIVs. The second Note to the ACTIONS ("Enter applicable Conditions and Required Actions for systems made inoperable by PIVs") facilitates the use and understanding of the intent to consider any system affected by inoperable PIVs, which is to have its ACTIONS also apply if it is determined to be inoperable. With the ITS LCO 3.0.6, this intent would not necessarily apply. This clarification is consistent with the intent and interpretation of the existing Technical Specifications, and is therefore considered an administrative presentation preference.

A.3 LaSalle Unit 1 CTS Table 3.4.3.2-1 contains a footnote * which provides a one time waiver to the surveillance test requirement for the certain valves which applied until the first refueling outage. LaSalle Unit 1 has completed a number of refueling outages, and the exception provided in the footnote is no longer applicable or meaningful. Therefore, the footnote has been deleted. This change does not affect the requirements associated with this specification and is administrative in nature.

TECHNICAL CHANGES - MORE RESTRICTIVE

M.1 CTS 3.4.3.2 Action c requires, with any reactor coolant system PIV leakage greater than the limit, the high pressure portion of the affected system to be isolated from the low pressure portion of the system by the use of two closed valves. The CTS do not include any restrictions concerning the types of valves that may be used to satisfy the isolation requirement of this action. In this same condition, ITS 3.4.6 Required Actions A.1 and A.2 are modified by a Note. The Note requires the valves used to provide isolation between the high pressure and low pressure portions of the affected system to have been verified to meet the PIV leakage limits within the required Surveillance Frequency and that the valves be in the reactor coolant system or the high pressure portion of the affected system. The addition of this Note to the CTS represents an additional

TECHNICAL CHANGES - MORE RESTRICTIVE

M.1 (cont'd) restriction on plant operation necessary to help ensure the valves used to isolate the high pressure portion from the low pressure portion of the affected system are capable of preventing the overpressurization of the low pressure portion of the system.

TECHNICAL CHANGES - LESS RESTRICTIVE

"Generic"

LA.1 The list of pressure isolation valves (PIVs) in CTS Table 3.4.3.2-1 are proposed to be relocated to the Technical Requirements Manual (TRM). The listing of valves which are subject to the RCS PIV Leakage Specification are related to design and are not necessary for ensuring PIV leakage is maintained within limits. ITS 3.4.6 requires the leakage from each RCS PIV to be within limits. These requirements are adequate for ensuring PIV leakage is maintained within limits for the required valves. Therefore, the relocated list is not required to be in the ITS to provide adequate protection of the public health and safety. This change is also consistent with Generic Letter 91-08, which allowed lists of components to be relocated to plant controlled documents. The TRM will be incorporated by reference into the UFSAR at ITS implementation. Changes to the TRM will be controlled by the provisions of 10 CFR 50.59. In addition to this relocation, all references to the Table in CTS 3.4.3.2.d and 4.4.3.2.2 have been deleted. The Bases identifies that the list of PIVs are located in the TRM.

LA.2 Details of the first Frequency for performing CTS 4.4.3.2.2.a are proposed to be relocated to the Inservice Testing (IST) Program (covered by CTS 4.0.5). The requirement to leak test each PIV "At least once per 18 months" is not required to be in Technical Specifications to assure the PIVs are leak tested at least once per 18 months since the IST Program, required by 10 CFR 50.55a, provides 18 month or less leak test requirements for these valves. Compliance with 10 CFR 50.55a, and as a result the IST Program, is required by the LaSalle Operating Licensse. these controls are adequate to ensure the required leak rate testing of PIV is performed and do not need to be in the ITS to provide adequate protection of the public health and safety. Changes to the IST Program will be controlled by the provisions of the proposed IST Program in Chapter 5 of the ITS.

TECHNICAL CHANGES - LESS RESTRICTIVE (continued)

LC.1 The requirements of CTS 3.4.3.2 ACTION d and 4.4.3.2.b concerning high/low pressure interface valve leakage pressure monitors and interlocks do not necessarily relate directly to the leakage limit requirements of the RCS PIVs. The BWR Standard Technical Specifications, NUREG-1434, Rev. 1, does not specify indication-only or alarm-only equipment to be OPERABLE to support OPERABILITY of a system or component. Control of the availability of, and necessary compensatory activities if not available, for indications and monitoring instrumentation are addressed by plant operational procedures and policies. In addition, the leakage limit requirements of ITS 3.4.6 and the leakage test requirements of SR 3.4.6.1 will ensure that the limits will be maintained or the appropriate ACTIONS will be taken. As such, the relocated requirements are not required to be in the ITS to provide adequate protection of the public health and safety. Therefore, this instrumentation, along with the supporting ACTIONS and Surveillances, is proposed to be relocated to the Technical Requirements Manual (TRM). The TRM will be incorporated by reference into the UFSAR at ITS implementation. Changes to the TRM will be controlled by the provisions of 10 CFR 50.59.

"Specific"

L.1 Currently, PIV leakage limits of CTS 3.4.3.2 are required to be met in MODES 1, 2, and 3. A MODE 3 exception is included in the APPLICABILITY of ITS 3.4.6 for valves in the shutdown cooling flow path when needed for the shutdown cooling function. This change resolves a conflict in the current Specifications that requires shutdown cooling flow path isolation if the pressure isolation valve leakage is not within limits, even with reactor coolant system pressure below the RHR cut-in permissive pressure when shutdown cooling is required to be OPERABLE and operating. Although alternative methods of decay heat removal could be established, shutdown cooling is the preferred method. Further, its use with leaky pressure isolation valves poses no risk at low pressure since the high to low pressure interface does not exist.

L.2 The requirement of CTS 3.4.3.2 ACTION c to isolate the high-pressure portion of the affected system from the low-pressure portion within 4 hours using at least two closed valves has been revised to require one valve to be closed within 4 hours (ITS 3.4.6 Required Action A.1), and a second valve to be closed within 72 hours (ITS 3.4.6 Required Action A.2). The 4 hour Completion Time to close the first valve provides time to reduce leakage in excess of the allowable limit and to isolate the flowpath if leakage cannot be reduced while corrective actions to reseat the leaking PIVs are taken. The 4 hours allows time for these

TECHNICAL CHANGES - LESS RESTRICTIVE

L.2 (cont'd) actions and restricts the time of operation with leaking valves. The 72 hour Completion Time to close the second valve considers the time required to complete the Required Action, the low probability of a second valve failing during this time period, and the low probability of a pressure boundary rupture of the low pressure ECCS piping when over pressurized to reactor pressure (NEDC-31339, "BWR Owners' Group Assessment of Emergency Core Cooling System Pressurization in Boiling Water Reactors," November 1986). This change is also consistent with the BWR ISTS, NUREG-1434, Rev. 1.

L.3 CTS 4.4.3.2.2.a.2 requires the PIV to be leak checked before returning the PIV to service following maintenance, repair, or replacement work on the PIV. The explicit post maintenance Surveillance Requirements of CTS 4.4.3.2.2.a.2 are not required and have been deleted from the Technical Specifications. Any time the OPERABILITY of a system or component has been affected by repair, maintenance, or replacement of a component, post maintenance testing is required to demonstrate OPERABILITY of the system or component. After restoration of a component that caused a required SR to be failed, CTS 4.0.1 and 4.0.3 (proposed SR 3.0.1) require the appropriate SRs (in this case CTS 4.4.3.2.2.a and proposed SR 3.4.6.1) to be performed to demonstrate OPERABILITY of the affected components. Entry into the applicable MODES without performing this post maintenance testing also continues to be precluded except where allowed, as discussed in the Bases for proposed SR 3.0.1.

L.4 The CTS 3.4.3.2.d limit on PIV leakage has been revised in ITS SR 3.4.6.1 to specify leakage based on valve size with a maximum limit of 5 gpm. This change acknowledges that smaller valves should not be allowed to leak as much as larger valves. The 1 gpm acceptance criteria is not an indicator of imminent accelerated deterioration of valves or of potential valve failure. The increase in individual valve allowable leakage will have no impact on allowable leakage from the total RCS. This change will also decrease the time spent on unnecessary maintenance of PIVs. Such maintenance contributes to increases in radiation exposures. In addition, a study has been performed, NEDC-31339, "BWR Owners' Group Assessment of Emergency Core Cooling System Pressurization in Boiling Water Reactors," November 1986, that showed the probability of a pressure boundary rupture of the low pressure ECCS piping when overpressurized to reactor pressure is very low. This change is consistent with ASME Code requirements and the BWR ISTS, NUREG-1434, Rev. 1.

RELOCATED SPECIFICATIONS

None

A.1

ITS 3.4.7

REACTOR COOLANT SYSTEM

3/4.4.3 REACTOR COOLANT SYSTEM LEAKAGE
LEAKAGE DETECTION SYSTEMS

LIMITING CONDITION FOR OPERATION

LCO 3.4.7 3.4.3.1 The following reactor coolant system leakage detection systems shall be OPERABLE:

a. The primary containment atmosphere particulate radioactivity monitoring system, L.1

b. The primary containment sump flow monitoring system, and

c. Either the primary containment air coolers condensate flow rate monitoring system or the primary containment atmosphere gaseous radioactivity monitoring system. L.1

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2 and 3.

ACTION: add proposed Note to Actions A and D L.2

ACTIONS A,B,C, and D With only two of the above required leakage detection systems OPERABLE, operation may continue for up to 30 days provided grab samples of the containment atmosphere are obtained and analyzed at least once per 24 hours when the required gaseous and/or particulate radioactive monitoring system is inoperable; L.1 otherwise, be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours. ACTION E

add proposed ACTION F A.2

SURVEILLANCE REQUIREMENTS

4.4.3.1 The reactor coolant system detection systems shall be demonstrated OPERABLE by: add proposed Note to SR Table L.3

SR 3.4.7.1 SR 3.4.7.2 SR 3.4.7.3 a. Primary containment atmosphere particulate and gaseous monitoring systems-performance L.1 of a CHANNEL CHECK at least once per 12 hours, a CHANNEL FUNCTIONAL TEST at least once per 31 days and a CHANNEL CALIBRATION at least once per ~~18~~ 24 months. LE.1

SR 3.4.7.2 SR 3.4.7.3 b. Primary containment sump flow monitoring system-performance of a CHANNEL FUNCTIONAL TEST at least once per 31 days and a CHANNEL CALIBRATION TEST at least once per ~~18~~ 24 months~~*~~. A.3 LE.1

SR 3.4.7.2 SR 3.4.7.3 c. Primary containment air coolers condensate flow rate monitoring system-performance of a CHANNEL FUNCTIONAL TEST at least once per 31 days and a CHANNEL CALIBRATION at least once per ~~18~~ 24 months. LE.1

~~*The specified 18 month interval may be waived for Cycle 1 provided the surveillance is performed during Refuel 1.~~ A.3

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ITS 3.4.7

3/4.4.3 REACTOR COOLANT SYSTEM LEAKAGE
LEAKAGE DETECTION SYSTEMS

A.1

LIMITING CONDITION FOR OPERATION

LCO 3.4.7

3.4.3.1 The following reactor coolant system leakage detection systems shall be OPERABLE:

a. The primary containment atmosphere particulate radioactivity monitoring system, L.1

b. The primary containment sump flow monitoring system, and

c. Either the primary containment air coolers condensate flow rate monitoring system or the primary containment atmosphere gaseous radioactivity monitoring system. L.1

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2 and 3. L.2

add proposed Note to Actions A and D

ACTION:

Actions A, B, C, and D

With only two of the above required leakage detection systems OPERABLE, operation may continue for up to 30 days provided grab samples of the containment atmosphere are obtained and analyzed at least once per 24 hours when the required gaseous and/or particulate radioactive monitoring system is inoperable; L.1

Action E

otherwise, be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

add proposed ACTION F A.2

SURVEILLANCE REQUIREMENTS

4.4.3.1 The reactor coolant system detection systems shall be demonstrated OPERABLE by: L.3

add proposed Note to SR Table

SR 3.4.7.1
SR 3.4.7.2
SR 3.4.7.3

a. Primary containment atmosphere particulate and gaseous monitoring systems L.1 performance of a CHANNEL CHECK at least once per 12 hours, a CHANNEL FUNCTIONAL TEST at least once per 31 days and a CHANNEL CALIBRATION at least once per ~~18~~ 24 months. LE.1

SR 3.4.7.2
SR 3.4.7.3

b. Primary containment sump flow monitoring system-performance of a CHANNEL FUNCTIONAL TEST at least once per 31 days and a CHANNEL CALIBRATION TEST at least once per ~~18~~ 24 months. LE.1

SR 3.4.7.2
SR 3.4.7.3

c. Primary containment air coolers condensate flow rate monitoring system-performance of a CHANNEL FUNCTIONAL TEST at least once per 31 days and a CHANNEL CALIBRATION at least once per ~~18~~ 24 months. LE.1

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ADMINISTRATIVE

A.1 In the conversion of the LaSalle 1 and 2 current Technical Specifications (CTS) to the proposed plant specific Improved Technical Specifications (ITS), certain wording preferences or conventions are adopted that do not result in technical changes (either actual or interpretational). Editorial changes, reformatting, and revised numbering are adopted to make the ITS consistent with the BWR Standard Technical Specifications, NUREG-1434, Rev. 1 (i.e., the Improved Standard Technical Specifications (ISTS)).

A.2 Currently, no Actions are provided in CTS 3.4.3.1 if all required RCS leakage detection systems are inoperable. Therefore, CTS 3.0.3 must be entered. The revised presentation of ACTIONS is proposed to explicitly identify that LCO 3.0.3 is required to be entered if all required RCS leakage detection systems are inoperable. Therefore ITS 3.4.7 ACTION F has been added to be consistent with the current requirements and is considered a presentation preference. Therefore, this change is administrative.

A.3 LaSalle Unit 1 CTS 4.4.3.1.b contains a footnote * which provides a one time exception to the surveillance test requirement for the drywell sump flow monitoring system channel calibration which applied until the first refueling outage. LaSalle Unit 1 has completed a number of refueling outages, and the exception provided in the footnote is no longer applicable or meaningful. Therefore, the footnote has been deleted. This change does not affect the requirements associated with this specification and is administrative in nature.

TECHNICAL CHANGES - MORE RESTRICTIVE

M.1 None

TECHNICAL CHANGES - LESS RESTRICTIVE

"Generic"

LE.1 The Frequencies for performing the Channel Calibrations required by CTS 4.4.3.1.a, 4.4.3.1.b, and 4.4.3.1.c (proposed SR 3.4.7.3) have been extended from 18 to 24 months. This SR ensures that the required primary containment atmosphere particulate, atmospheric gaseous, floor drain sump flow, and air cooler condensate flow rate monitoring systems are operable and within the established calibration requirements. The proposed change will allow these Surveillances to extend their Surveillance Frequency from the current 18 month Surveillance Frequency (i.e., a maximum of 22.5 months accounting for the allowable grace period specified in current Specification 4.0.2 and proposed SR 3.0.2) to a 24 month Surveillance Frequency (i.e., a maximum of 30 months accounting for the allowable grace period specified in current Specification 4.0.2 and proposed SR 3.0.2). This proposed change was evaluated in accordance with the guidance provided in NRC Generic Letter No. 91-04, "Changes in Technical Specification Surveillance Intervals to Accommodate a 24-Month Fuel Cycle," dated April 2, 1991. Reviews of historical maintenance and surveillance data have shown that these tests normally pass their Surveillances at the current Frequency. An evaluation has been performed using this data, and it has been determined that the effect on safety due to the extended Surveillance Frequency will be minimal. The primary containment atmosphere particulate and gaseous, and air cooler condensate flow rate monitoring systems provide backup to the primary containment floor drain sump flow monitoring system, in that they both will alert the operators to unanticipated unidentified leakage from the RCPB. The primary coolant air cooler condensate flow rate monitoring system monitors condensate from the drywell coolers that is routed to the floor drain sump. The instrumentation consists of a flow transmitter mounted locally while having indicating and alarm instrumentation in the control room. An adjustable alarm is set to annunciate on the condensate flow rate approaching the Technical Specification limit. These systems do not provide for the actuation of any safety devices. The equipment provides a monitoring function only which alerts the operator to a potential plant problem. The setpoint of these devices is not an assumption in any event analysis. Based on the redundancy of detection methods, the other functional test performed on this equipment, and the historical calibration records, the impact, if any, of this change on system availability is minimal. In addition, the proposed 24 month Surveillance Frequencies, if performed at the maximum interval allows by proposed SR 3.0.2 (30 months) do not invalidate any assumptions in the plant licensing basis.

TECHNICAL CHANGES - LESS RESTRICTIVE (continued)

"Specific"

L.1 CTS 3.4.3.1 requires the primary containment atmospheric particulate, sump flow, and either the air coolers condensate flow rate or atmospheric gaseous radioactivity monitoring systems to be OPERABLE. These required RCS leakage detection monitoring systems are rearranged in ITS 3.4.7 to require one method which can quantify the unidentified leakage and two diverse detection methods which provide indication of increased leakage.

The CTS 3.4.3.1 Action allows only one of the required RCS leakage detection monitoring systems to be inoperable (i.e., two required systems OPERABLE) for 30 days. ITS 3.4.7 Actions have been developed to allow up to two of the required RCS leakage detection systems to be inoperable (i.e., one required system OPERABLE) for 30 days. In this proposed condition, one method to detect indication of increased RCS leakage will still be OPERABLE. In addition, ITS SR 3.4.5.1 will continue to require the unidentified leakage and identified leakage (as part of the total leakage limit) to be quantified once per 12 hours.

The primary containment atmospheric particulate and gaseous monitoring system in CTS 3.4.3.1.a and c are grouped so that only one of the two is required in ITS LCO 3.4.7.b, instead of the current requirement that may require both systems to be OPERABLE, since they provide the same type of indication. A diverse method to quantify increased leakage is still provided by the primary containment sump flow monitoring system, and this is the primary method of quantifying unidentified leakage. The CTS 3.4.3.1 Action, which allows only one of the two atmospheric monitoring systems to be inoperable, has been modified in ITS 3.4.7 ACTION B to allow the "required" atmospheric monitoring systems (i.e., both particulate and gaseous monitors), to be inoperable indefinitely if grab samples are taken every 12 hours and the condensate flow monitoring system is OPERABLE, and for 30 days if both the atmospheric monitoring and condensate flow monitoring systems are inoperable and a grab sample is taken every 12 hours, consistent with the new requirement in ITS LCO 3.4.7.b that only one of these two atmospheric monitors be OPERABLE. Also, the condensate flow monitor can be inoperable indefinitely if a channel check is performed every 8 hours and the atmospheric monitor is OPERABLE. In addition, CTS 4.4.3.1.a has also been modified to only require the "required" primary containment atmospheric monitoring system to be tested (proposed ITS SRs 3.4.7.1, 3.4.7.2, and 3.4.7.3) to reflect these new requirements.

TECHNICAL CHANGES - LESS RESTRICTIVE

L.1 (cont'd) Therefore, since the capability to detect unidentified leakage will still be maintained, and identified and unidentified leakage is still required to be quantified, this change, which is consistent with the BWR/6 ISTS, is considered to be acceptable.

L.2 Currently, CTS 3.0.4 would preclude a change in MODES with a leakage detection monitoring system inoperable. A statement is added, as a NOTE to ITS 3.4.7 ACTION A and ACTION D, that LCO 3.0.4 is not applicable for the condition of the primary containment sump flow rate monitoring system or the combination of an inoperable atmospheric monitoring system and primary containment air cooler condensate flow rate inoperable. When this allowance is used, other means of detection, the compensatory actions for the inoperable system, or the requirement that unidentified leakage be quantified in accordance with ITS 3.4.5, will provide adequate indication of RCS leakage. Since 1) probability assessments have determined that a 30 day allowed out of service time for two of the three leakage detection systems is acceptable; 2) a leakage detection system is still OPERABLE; and 3) compensatory measures will ensure leakage is quantified, the LCO 3.0.4 exception is considered to provide no significant impact on safety and is acceptable. This change is consistent with NUREG-1434.

L.3 A Note has been added to CTS 4.4.3.1 (Note to ITS 3.4.7 Surveillance Requirements) to allow a channel to be inoperable for up to 6 hours solely for performance of required Surveillances provided the other Leakage Detection System channel is OPERABLE. The 6 hour testing allowance has been granted by the NRC in Technical Specification amendments for Georgia Power Company's Hatch Unit 1 (Amendment 185) and Unit 2 (Amendment 125) and in the ITS amendment for Washington Public Power Supply System Unit 2 (Amendment 149). The NRC has also granted this allowance in other topical reports for the Reactor Protection System, Emergency Core Cooling System, and Isolation System Instrumentation. The 6 hour testing allowance does not significantly reduce the probability of properly monitoring leakage since the other channel must be OPERABLE for this allowance to be used.

RELOCATED SPECIFICATIONS

None

REACTOR COOLANT SYSTEM

3/4.4.5 SPECIFIC ACTIVITY

LIMITING CONDITION FOR OPERATION

LCO 3.4.8 3.4.5 The specific activity of the primary coolant shall be limited to:

a. Less than or equal to 0.2 microcurie per gram DOSE EQUIVALENT I-131, and

~~b. Less than or equal to 100/Ē microcuries per gram.~~ L.1

APPLICABILITY: OPERATIONAL CONDITIONS 1, and 2, 3 ~~and 4~~ with any main steam line not isolated L.2

ACTION:

a. In OPERATIONAL CONDITIONS 1, 2 or 3 with the specific activity of the primary coolant; add proposed Note to Action A L.3

ACTION A 1. 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 for more than 48 hours during one continuous time interval or greater than 4.0 microcuries per gram DOSE EQUIVALENT

ACTION B I-131, be in at least HOT SHUTDOWN with the main steam line isolation valves closed within 12 hours. L.2

~~2. Greater than 100/Ē microcuries per gram, be in at least HOT SHUTDOWN with the main steamline isolation valves closed within 12 hours.~~ L.1

b. In OPERATIONAL CONDITIONS 1, 2, 3 ~~or 4~~, with the specific activity of the primary coolant greater than 0.2 microcurie per gram DOSE EQUIVALENT I-131 ~~or greater than 100 Ē microcuries per gram,~~ perform the sampling and analysis requirements of Item 4a of Table 4.4.5-1 until the specific activity of the primary coolant is restored to within the limit. L.2 L.1

Required Actions A.1 and B.1

A.2

~~c. In OPERATIONAL CONDITION 1 or 2, with:~~

~~1. THERMAL POWER changed by more than 15% of RATED THERMAL POWER in 1 hour*, or~~

~~2. The off-gas level, prior to the holdup line, increased by more than 25,000 microcuries per second in one hour during steady state operation at release rates less than 100,000 microcuries per second, or~~

~~*Not applicable during the Startup Test Program.~~

REACTOR COOLANT SYSTEM

LIMITING CONDITION FOR OPERATION (Continued)

ACTION (Continued)

A.2

~~3. The off-gas level, prior to the holdup line, increased by more than 15% in 1 hour during steady state operation at release rates greater than 100,000 microcuries per second,~~

~~perform the sampling and analysis requirements of Item 4b of Table 4.4.5-1 until the specific activity of the primary coolant is restored to within its limit.~~

SURVEILLANCE REQUIREMENTS

SR3.4.8.1 4.4.5 The specific activity of the reactor coolant shall be demonstrated to be within the limits by performance of the sampling and analysis program of Table 4.4.5-1.

TABLE 4.4.5-1

PRIMARY COOLANT SPECIFIC ACTIVITY SAMPLE AND ANALYSIS PROGRAM

TYPE OF MEASUREMENT AND ANALYSIS	SAMPLE AND ANALYSIS FREQUENCY	OPERATIONAL CONDITIONS IN WHICH SAMPLE AND ANALYSIS REQUIRED	
1. ~~Gross Beta and Gamma Activity Determination~~	~~At least once per 72 hours~~	~~1, 2, 3~~	L.1
SR 3.4.8.1 — 2. Isotopic Analysis for DOSE EQUIVALENT I-131 Concentration	At least once per ~~31~~ 7 days	1	M.1
3. ~~Radiochemical for Ē Determination~~	~~At least once per 6 months*~~	~~1~~	L.1
4. Isotopic Analysis for Iodine Including I-131, I-133 and I-135 Required Actions A.1 and B.1	a) At least once per 4 hours, whenever the specific activity exceeds a limit, as required by ACTION b.	1#, 2#, 3#, 4#	L.2
	~~b) At least one sample, between 2 and 6 hours following the change in THERMAL POWER or off-gas level, as required by ACTION c.~~	~~1, 2~~	A.2
5. ~~Isotopic Analysis of an Off-gas Sample Including Quantitative Measurements for at least Xe-133, Xe-135 and Kr-88~~	~~At least once per 31 days~~	~~1~~	LA.1

~~*Sample to be taken after a minimum of 2 EFPD and 20 days of POWER OPERATION have elapsed since reactor was last subcritical for 48 hours or longer.~~ L.1

#Until the specific activity of the primary coolant system is restored to within its limits.
Required Actions A.1 and B.1

A.1

Page 3 of 6

ITS 3.4.8

A.1

REACTOR COOLANT SYSTEM

3/4.4.5 SPECIFIC ACTIVITY

LIMITING CONDITION FOR OPERATION

LCO 3.4.8

3.4.5 The specific activity of the primary coolant shall be limited to:

a. Less than or equal to 0.2 microcurie per gram DOSE EQUIVALENT I-131, ~~and~~

~~b. Less than or equal to 100/E microcuries per gram.~~ L.1

APPLICABILITY: OPERATIONAL CONDITIONS 1, and 2, 3, ~~and 4.~~ with any main steam line not isolated

ACTION:

a. In OPERATIONAL CONDITION 1, 2, or 3 with the specific activity of the primary coolant; L.2

add proposed Note A to Action A L3

ACTION A

ACTION B

1. Greater than 0.2 microcurie per gram DOSE EQUIVALENT I-131 but less than or equal to 4 microcurie per gram DOSE EQUIVALENT I-131 for more than 48 hours during one continuous time interval or greater than 4.0 microcuries per gram DOSE EQUIVALENT I-131, be in at least HOT SHUTDOWN with the main steam line isolation valves closed within 12 hours. L.2

~~2. Greater than 100/E microcuries per gram, be in at least HOT SHUTDOWN with the main steamline isolation valves closed within 12 hours.~~ L.1

Required Actions A.1 and B.1

b. In OPERATIONAL CONDITION 1, 2, 3 ~~or 4~~ with the specific activity of the primary coolant greater than 0.2 microcurie per gram DOSE EQUIVALENT I-131 ~~or greater than 100/E microcuries per gram~~, perform the sampling and analysis requirements of Item 4a of Table 4.4.5-1 until the specific activity of the primary coolant is restored to within the limit. L.2 L.1

~~c. In OPERATIONAL CONDITION 1 or 2, with:~~

~~1. THERMAL POWER changed by more than 15% of RATED THERMAL POWER in 1 hour*, or~~

~~2. The off-gas level, prior to the holdup line, increased by more than 25,000 microcuries per second in one hour during steady state operation at release rates less than 100,000 microcuries per second, or~~

A.2

~~*Not applicable during the Startup Test Program.~~

ITS 3.4.8

A.1

REACTOR COOLANT SYSTEM

LIMITING CONDITION FOR OPERATION (Continued)

ACTION: (Continued)

A.2

~~3. The off-gas level, prior to the holdup line, increased by more than 15% in 1 hour during steady state operation at release rates greater than 100,000 microcuries per second,~~

~~perform the sampling and analysis requirements of Item 4b of Table 4.4.5-1 until the specific activity of the primary coolant is restored to within its limit.~~

SURVEILLANCE REQUIREMENTS

SR 3.4.8.1 4.4.5 The specific activity of the reactor coolant shall be demonstrated to be within the limits by performance of the sampling and analysis program of Table 4.4.5-1.

TABLE 4.4.5-1

PRIMARY COOLANT SPECIFIC ACTIVITY SAMPLE AND ANALYSIS PROGRAM

	TYPE OF MEASUREMENT AND ANALYSIS	SAMPLE AND ANALYSIS FREQUENCY	OPERATIONAL CONDITIONS IN WHICH SAMPLE AND ANALYSIS REQUIRED	
	~~1. Gross Beta and Gamma Activity Determination~~	~~At least once per 72 hours~~	~~1, 2, 3~~	L.1
SR 3.4.8.1	2. Isotopic Analysis for DOSE EQUIVALENT I-131 Concentration	At least once per ~~31~~ 7 days M.1	1	
	~~3. Radiochemical for Ē Determination~~	~~At least once per 6 months*~~	~~1~~	L.1
	4. Isotopic Analysis for Iodine Including I-131, I-133 and I-135 Required Actions A.1 and B.1	a) At least once per 4 hours, whenever the specific activity exceeds a limit, as required by ACTION b.	1#, ~~2#, 3#, 4#~~	L.2
		~~b) At least one sample, between 2 and 6 hours following the change in THERMAL POWER or off-gas level, as required by ACTION c.~~	~~1, 2~~	A.2
	~~5. Isotopic Analysis of an Off-gas Sample Including Quantitative Measurements for at least Xe-133, Xe-135 and Kr-88~~	~~At least once per 31 days~~	~~1~~	LA.1

~~*Sample to be taken after a minimum of 2 EFPD and 20 days of POWER OPERATION have elapsed since reactor was last subcritical for 48 hours or longer.~~ L.1

#Until the specific activity of the primary coolant system is restored to within its limits.

Required Actions A.1 and B.1

A.1

ITS 3.4.8

ADMINISTRATIVE

A.1 In the conversion of the LaSalle 1 and 2 current Technical Specifications (CTS) to the proposed plant specific Improved Technical Specifications (ITS), certain wording preferences or conventions are adopted that do not result in technical changes (either actual or interpretational). Editorial changes, reformatting, and revised numbering are adopted to make the ITS consistent with the BWR Standard Technical Specifications, NUREG-1434, Rev. 1 (i.e., the Improved Standard Technical Specifications (ISTS)).

A.2 CTS 3.4.5 Action c requires increased sampling under certain conditions (as specified in CTS Table 4.4.5-1, Item 4.b), when the LCO 3.4.5.a limit is exceeded. (As described in CTS 3.0.1, the Action is only required when the LCO is not met.) CTS Table 4.4.5-1, Item 4.b requires sampling and analysis once between 2 and 6 hours after the special conditions specified in Action c are met. However, CTS 3.4.5 Action b (ITS 3.4.8, Required Actions A.1 and B.1), which is also required to be taken when the LCO 3.4.5.a limit is not met, already requires the same sampling to be performed every 4 hours at all times when the LCO 3.4.5.a limit is not met, not just when the special conditions specified in Action c are met. Thus, the sampling and analysis requirements of CTS 3.4.5 Action c are redundant to the sampling and analysis requirements of CTS 3.4.5 Action b. Therefore, CTS 3.4.5 Action c has been deleted and its deletion is administrative.

TECHNICAL CHANGES - MORE RESTRICTIVE

M.1 This proposed change modifies CTS Table 4.4.5-1, Item 2 (proposed SR 3.4.8.1), to change the Frequency for isotopic analysis for dose equivalent I-131 concentration from at least once per 31 days to at least once per 7 days. The increased Frequency provides a compensatory measure for ensuring that even with deletion of the requirement that gross specific activity remain less than or equal to 100/E-bar μCi/gram, offsite doses will remain within a small fraction of the limits of 10 CFR 100. This change is more restrictive on plant operations.

TECHNICAL CHANGES - LESS RESTRICTIVE

"Generic"

LA.1 The CTS Table 4.4.5-1, Item 5, requires an isotopic analysis of an offgas sample, including quantitative measurements for xenon and krypton. The offgas isotopic analysis for xenon and krypton are not direct measurements related to the limits of ITS 3.4.8. These analyses are used to routinely monitor and trend

TECHNICAL CHANGES - LESS RESTRICTIVE

LA.1 (cont'd) coolant activity and are applicable to plant specific controls and administrative limits only. Therefore, this Surveillance is proposed to be relocated to the Technical Requirements Manual (TRM). The requirements of proposed SR 3.4.8.1 provide adequate assurance that RCS specific activity will be maintained within required limits. As a result, the additional analysis requirements for xenon and krypton are not necessary for assuring RCS specific activity is within required limits do not need to be in the ITS to provide adequate protection of the public health and safety. The TRM will be incorporated by reference into the UFSAR at ITS implementation. Changes to TRM will be controlled by the provisions of 10 CFR 50.59.

"Specific"

L.1 The CTS LCO 3.4.5.b requirement to maintain specific activity ≤ 100/E-bar μCi/gm has been deleted. The current Bases state that the intent of the requirement to limit the specific activity of the reactor coolant is to ensure that whole body and thyroid doses at the site boundary would not exceed a small fraction of the 10 CFR 100 limits (i.e., 10% of 25 rem and 300 rem, respectively) in the limiting event of a main steam line failure outside containment. To ensure that offsite thyroid doses do not exceed 30 rem, reactor coolant DOSE EQUIVALENT I-131 (DEI) is limited to less than or equal to 0.2 μCi/gm. Current Technical Specifications also limit reactor coolant gross specific activity to less than or equal to 100/E-bar μCi/gm to ensure that whole body doses do not exceed 2.5 rem.

CTS 3.11.2.2 (ITS 3.7.6) associated with radioactive effluents requires that the gross gamma radioactivity rate of the noble gases measured at the Offgas System pretreatment monitor station be limited to less than or equal to 340,000 μCi/second. The current Bases for CTS 3.11.2.2 state that 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 10 CFR 100 limits in the event this effluent is inadvertently discharged without treatment directly to the environment.

The Offgas System, as required by CTS 3.11.2.2 and ITS 3.7.6, provides reasonable assurance the reactor coolant gross specific activity is maintained at a sufficiently low level to preclude offsite doses from exceeding a small fraction of the 10 CFR 100 limits in the event of a main steam line failure. Therefore, CTS 3.4.5.b is redundant and places an unnecessary burden on the licensee without a commensurate increase in the margin of safety. Elimination of CTS 3.4.5.b will

TECHNICAL CHANGES - LESS RESTRICTIVE

L.1 (cont'd) allow plant personnel to focus attention on efficient, safe operation of the plant without the unnecessary distraction of the redundant Surveillance Requirement. Additional assurance that the offsite doses will not exceed a small fraction of the 10 CFR 100 limits is provided by increasing the frequency of sampling and analysis of the reactor coolant for DEI from at least once per 31 days to at least once per 7 days, (see comment M.1). Since (1) the reactor coolant limit on DEI adequately assures that offsite doses will not exceed small fractions of the 10 CFR 100 limits in the event of a main steam line failure outside containment and (2) gross gamma radioactivity rate of the noble gases measured at the Offgas System pretreatment monitor station is limited by ITS 3.7.6 to a value that provides reasonable assurance the reactor coolant gross specific activity is maintained at a sufficiently low level to preclude offsite doses from exceeding a small fraction of the 10 CFR 100 limits, the requirements associated with CTS 3.4.5.b are unnecessary. The associated ACTIONS and Surveillance Requirements are also being deleted, consistent with the LCO requirement deletion.

L.2 The Applicability of CTS 3.4.5 (including Table 4.4.5-1 measurement 4) is Operating Conditions 1, 2, 3, and 4. In ITS 3.4.8, the Applicability is proposed to be limited to those conditions which represent a potential for release of significant quantities of radioactive coolant to the environment. MODE 4 is omitted since the reactor is not pressurized and the potential for leakage is significantly reduced. In MODES 2 and 3, with the main steam lines isolated, no escape path exists for significant releases and requirements for limiting the specific activity are not required. CTS 3.4.5 Actions a and b (ITS 3.4.8, ACTIONS A and B) are also modified to reflect the new Applicability, and an option for exiting the applicable MODES is provided for cases where isolation is not desired (ITS 3.4.5 Required Actions B.2.2.1 and B.2.2.2).

L.3 Currently, MODE changes are precluded by CTS 3.0.4 if the limit of CTS 3.4.5.a is not met. A Note is added to CTS 3.4.5 Action a (ITS 3.4.8 ACTION A) to indicate that LCO 3.0.4 is not applicable during the first 48 hours of failure to meet the LCO limit provided the specific activity is $\leq 4.0\ \mu$Ci/gm DEI. Entry into the applicable MODES should not be restricted since the most likely response to the condition is restoration of compliance within the allowed 48 hours. Further, since the LCO limits assure the dose due to a MSLB would be a small fraction of the 10 CFR 100 limits, operation during the allowed time frame would not represent a significant impact to the health and safety of the public.

RELOCATED SPECIFICATIONS

None

A.1

REACTOR COOLANT SYSTEM

3/4.4.9 RESIDUAL HEAT REMOVAL

HOT SHUTDOWN

LIMITING CONDITION FOR OPERATION

, unless at least one recirculation pump is in operation, L.1

LCO 3.4.9 A.2 **3.4.9.1** Two# shutdown cooling mode loops of the residual heat removal (RHR) system shall be OPERABLE and at least one shutdown cooling mode loop shall be in operation* ~~##~~ ~~with each loop consisting of at least:~~

~~a. One OPERABLE RHR pump, and~~

~~b. One OPERABLE RHR heat exchanger.~~ LA.1

APPLICABILITY: OPERATIONAL CONDITION 3, with reactor vessel pressure less than the RHR cut-in permissive setpoint.

ACTION:

add proposed ACTIONS Note 1 L.2

add proposed ACTIONS Note 2 A.3

Action A a. With less than the above required RHR shutdown cooling mode loops OPERABLE, immediately initiate corrective action to return the required loops 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 loop. Be in at least COLD SHUTDOWN within 24 hours. ~~**~~ A.4 A.5

or recirculation loop L.1

Action B b. With no RHR shutdown cooling mode loop in operation, immediately initiate corrective action to return at least one loop to operation as soon as possible. Within 1 hour establish reactor coolant circulation by an alternate method and monitor reactor coolant temperature and pressure at least once per hour.

add proposed SR 3.4.9.1 Note L.2

SURVEILLANCE REQUIREMENTS

SR 3.4.9.1 **4.4.9.1** At least one shutdown cooling mode loop of the residual heat removal system (or alternate method) shall be determined to be in operation ~~and circulating reactor coolant~~ at least once per 12 hours. LA.2

Required Action B.2

or recirculation loop L.1

LCO Note 2 #One RHR shutdown cooling mode loop may be inoperable for up to 2 hours for surveillance testing ~~provided the other loop is OPERABLE and in operation.~~

LCO Note 1 *The shutdown cooling pump may be removed from operation for up to 2 hours per 8 hour period ~~provided the other loop is OPERABLE~~ L.3

~~##The RHR shutdown cooling mode loop may be removed from operation during hydrostatic testing.~~ A.2

~~**Whenever two or more RHR 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.~~ A.5

A.1

REACTOR COOLANT SYSTEM

3/4.4.9 RESIDUAL HEAT REMOVAL

HOT SHUTDOWN

LIMITING CONDITION FOR OPERATION

, unless at least one recirculation pump is in operation, L.1

LCO 3.4.9 3.4.9.1 ~~Two~~# shutdown cooling mode loops of the residual heat removal (RHR) system shall be OPERABLE ~~and at~~ least one shutdown cooling mode loop shall be in operation* ~~##~~ ~~with each loop consisting of at least:~~ A.2

~~a. One OPERABLE RHR pump, and~~

~~b. One OPERABLE RHR heat exchanger.~~ LA.1

APPLICABILITY: OPERATIONAL CONDITION 3, with reactor vessel pressure less than the RHR cut-in permissive setpoint.

ACTION:

add proposed Actions Note 1 L.2

add proposed Actions Note 2 A.3

ACTION A a. With less than the above required RHR shutdown cooling mode loops OPERABLE, immediately initiate corrective action to return the required loops 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 loop. Be in at least COLD SHUTDOWN within 24 hours.** A.4 A.5

or recirculation loop L.1

ACTION B b. With no RHR shutdown cooling mode loop in operation, immediately initiate corrective action to return at least one loop to operation as soon as possible. Within 1 hour establish reactor coolant circulation by an alternate method and monitor reactor coolant temperature and pressure at least once per hour.

SURVEILLANCE REQUIREMENTS

add proposed SR 3.4.9.1 Note L.2

SR 3.4.9.1 4.4.9.1 ~~At least one~~ shutdown cooling mode loop of the residual heat removal system ~~or alternate method~~ shall be determined to be in operation ~~and circulating reactor coolant~~ at least once per 12 hours. LA.2

Required Action B.2

or recirculation loop L.1

LCO Note 2 #One RHR shutdown cooling mode loop may be inoperable for up to 2 hours for surveillance testing ~~provided the other loop is OPERABLE and in operation.~~

LCO Note 1 *The shutdown cooling pump may be removed from operation for up to 2 hours per 8-hour period ~~provided the other loop is OPERABLE.~~ L.3

~~##The RHR shutdown cooling mode loop may be removed from operation during hydrostatic testing.~~ A.2

~~**Whenever two or more RHR 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.~~ A.5

ADMINISTRATIVE

A.1 In the conversion of the LaSalle 1 and 2 current Technical Specifications (CTS) to the proposed plant specific Improved Technical Specifications (ITS), certain wording preferences or conventions are adopted that do not result in technical changes (either actual or interpretational). Editorial changes, reformatting, and revised numbering are adopted to make the ITS consistent with the BWR Standard Technical Specifications, NUREG-1434, Rev. 1 (i.e., the Improved Standard Technical Specifications (ISTS)).

A.2 The CTS 3.4.9.1 footnote ## allowance to remove the RHR shutdown cooling loop from operation during hydrostatic testing has been deleted since these tests are not performed during MODE 3 operation. Since the footnote does not provide any additional allowance, its removal is considered administrative.

A.3 The proposed ACTION Note 2, "Separate Condition entry is allowed for each...," has been added to CTS 3.4.9.1 ACTIONS (ITS 3.4.9 ACTIONS Note 2) and provides more explicit instructions for proper application of the ACTIONS for Technical Specification compliance. In conjunction with the proposed Specification 1.3 - "Completion Times," this Note provides direction consistent with the intent of the existing Actions for inoperable RHR shutdown cooling subsystems.

A.4 The requirement of CTS 3.4.9.1 Action a to demonstrate every 24 hours the operability of at least one alternate method capable of decay heat removal for each inoperable RHR shutdown cooling loop is unnecessary and moot since the reactor is currently required to be in MODE 4 within 24 hours (i.e., exit this Specification). Once in MODE 4, CTS 3.4.9.2 and the ITS 3.4.10 both require the periodic verification of the availability of an alternate decay heat removal method. Since the frequency of the requirement in CTS 3.4.9.1 Action a is of no consequence, its omission is considered an administrative change.

A.5 The CTS 3.4.9.1 footnote ** requirement that if unable to attain cold shutdown when two or more RHR subsystems are inoperable, then maintain reactor coolant temperature as low as practical by use of alternate heat removal methods is deleted since it provides unnecessary duplication of the Actions, contains no additional restrictions on the operation of the plant, and in fact, could be interpreted as a relaxation of the requirements to achieve MODE 4. The Action to be in MODE 4, which is modified by the footnote, adequately prescribes the requirement to make efforts to "maintain reactor coolant temperature as low as

ADMINISTRATIVE

A.5 (cont'd) practical" (i.e., the duplicative requirement of the footnote). If conditions are such that MODE 4 cannot be attained, the Action remains in effect, essentially requiring efforts to reach MODE 4 to continue. Elimination of the footnote reflects an administrative presentation preference.

TECHNICAL CHANGES - MORE RESTRICTIVE

None

TECHNICAL CHANGES - LESS RESTRICTIVE

"Generic"

LA.1 The details in CTS 3.4.9.1.a and CTS 3.4.9.1.b of what constitutes an OPERABLE RHR shutdown cooling subsystem are proposed to be relocated to the Bases. The Bases will indicate that an OPERABLE RHR shutdown cooling subsystem consists of an OPERABLE pump, heat exchanger, and the associated piping and valves. The details for subsystem OPERABILITY are not necessary in ITS 3.4.9. The definition of OPERABILITY suffices. Therefore, the relocated details are not required to be in the ITS to provide adequate protection of the public health and safety. Changes to the Bases will be controlled by the provisions of the proposed Bases Control Program described in Chapter 5 of the ITS.

LA.2 The detail of the method in CTS 4.4.9.1 of verifying operation of the RHR shutdown cooling subsystem (circulating reactor coolant) is proposed to be relocated to the Bases. This detail is not necessary for assuring the RHR shutdown cooling subsystem is in operation. Proposed SR 3.4.9.1 requires verification an RHR shutdown cooling subsystem is operating and is adequate to ensure an RHR shutdown cooling subsystem is circulating reactor coolant. Therefore, the relocated detail is not required to be in the ITS to provide adequate protection of the public health and safety. Changes to the Bases will be controlled by the provisions of the proposed Bases Control Program described in Chapter 5 of the ITS.

TECHNICAL CHANGES - LESS RESTRICTIVE (continued)

"Specific"

L.1 CTS 3.4.9.1, ACTION b, and CTS 4.4.9.1 are proposed to be revised to allow a recirculation pump to be in operation in lieu of a shutdown cooling mode loop of the RHR System. This proposed change provides the "alternate method" as cited in CTS 4.4.9.1 and, therefore, ensures consistency between the SR and LCO. The recirculation pump provides the necessary forced circulation through the core during shutdown for removal of decay heat; therefore, the intent of the CTS LCO continues to be met. Though this proposed change is less restrictive for LaSalle, its contents are consistent with the current Technical Specifications for Washington Nuclear Plant - Unit 2 and Susquehannah Steam Electric Station - Units 1 and 2 and it continues to ensure the decay heat removal function is satisfied.

L.2 CTS 3.4.9.1 requires one RHR shutdown cooling loop to be in operation in Operational Condition 3, with reactor vessel pressure less than the RHR cut-in permissive setpoint. CTS 4.4.9.1 requires a verification that a loop is in operation. CTS 3.0.4 and 4.0.4 require the LCO and Surveillances to be met prior to entry into the applicable mode or other specified conditions. The RHR System cannot be placed in operation until after the applicable conditions necessary to open the RHR shutdown cooling suction valves are met (the RHR shutdown cooling suction valves are interlocked closed at high pressure). Therefore, entry into the conditions should be allowed while depending on the ACTIONS and without performing the Surveillance Requirement. Both LCO 3.0.4 and SR 3.0.4 must be "not applicable" to provide the necessary time to place the system in service following the reduction of pressure to below the cut-in permissive pressure setpoint. Therefore, a Note to the CTS 3.4.9.1 Actions (ITS 3.4.9 ACTIONS Note 1) and a Note to CTS 4.4.9.1 (proposed SR 3.4.9.1) have been added. Without this change, certain entries into the applicable operating conditions would result in intentional temporary noncompliance until the system is placed in service.

L.3 CTS 3.4.9.1 footnote # allows one RHR shutdown cooling loop to be inoperable for 2 hours for surveillance testing, provided the other loop is Operable and in operation. CTS 3.4.9.1 footnote * allows the RHR shutdown cooling pump to be removed from operation for up to 2 hours per 8 hour period, provided the other loop is Operable. The requirements in CTS 3.4.9.1 footnotes # and * (ITS 3.4.9 LCO Notes 1 and 2) are proposed to be changed to delete the "provided" requirements. The allowances of the Notes may be required even when no RHR shutdown cooling loop is in operation or one RHR shutdown

TECHNICAL CHANGES - LESS RESTRICTIVE

L.3 (cont'd) cooling loop is inoperable. Some Surveillances result in shutdown or inoperability of one or both RHR shutdown cooling loops (e.g., Surveillances on the common suction line valves). With one or more RHR shutdown cooling loops inoperable, and Note 2 not being utilized, CTS 3.4.9.1 Action a (ITS 3.4.9 ACTION A) requires an alternate method capable of decay heat removal to be established within 1 hour for each inoperable RHR shutdown cooling loop and, with no RHR or recirculation pump in operation, and Note 1 not being utilized, CTS 3.4.9.1 Action b (ITS 3.4.9 ACTION B) requires establishment of reactor coolant recirculation by an alternate method within 1 hour. If acceptable alternatives are available for decay heat removal (i.e., complying with the Actions), the temporary allowances of the Notes should apply since the alternate methods must be capable of providing adequate decay heat removal. In the case of either Note, at least one RHR shutdown cooling loop will be operable and capable of being placed in service.

RELOCATED SPECIFICATIONS

None

ITS 3.4.10

A.1

REACTOR COOLANT SYSTEM

COLD SHUTDOWN

LIMITING CONDITION FOR OPERATION

unless at least one recirculation pump is in operation, L.1

LCO 3.4.10 3.4.9.2 ~~Two#~~ shutdown cooling mode loops of the residual heat removal (RHR) system shall be ~~OPERABLE* and~~ at least one shutdown cooling mode loop shall be in operation** ## ~~with each loop consisting of at least:~~

~~a. One OPERABLE RHR pump, and~~

~~b. One OPERABLE RHR heat exchanger.~~ LA.1

APPLICABILITY: OPERATIONAL CONDITION 4.

ACTION: add proposed ACTIONS Note A.3

ACTION A a. With less than the above required RHR shutdown cooling mode loops 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 loop.

or recirculation loop L.1

ACTION B b. With no RHR shutdown cooling mode loop in operation, within 1 hour establish reactor coolant circulation by an alternate method and monitor reactor coolant temperature and pressure at least once per hour.

SURVEILLANCE REQUIREMENTS

or recirculation loop L.1

SR 3.4.10.1 4.4.9.2 At least one shutdown cooling mode loop of the residual heat removal system or alternate method shall be determined to be in operation ~~and circulating reactor coolant~~ at least once per 12 hours. LA.2

Required Action B.1

LCO Note 3 #One RHR shutdown cooling mode loop may be inoperable for up to 2 hours for surveillance testing ~~provided the other loop is OPERABLE and in operation.~~ L.2

~~*The normal or emergency power source may be inoperable.~~ A.4

LCO Note 2 **The shutdown cooling pump may be removed from operation for up to 2 hours per 8 hour period ~~provided the other loop is OPERABLE.~~ L.2

LCO Note 1 ##The shutdown cooling mode loop may be removed from operation during hydrostatic testing. L3

A.1

REACTOR COOLANT SYSTEM

COLD SHUTDOWN

LIMITING CONDITION FOR OPERATION

unless at least one recirculation pump is in operation, L.1

LCO 3.4.10 3.4.9.2 ~~Two#~~ shutdown cooling mode loops of the residual heat removal (RHR) system shall be OPERABLE* and at least one shutdown cooling mode loop shall be in operation** ## ~~with each loop consisting of at least:~~

~~a. One OPERABLE RHR pump, and~~

~~b. One OPERABLE RHR heat exchanger.~~ LA.1

APPLICABILITY: OPERATIONAL CONDITION 4.

ACTION: add proposed ACTIONS NOTE A.3

Action A a. With less than the above required RHR shutdown cooling mode loops 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 loop.

or recirculation loop L.1

Action B b. With no RHR shutdown cooling mode loop in operation, within 1 hour establish reactor coolant circulation by an alternate method and monitor reactor coolant temperature and pressure at least once per hour.

SURVEILLANCE REQUIREMENTS

or recirculation loop L.1

SR 3.4.10.1 4.4.9.2 At least one shutdown cooling mode loop of the residual heat removal system ~~or alternate method~~ shall be determined to be in operation ~~and circulating reactor coolant~~ at least once per 12 hours. LA.2

Required Action B.1

LCO Note 3 #One RHR shutdown cooling mode loop may be inoperable for up to 2 hours for surveillance testing ~~provided the other loop is OPERABLE and in operation.~~ L.2

~~*The normal or emergency power source may be inoperable.~~ A.4

LCO Note 2 **The shutdown cooling pump may be removed from operation for up to 2 hours per 8-hour period ~~provided the other loop is OPERABLE.~~ L.2

LCO Note 1 ##The shutdown cooling mode loop may be removed from operation during hydrostatic testing. L.3

ADMINISTRATIVE

A.1 In the conversion of the LaSalle 1 and 2 current Technical Specifications (CTS) to the proposed plant specific Improved Technical Specifications (ITS), certain wording preferences or conventions are adopted that do not result in technical changes (either actual or interpretational). Editorial changes, reformatting, and revised numbering are adopted to make the ITS consistent with the BWR Standard Technical Specifications, NUREG-1434, Rev. 1 (i.e., the Improved Standard Technical Specifications (ISTS)).

A.2 Not used.

A.3 The proposed ACTION Note, "Separate Condition entry is allowed for each RHR shutdown cooling subsystem" has been added to CTS 3.4.9.2 ACTIONS (ITS 3.9.10 ACTIONS Note) and provides more explicit instructions for proper application of the ACTIONS for Technical Specification compliance. In conjunction with the proposed Specification 1.3 - "Completion Times," this Note provides direction consistent with the intent of the existing ACTIONS for inoperable RHR shutdown cooling subsystems.

A.4 CTS 3.4.9.2 footnote * is proposed to be deleted since the allowance it provides (only one power source for the RHR subsystem is required) is duplicative of the ITS definition of OPERABILITY. Any technical changes to the definition of OPERABILITY will be addressed in the Discussion of Changes for ITS 1.0. Therefore, its deletion is considered an administrative change.

TECHNICAL CHANGES - MORE RESTRICTIVE

None

TECHNICAL CHANGES - LESS RESTRICTIVE

"Generic"

LA.1 The details in CTS 3.4.9.2.a and CTS 3.4.9.2.b of what constitutes an OPERABLE RHR shutdown subsystem are proposed to be relocated to the Bases. The Bases will indicate that an OPERABLE RHR shutdown cooling subsystem consists of an OPERABLE pump, heat exchanger, service water providing cooling to the heat exchanger, and the associated piping and valves. The details for subsystem OPERABILITY are not necessary in ITS 3.4.10. The definition of OPERABILITY suffices. Therefore, the relocated details are not required to be in the ITS to provide adequate protection of the public health and safety. Changes to the Bases will be controlled by the provisions of the proposed Bases Control Program described in Chapter 5 of the ITS.

TECHNICAL CHANGES - LESS RESTRICTIVE (continued)

LA.2 The detail of the method in CTS 4.4.9.2 of verifying operation of the RHR shutdown cooling subsystem (circulating reactor coolant) is proposed to be relocated to the Bases. This detail is not necessary for assuring the RHR shutdown cooling subsystem is in operation. Proposed SR 3.4.10.1 requires verification an RHR shutdown cooling subsystem is operating and is adequate to ensure an RHR shutdown cooling subsystem is circulating reactor coolant. Therefore, the relocated detail is not required to be in the ITS to provide adequate protection of the public health and safety. Changes to the Bases will be controlled by the provisions of the proposed Bases Control Program described in Chapter 5 of the ITS.

"Specific"

L.1 CTS 3.4.9.2, ACTION b and CTS 4.4.9.2 are proposed to be revised to allow a recirculation pump to be in operation in lieu of a shutdown cooling mode loop of the RHR System. This proposed change provides the "alternate method" as cited in CTS 4.4.9.2 and, therefore, ensures consistency between the SR and LCO. The recirculation pump provides the necessary forced flow through the core during shutdown; therefore, the intent of the CTS LCO continues to be met. Though this proposed change is less restrictive for LaSalle, its contents are consistent with the current Technical Specifications for Washington Nuclear Plant - Unit 2 and Susquehannah Steam Electric Station - Units 1 and 2 and it continues to ensure the decay heat removal function is satisfied.

L.2 CTS 3.4.9.2 footnote # allows one RHR shutdown cooling loop to be inoperable for 2 hours for surveillance testing, provided the other loop is Operable and in operation. CTS 3.4.9.2 footnote ** allows the RHR shutdown cooling pump to be removed from operation for up to 2 hours per 8 hour period, provided the other loop is Operable. The requirements in CTS footnotes # and ** (ITS 3.4.10 LCO Notes 1 and 2) are proposed to be changed to delete the "provided" requirements. The allowances of the Notes may be required even when no RHR shutdown cooling loop remains Operable or in operation. Some Surveillances result in the inoperability of both RHR shutdown cooling loops (e.g., Surveillances on the common suction line valves). With one or more RHR shutdown cooling loops inoperable, CTS 3.4.9.2 Action a (ITS 3.4.10 ACTION A) requires an alternate method capable of decay heat removal to be established within 1 hour for each inoperable RHR shutdown cooling loop and, with no RHR or recirculation pump in operation, CTS 3.4.9.2 Action b (ITS 3.4.10 ACTION B) requires establishment of reactor coolant recirculation by an alternate method within 1 hour. If acceptable alternatives are available for decay heat removal (i.e., complying with the Actions), the temporary allowances of the Notes should apply since the alternate methods must be capable of providing adequate decay heat removal.

TECHNICAL CHANGES - LESS RESTRICTIVE (continued)

L.3 CTS 3.4.9.2 footnote ## allows the RHR shutdown cooling mode loop to be removed from operation during hydrostatic testing. In the ITS (ITS 3.4.10 LCO Note 1), this allowance has been changed to also allow the system to be inoperable during hydrostatic testing. The RHR Shutdown Cooling System is in fact inoperable during hydrostatic testing since the system is not capable of circulating reactor coolant. The RHR Shutdown Cooling System is automatically isolated above the RHR cut-in permissive pressure. This isolation is necessary since the RHR Shutdown Cooling System is not designed to operate at the Reactor Coolant System pressure achieved during hydrostatic testing. Hydrostatic tests of the Reactor Coolant System are performed after each refueling prior to the reactor criticality. Under these conditions, decay heat levels are low. Adequate reactor coolant circulation will be achieved by operation of a reactor recirculation pump and other systems are available to control reactor coolant temperature. Therefore, this change is acceptable.

RELOCATED SPECIFICATIONS

None

A.1

REACTOR COOLANT SYSTEM

3/4.4.6 PRESSURE/TEMPERATURE LIMITS

REACTOR COOLANT SYSTEM

LIMITING CONDITION FOR OPERATION

A.1

within limits

LCO 3.4.11 3.4.6.1 The reactor coolant system temperature and pressure shall be limited in accordance with the limit lines shown on Figures 3.4.6.1-1 and 3.4.6.1-1a (1) curves A for hydrostatic or leak testing; (2) curves B for heatup by non-nuclear means, cooldown following a nuclear shutdown and low power PHYSICS TESTS; and (3) curves C for operations with a critical core other than low power PHYSICS TESTS, with: Figure 3.4.6.1-1b Figure 3.4.6.1-1a

SR 3.4.11.1

SR 3.4.11.2

SR 3.4.11.1
- a. A maximum heatup of 100°F in any one hour period,
- b. A maximum cooldown of 100°F in any one hour period,
- c. A maximum temperature change of less than or equal to 20°F in any one hour period during inservice hydrostatic and leak testing operations above the heatup and cooldown limit curves, and

SR 3.4.11.5 SR 3.4.11.6 SR 3.4.11.7
- d. The reactor vessel flange and head flange temperature greater than or equal to 80°F when reactor vessel head bolting studs are under tension. 72

<LAR297> A.9

APPLICABILITY: At all times.*

ACTION: add proposed Conditions A and C Notes A.2 A.3

M.1 add Required Actions A.2 and C.2 Completion Times

ACTIONS A & C With any of the above limits exceeded, restore the temperature and/or pressure to within the limits within 30 minutes; perform an engineering evaluation to determine the effects of the out-of-limit condition on the structural integrity of the reactor coolant system; determine that the reactor coolant system remains acceptable for continued operations or be in at least HOT SHUTDOWN

ACTION B within 12 hours and in COLD SHUTDOWN within the following 24 hours.

LA.2

SURVEILLANCE REQUIREMENTS

SR 3.4.11.1 4.4.6.1.1 During system heatup, cooldown and inservice leak and hydrostatic testing operations, the reactor coolant system temperature and pressure shall be determined to be within the above required heatup and cooldown limits and to the right of the limit lines of Figures 3.4.6.1-1 and 3.4.6.1-1a curves A or B, as applicable, at least once per 30 minutes. , 3.4.6.1-1a, and 3.4.6.1-1b

LA.1

~~*During shutdown conditions for hydrostatic or leak testing or heatup by nonnuclear means the average coolant temperature limit of Table 1.2 for Cold Shutdown and Hot Shutdown may be increased to 212°F.~~

<LAR297> A.9

A.1

REACTOR COOLANT SYSTEM

SURVEILLANCE REQUIREMENTS (Continued)

within LA.1

SR 3.4.11.2 4.4.6.1.2 The reactor coolant system temperature and pressure shall be determined to be ~~to the right of~~ the criticality limit line of Figure~~s 3.4.6.1-1~~ 3.4.6.1-1b ~~and 3.4.6.1-1a curves C~~ within 15 minutes prior to the withdrawal of control rods to bring the reactor to criticality. <LAR297> A9

~~4.4.6.1.3 The reactor vessel material specimens shall be removed and examined to determine reactor pressure vessel fluence as a function of time and THERMAL POWER as required by 10 CFR Part 50, Appendix H in accordance with the schedule in Table 4.4.6.1.3-1. The results of these fluence determinations shall be used to update the curves of Figures 3.4.6.1-1 and 3.4.6.1-1a.~~ A.4 <LAR297> A9

, and 3.4.6.1-1a

SR 3.4.11.5
SR 3.4.11.6
SR 3.4.11.7 4.4.6.1.4 The reactor vessel flange and head flange temperature shall be verified to be greater than or equal to ~~80~~ 72°F:

a. In OPERATIONAL CONDITION 4 when the reactor coolant temperature is:

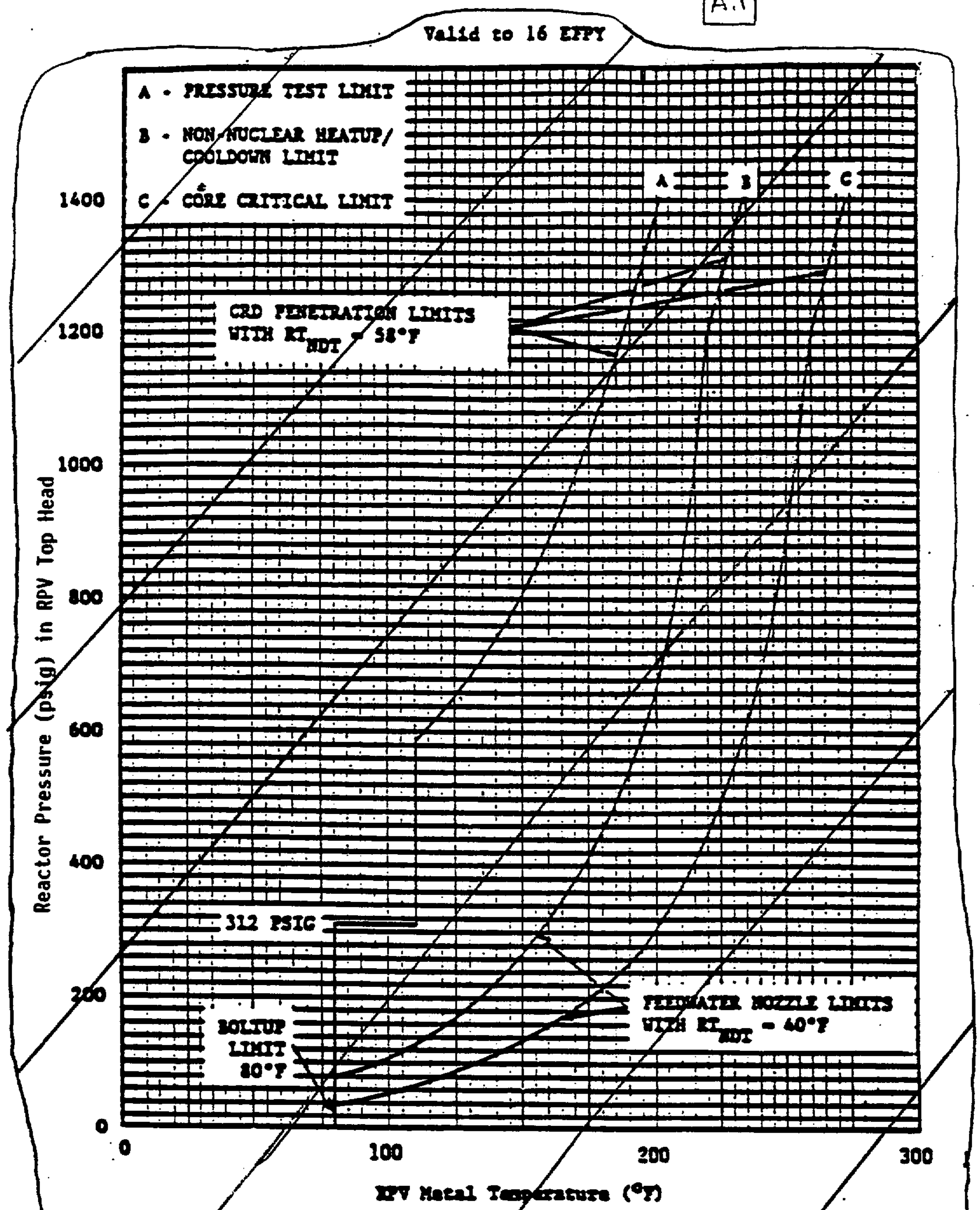
SR 3.4.11.7 92 1. ≤ ~~100~~°F, at least once per 12 hours. ← add proposed SR 3.4.11.7 Note A.5

SR 3.4.11.6 77 2. ≤ ~~85~~°F, at least once per 30 minutes. ← add proposed SR 3.4.11.6 Note

SR 3.4.11.5 b. ~~Within 30 minutes prior to and~~ at least once per 30 minutes during tensioning of the reactor vessel head bolting studs. A.6

<LAR297> A9

A.1

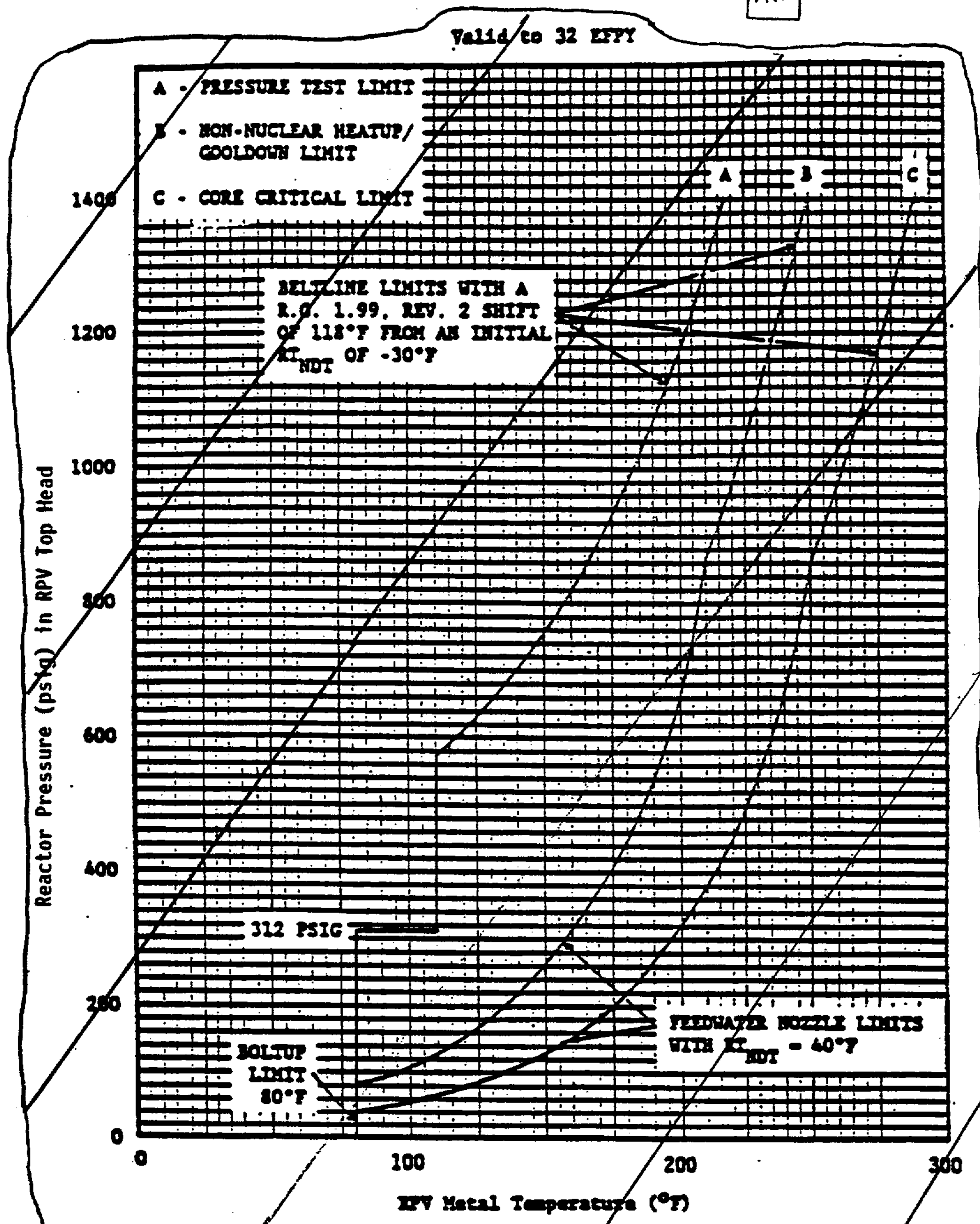


Minimum Reactor Vessel Metal Temperature vs
Reactor Vessel Pressure

Figure 3.4.6.1-1

<LAR297> A.9

A.1

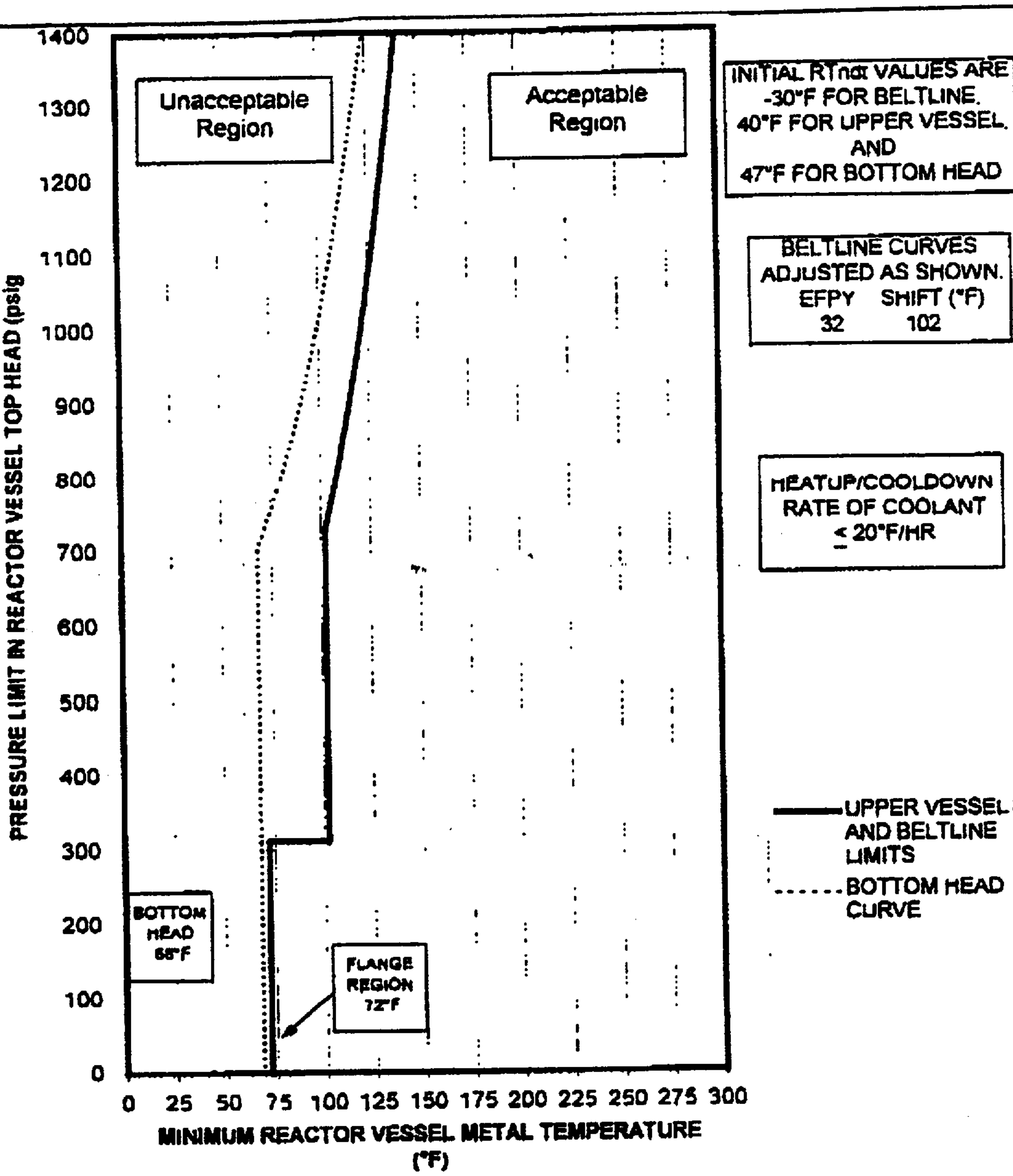


Minimum Reactor Vessel metal Temperature vs.
Reactor Vessel Pressure

Figure 3.4.6.1-1a

Replace with Figures 3.4.6.1-1, 3.4.6.1-1a, and 3.4.6.1-1b <LAR 247> A.9

A.1



P-T Curve for Hydrostatic or Leak Testing

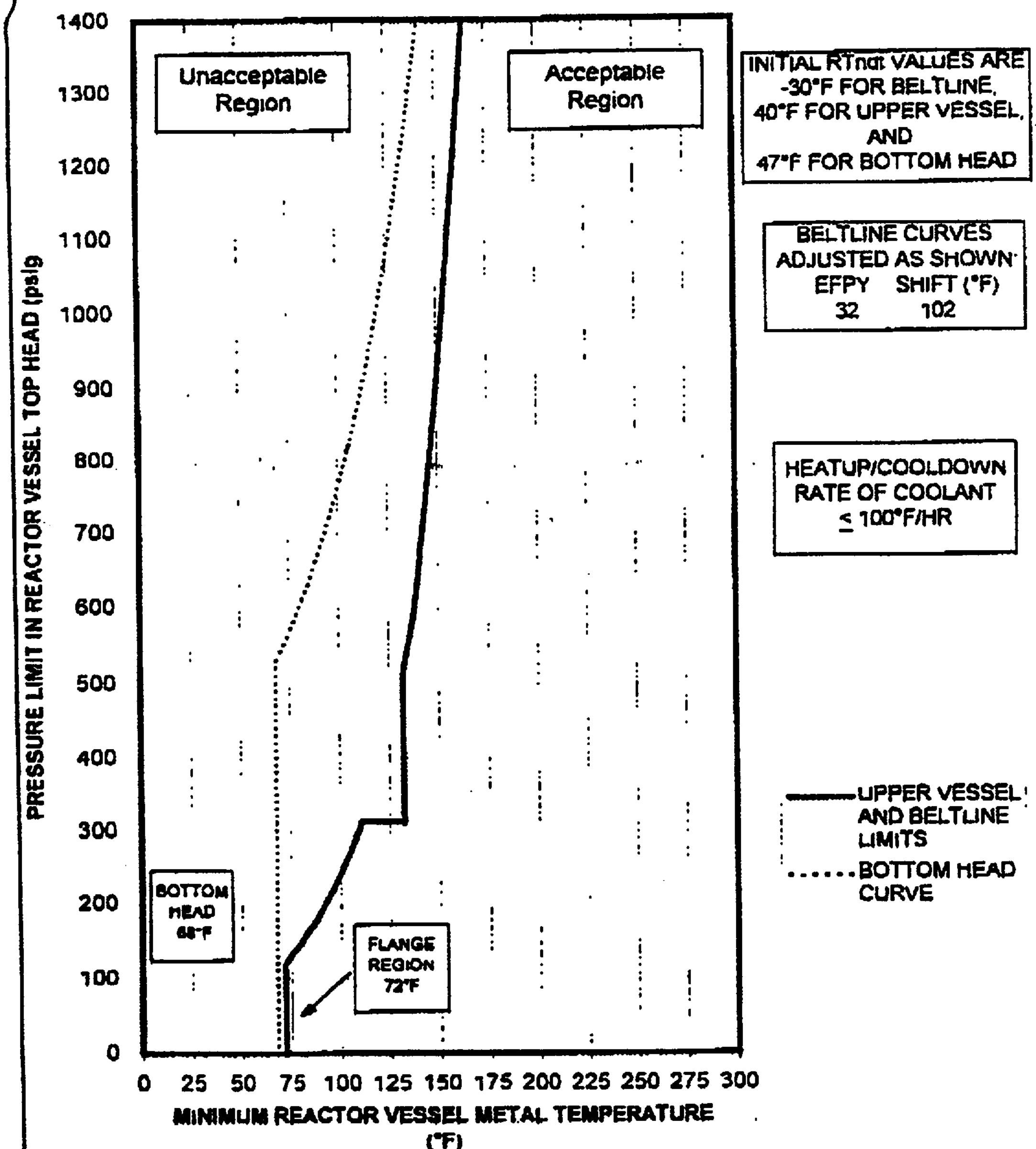
Minimum Reactor Vessel Metal Temperature vs. Reactor Vessel Pressure

Valid to 32 EFPY

Figure 3.4.6.1-1

<LAR 297> A.9

A.1



P-T Curve for Heatup by Non-Nuclear Means, Cooldown Following a Nuclear Shutdown and Low Power PHYSICS TESTS

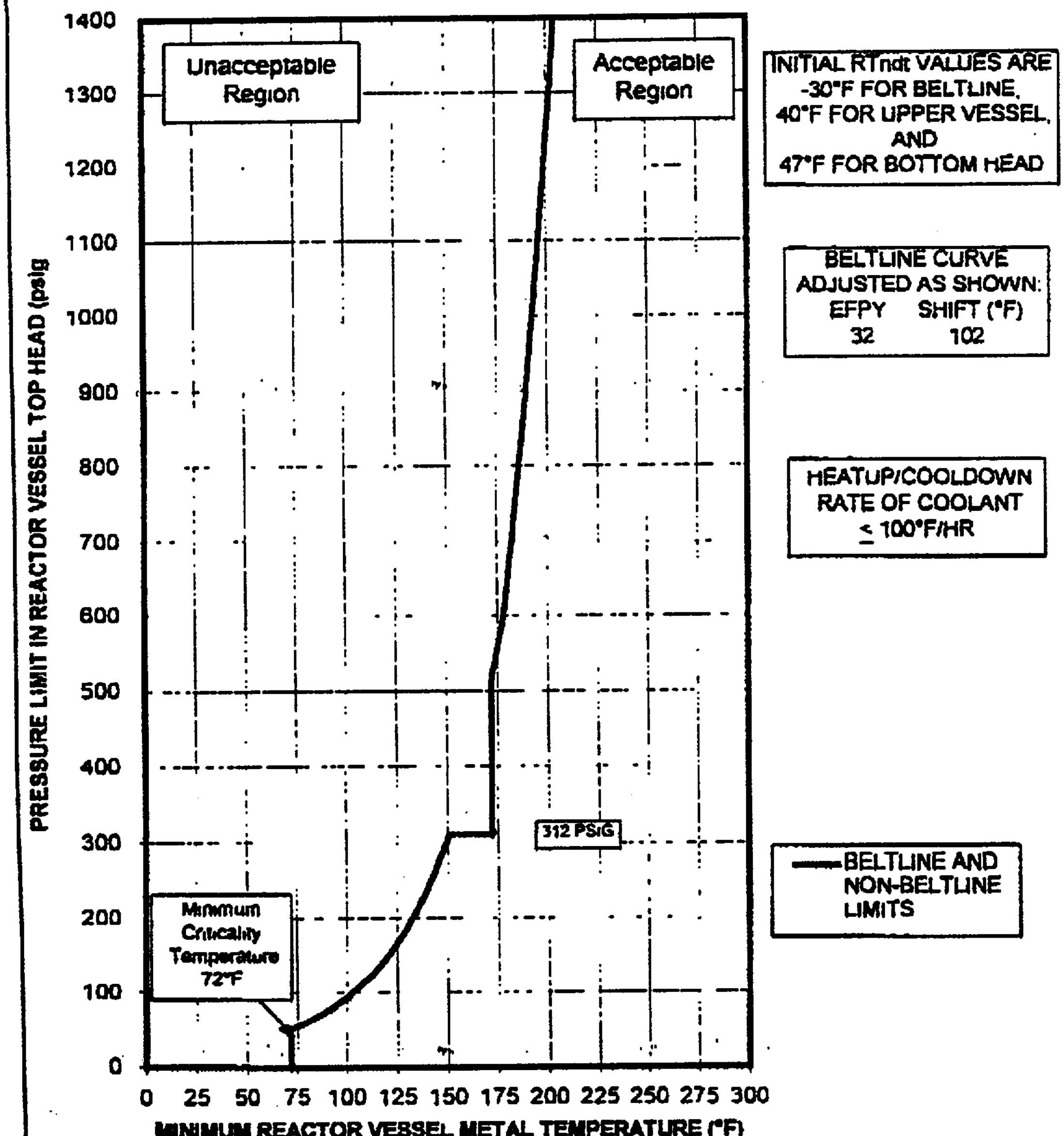
Minimum Reactor Vessel Metal Temperature vs. Reactor Vessel Pressure

Valid to 32 EFPY

Figure 3 4 6 1-1a

<LAR297> A9

A.1



P-T Curve for Operation with a Critical Core other than low power PHYSICS TESTS

Minimum Reactor Vessel Metal Temperature vs.
Reactor Vessel Pressure

Valid to 32 EFPY

Figure 3.4.6.1-1b

<LAR297> A9

ITS 3.4.11

A.1

Table 4.4.6.1.3-1

Reactor Vessel Material Surveillance Program Withdrawal Schedule

Specimen holder	Vessel location	Lead factor	Withdrawal time (Effective Full Power Years)
117C4936G010	300°	0.6	6
117C4936G011	120°	0.6	15
117C4936G012	30°	0.6	Spare
Neutron Dosimeter	30°		1st Refuel Outage

<LA2297> A.9

REACTOR COOLANT SYSTEM

IDLE RECIRCULATION LOOP STARTUP

LIMITING CONDITION FOR OPERATION

LA.3

LCO 3.4.11
SR 3.4.11.3

3.4.1.4 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 less than or equal to 145°F, and:

A.7

SR 3.4.11.4

a. 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 less than or equal to 50°F, or

b. When only one loop has been idle, unless the temperature differential between the reactor coolant within the idle ~~and operating~~ recirculation loops is less than or equal to 50°F ~~and the operating loop flow rate is less than or equal to 50% of rated loop flow~~.

A.8

LA.4

Note to SR 3.4.11.3 and SR 3.4.11.4

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2, 3, and 4.

ACTION:

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

add proposed ACTIONS A, B, and C

M.2

SURVEILLANCE REQUIREMENTS

SR 3.4.11.3
SR 3.4.11.4

4.4.1.4 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.

LA.4

REACTOR COOLANT SYSTEM

A.1

3/4.4.6 PRESSURE/TEMPERATURE LIMITS

REACTOR COOLANT SYSTEM

LIMITING CONDITION FOR OPERATION

A.1

within limits

LCO 3.4.11 3.4.6.1 The reactor coolant system temperature and pressure shall be limited in accordance with the limit lines shown on Figures 3.4.6.1-1 ~~and 3.4.6.1-1a:~~
SR 3.4.11.1 ~~(1) curves A~~ for hydrostatic or leak testing; ~~(2) curves B~~ for heatup by non-nuclear means, cooldown following a nuclear shutdown and low power PHYSICS
SR 3.4.11.2 TESTS; and ~~(3) curves C~~ for operations with a critical core other than low power PHYSICS TESTS, with:

Figure 3.4.6.1-1b

Figure 3.4.6.1-1a

SR 3.4.11.1

a. A maximum heatup of 100°F in any one hour period,

b. A maximum cooldown of 100°F in any one hour period,

c. A maximum temperature change of less than or equal to 20°F in any one hour period during inservice hydrostatic and leak testing operations above the heatup and cooldown limit curves, and

LAR297 A9

SR 3.4.11.5
SR 3.4.11.6
SR 3.4.11.7

d. The reactor vessel flange and head flange temperature greater than or equal to 86°F when reactor vessel head bolting studs are under tension.

APPLICABILITY: At all times.*

ACTION: add proposed Conditions A and C Notes A.2

A.3

M.1

add Required Actions A.2 and C.2 Completion times

ACTIONS A and C With any of the above limits exceeded, restore the temperature and/or pressure to within the limits within 30 minutes, ~~perform an engineering evaluation to determine the effects of the out-of-limit condition on the structural integrity of the reactor coolant system;~~ determine that the reactor coolant system remains acceptable for continued operations or be in at least HOT SHUTDOWN
ACTION B within 12 hours and in COLD SHUTDOWN within the following 24 hours.

LA.2

SURVEILLANCE REQUIREMENTS

SR 3.4.11.1 4.4.6.1.1 During system heatup, cooldown and inservice leak and hydrostatic testing operations, the reactor coolant system temperature and pressure shall be determined to be within the above required heatup and cooldown limits ~~and to the right of the limit lines~~ of Figures 3.4.6.1-1 ~~and 3.4.6.1-1a curves A or B~~, as applicable, at least once per 30 minutes. , 3.4.6.1-1a, and 3.4.6.1-1b

LA.1

~~*During shutdown conditions for hydrostatic or leak testing or heatup by nonnuclear means, the average coolant temperature limit of Table 1.2 for Cold Shutdown and Hot Shutdown may be increased to 212°F.~~

LAR297 A9

A.1

REACTOR COOLANT SYSTEM

SURVEILLANCE REQUIREMENTS (Continued)

within LA.1

SR 3.4.11.2 4.4.6.1.2 The reactor coolant system temperature and pressure shall be determined to be ~~to the right of~~ the criticality limit line of Figures ~~3.4.6.1-1 and 3.4.6.1-1a curves C~~ within 15 minutes prior to the withdrawal of control rods to bring the reactor to criticality. 3.4.6.1-1b

<LAR297> A.9

~~4.4.6.1.3 The reactor vessel material specimens shall be removed and examined to determine reactor pressure vessel fluence as a function of time and THERMAL POWER as required by 10 CFR Part 50, Appendix H in accordance with the schedule in Table 4.4.6.1.3-1. The results of these fluence determinations shall be used to update the curves of Figures 3.4.6.1-1 and 3.4.6.1-1a.~~ , and 3.4.6.1-1b A.4

SR 3.4.11.5
SR 3.4.11.6
SR 3.4.11.7
4.4.6.1.4 The reactor vessel flange and head flange temperature shall be verified to be greater than or equal to 86°F:

a. In OPERATIONAL CONDITION 4 when the reactor coolant temperature is:

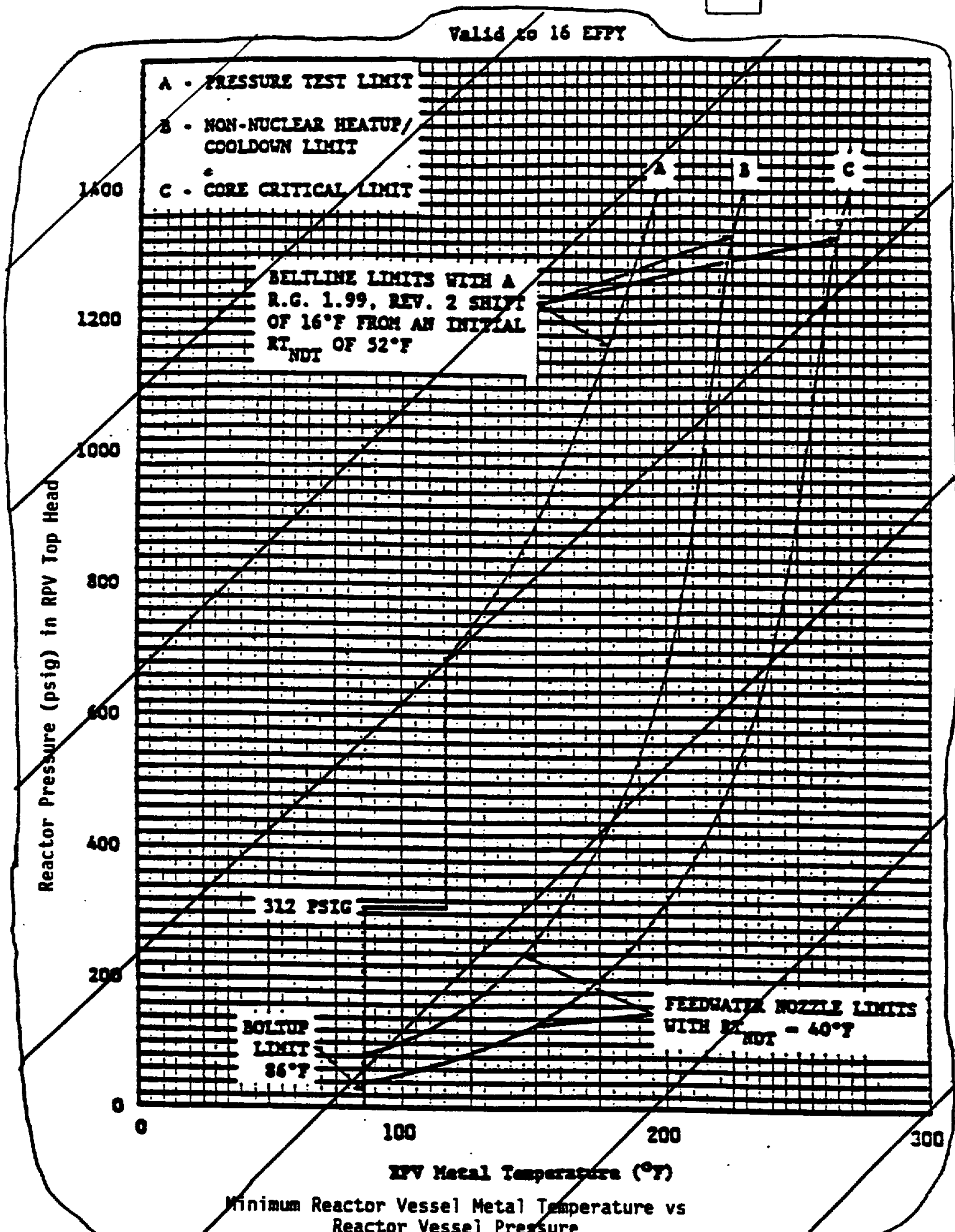
SR 3.4.11.7 1. ≤ 106°F, at least once per 12 hours. add proposed SR 3.4.11.7 Note A.5

SR 3.4.11.6 2. ≤ 91°F, at least once per 30 minutes. add proposed SR 3.4.11.6 Note

SR 3.4.11.5 b. ~~Within 30 minutes prior to and~~ at least once per 30 minutes during tensioning of the reactor vessel head bolting studs. A.6

A.1

ITS 3.4.11



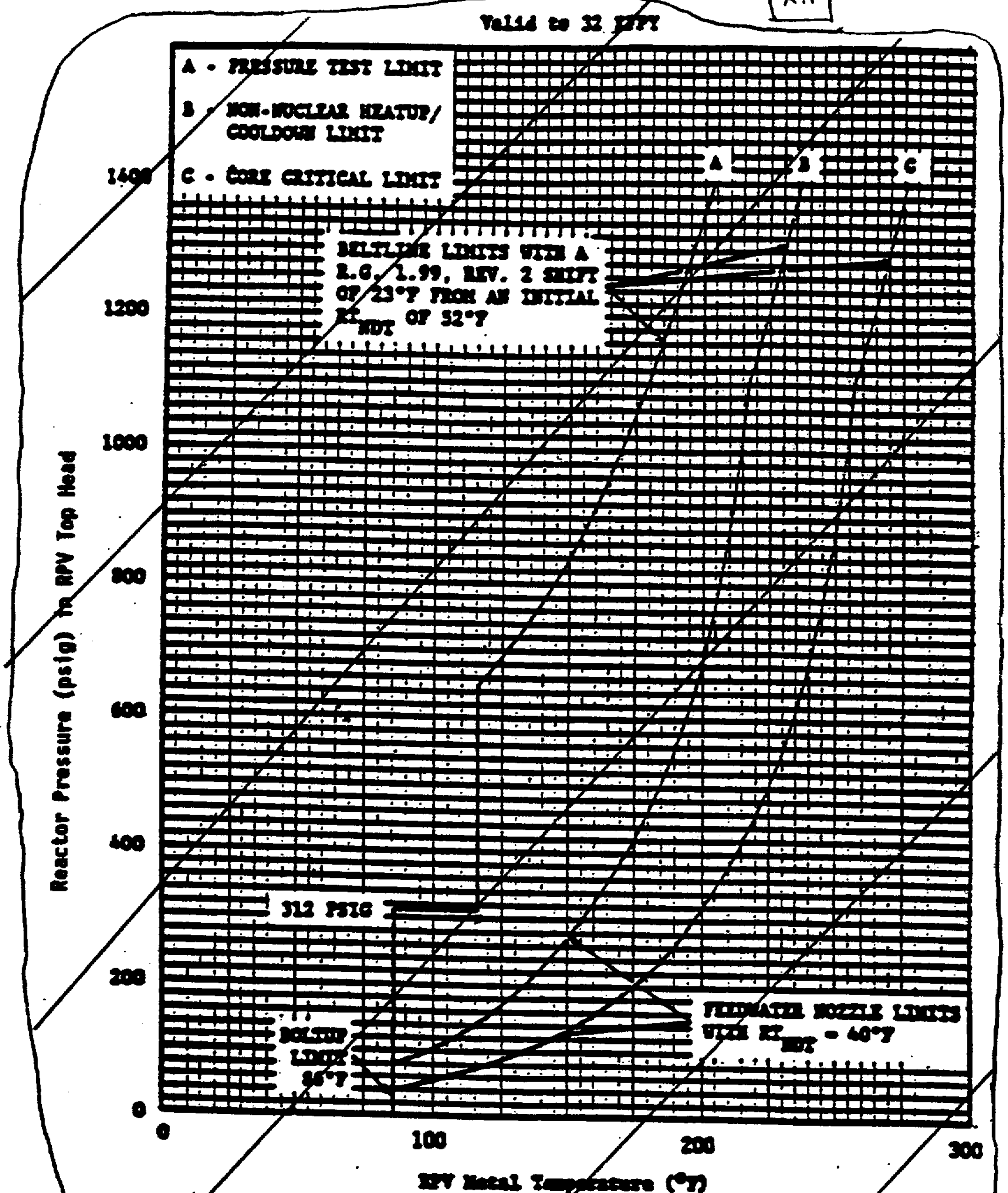
Minimum Reactor Vessel Metal Temperature vs Reactor Vessel Pressure

Figure 3.4.6.1-1

<LAR297>

A9

A.1



Minimum Reactor Vessel Metal Temperature vs. Reactor Vessel Pressure

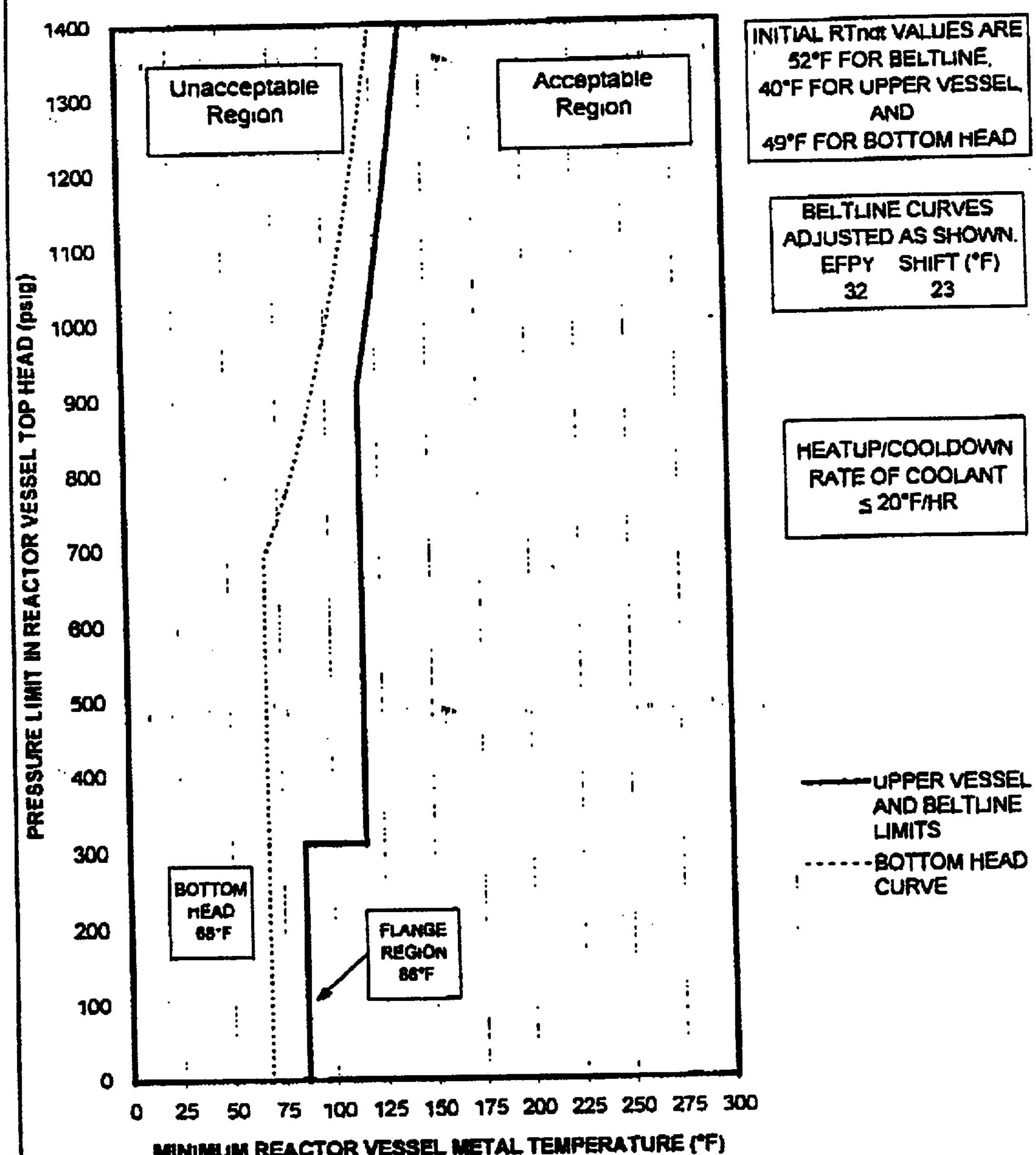
Figure 3.4.6.1-1a

Replace with Figures 3.4.6.1-4, 3.4.6.1-5, and 3.4.6.1-6

<LAR297> A.9

A.1

ITS 3.4.11



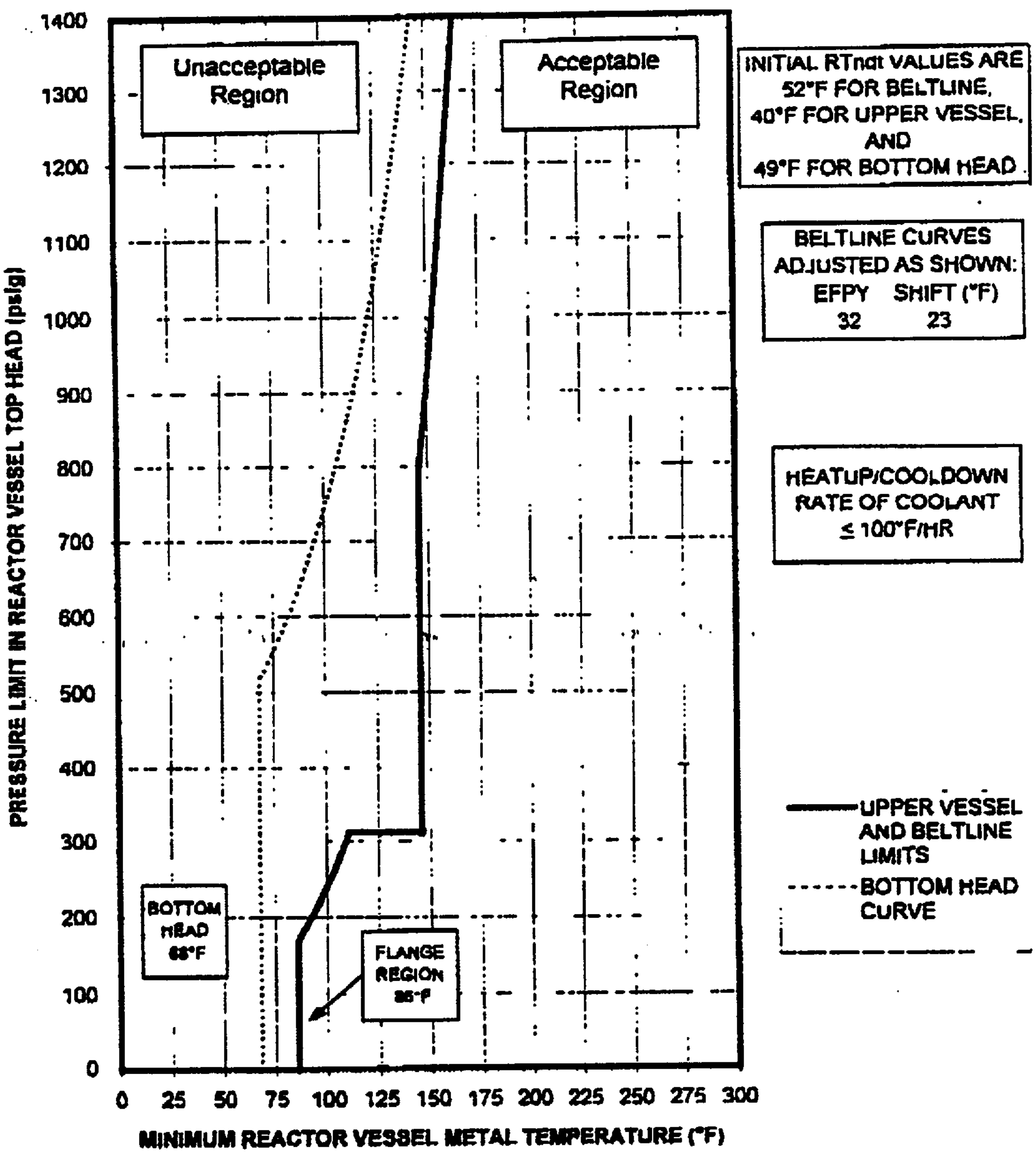
P-T Curve for Hydrostatic or Leak Testing

Minimum Reactor Vessel Metal Temperature vs.
Reactor Vessel Pressure

Valid to 32 EFPY

Figure 3.4.6.1-1

<LAR297> A.9



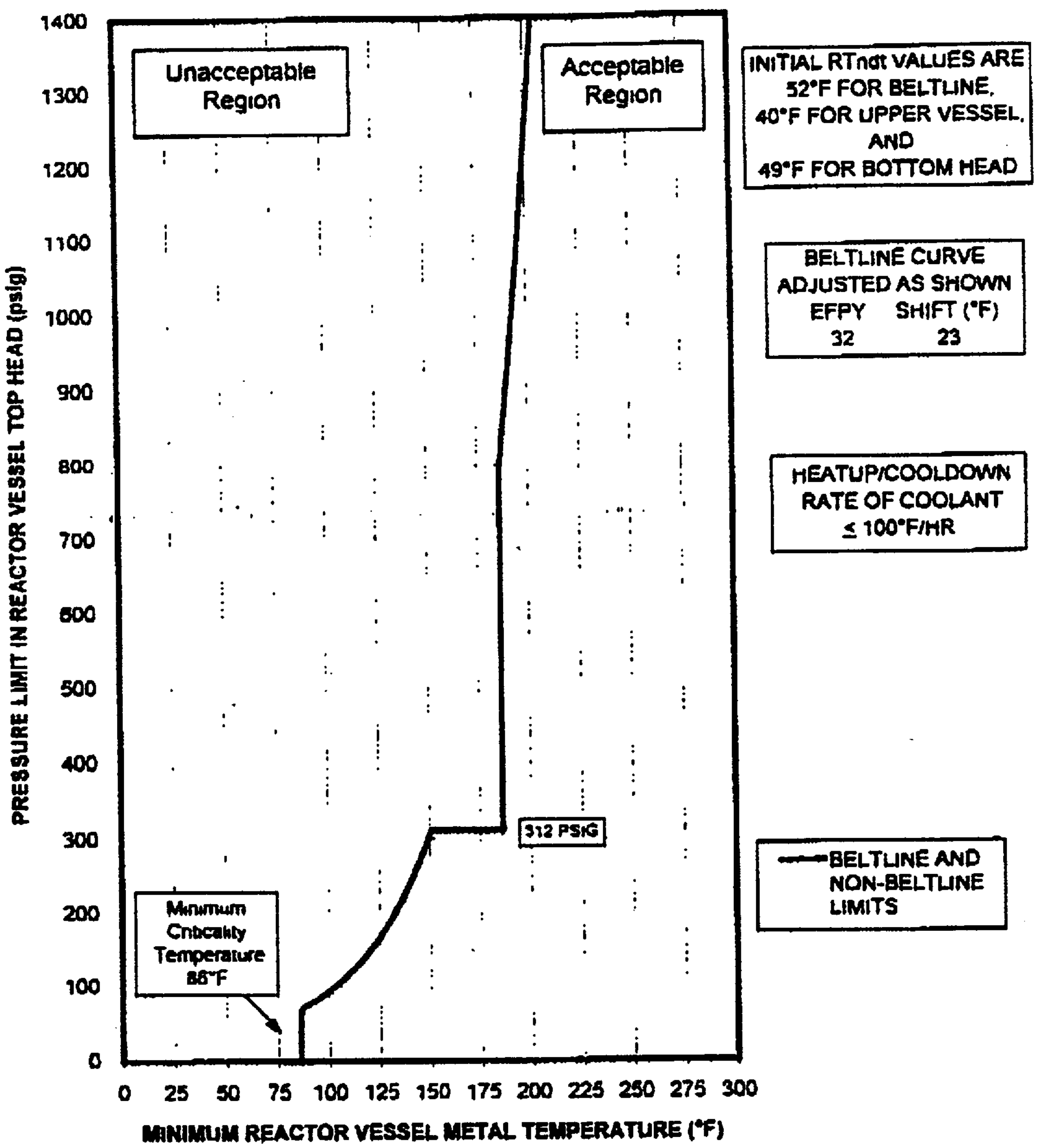
P-T Curve for Heatup by Non-Nuclear Means, Cooldown Following a Nuclear Shutdown and Low Power PHYSICS TESTS

Minimum Reactor Vessel Metal Temperature vs. Reactor Vessel Pressure

Valid to 32 EFPY

Figure 3.4.6.1-1a

<LAR297> A.9



P-T Curve for Operation with a Critical Core other than low power PHYSICS TESTS

Minimum Reactor Vessel Metal Temperature vs.
Reactor Vessel Pressure

Valid to 32 EFPY

Figure 3.4.6.1-1b

<LAR297> A.9

A.1

TABLE 4.4.6.1.3-1

REACTOR VESSEL MATERIAL SURVEILLANCE PROGRAM-WITHDRAWAL SCHEDULE

SPECIMEN HOLDER*	VESSEL LOCATION	LEAD FACTOR	WITHDRAWAL TIME (EFFECTIVE FULL POWER YEARS)
Capsule 1	300°	0.6	6
Capsule 2	120°	0.6	15
Capsule 3	30°	0.6	Spare
Neutron Dosimeter	30°	-	1st Refueling Outage

*Each capsule includes an Fe, Ni, and Cu flux wire. The neutron dosimeter contains three Cu and three Fe flux wires.

<LAR297> A.9

A.1 ITS 3.4.11

REACTOR COOLANT SYSTEM

IDLE RECIRCULATION LOOP STARTUP

LIMITING CONDITION FOR OPERATION

LA.3

LCO 3.4.11
SR 3.4.11.3

3.4.1.4 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 less than or equal to 145°F, and:

A.7

SR 3.4.11.4

a. 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 less than or equal to 50°F, or

b. When only one loop has been idle, unless the temperature differential between the reactor coolant within the idle ~~and operating~~ recirculation loops is less than or equal to 50°F ~~and the operating loop flow rate is less than or equal to 50% of rated loop flow~~. A.8

LA.4

Note to SR 3.4.11.3 and SR 3.4.11.4

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2, 3, and 4.

ACTION:

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

add proposed ACTIONS A, B, and C

M.2

SURVEILLANCE REQUIREMENTS

SR 3.4.11.3
SR 3.4.11.4

4.4.1.4 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.

LA.4

ADMINISTRATIVE

A.1 In the conversion of the LaSalle 1 and 2 current Technical Specifications (CTS) to the proposed plant specific Improved Technical Specifications (ITS), certain wording preferences or conventions are adopted that do not result in technical changes (either actual or interpretational). Editorial changes, reformatting, and revised numbering are adopted to make the ITS consistent with the BWR Standard Technical Specifications, NUREG-1434, Rev. 1 (i.e., the Improved Standard Technical Specifications (ISTS)).

A.2 The CTS 3.4.6.1 ACTION to "determine that the reactor coolant system remains acceptable for continued operation" is proposed to be clarified so that no confusion exists as to the requirements once the temperature and pressure are restored to within limits. The current intent of the ACTION is appropriately presented in ITS 3.4.11 Conditions A and C Notes. These Notes state that the determination of the acceptability of the RCS for continued operation must be completed any time the requirements of the LCO are not met. This interpretation of the intent is supported by the BWR Standard Technical Specifications, NUREG-1434, Rev. 1. Because this is an enhanced presentation of the existing intent, the proposed change is administrative.

A.3 The CTS 3.4.6.1 ACTION to "restore...within 30 minutes" is proposed to be revised to "initiate action to restore ...Immediately" for conditions other than MODES 1, 2, and 3. The existing ACTION would appear to provide a half hour in which pressure and temperature requirements could exceed the limits, even if capable of being returned to within limits. Also, if the parameters are incapable of being restored to within the limits within 30 minutes, the existing ACTION would appear to result in the requirement for an LER. The intent of the ACTION is believed to be more appropriately presented in ITS 3.4.11 Required Action C.1. This interpretation of the intent is supported by the BWR Standard Technical Specifications, NUREG-1434, Rev. 1. Because this is an enhanced presentation of the existing intent, the proposed change is administrative.

A.4 CTS 4.4.6.1.3 is a duplication of the regulations found in 10 CFR 50 Appendix H. These regulations require licensee compliance and can not be revised by the licensee without prior NRC approval. Therefore, these details of the regulations within the Technical Specifications are repetitious. Furthermore, approved exemptions to the regulations, and exceptions presented within the regulations themselves, are also details which are adequately presented without repeating the details within the Technical Specifications. Therefore, deleting the requirement to meet the requirements of 10 CFR 50 Appendix H, in accordance with the schedule in Table 4.4.6.1.3-1, and eliminating the Technical Specification details that are also found in Appendix H, is considered a presentation preference which is administrative. The actual table schedule is addressed in Discussion of Change A.9 below.

ADMINISTRATIVE (continued)

A.5 CTS 4.4.6.1.4.a requires periodic verification that reactor vessel and head flange temperatures are ≥ 72°F for Unit 1 and ≥ 86°F for Unit 2. The Frequency of this verification changes based on reactor coolant system temperature. Notes have been provided in proposed SR 3.4.11.6 and 3.4.11.7 to clarify the current intent in CTS 4.4.6.1.4.a of allowing entry into the applicable conditions (i.e., when ≤ 92°F for Unit 1 and ≤ 106°F for Unit 2, and ≤ 77°F for Unit 1 and ≤ 91°F for Unit 2) without having performed these Surveillance Requirements. Since this requirement is currently only performed during the specified conditions (i.e., when ≤ 92°F for Unit 1 and ≤ 106°F for Unit 2, and ≤ 77°F for Unit 1 and ≤ 91°F for Unit 2), these changes (the two Notes) are considered administrative.

A.6 The CTS 4.4.6.1.4.b requirement to verify the reactor vessel and head flange temperatures within 30 minutes prior to tensioning of the head bolting studs has been deleted. This requirement is duplicative of CTS 4.0.1 and proposed SR 3.0.1, which require the Surveillance to be current when in the applicable Mode or condition. CTS 4.0.3 and proposed SR 3.0.1 also state that failure to meet the Surveillance constitutes failure to meet the LCO, which would then require the ACTIONS of the LCO to be taken. The ACTIONS for CTS 3.4.6.1 (ITS 3.4.11 ACTION C) requires action to be taken to restore the limit. Therefore, this effectively ensures that the Applicability of this SR (as stated in the Note to the SR) is not entered with CTS 4.4.6.1.4.b (proposed SR 3.4.11.7) not current. Therefore, this change is considered administrative.

A.7 The CTS 3.4.1.4 requirements have been combined into the RCS P/T Limits Specification, with the words "and the recirculation pump starting temperature requirements" added to the ITS 3.4.11 LCO statement. The actual description of the requirements and the limits are found in proposed SR 3.4.11.3 and SR 3.4.11.4. As such, this change is administrative.

A.8 Thermal stresses on vessel components are dependent upon the temperature difference between the idle loop coolant and the RPV coolant. CTS 3.4.1.4.a and 4.4.1.4 (proposed SR 3.4.11.4) ensure the temperature difference between the idle loop and the RPV coolant is acceptable. The CTS 3.4.1.4.b requirement to monitor the temperature difference between an idle loop and an operating loop are unnecessary and have been deleted since they are redundant to the loop-to-coolant requirement of CTS 3.4.1.4.a and 4.4.1.4 (proposed SR 3.4.11.4). However, the loop-to-coolant temperature check may use the operating loop temperature as representative of "coolant temperature."

ADMINISTRATIVE (continued)

A.9 These changes to CTS 3/4.4.6.1 are provided in the LaSalle ITS consistent with the Technical Specifications Change Request submitted to the NRC for approval per ComEd letter dated February 29, 2000. The changes identified revise the heatup, cooldown, and inservice test limitations for the reactor pressure vessel of each unit to a maximum of 32 Effective Full Power Years. The proposed changes rely on recently approved American Society of Mechanical Engineers methodology for determining allowable pressure and temperature limits. A similar Technical Specifications amendment was recently issued for Duke Energy, Oconee Nuclear Station. As such, this change is administrative.

TECHNICAL CHANGES - MORE RESTRICTIVE

M.1 The ACTION of CTS 3.4.6.1 does not specify a Completion Time for completion of the engineering evaluation and determination. A specific Completion Time for the engineering evaluation and determination in the CTS 3.4.6.1 ACTION is provided in ITS 3.4.11 Required Actions A.2 and C.2. The ITS 3.4.11 Required Action A.2 Completion Time of 72 hours is considered reasonable for operation in MODES 1, 2, and 3 because the P/T limits are based on very conservative flaw assumptions and large factors of safety. In conditions other than MODES 1, 2, and 3, the ITS 3.4.11 Required Action C.2 Completion Time of prior to entering MODE 2 or 3 would prevent entry in the operating MODES which is consistent with the current LCO 3.0.4. This change is an additional restriction on plant operation.

M.2 The CTS 3.4.1.4 ACTION required to be taken when a recirculation pump is started without having met the temperature requirements has been changed. Currently, the CTS 3.4.1.4 ACTION only states to suspend the startup of a recirculation loop. This however, does not provide an action if the loop is already operating. ITS 3.4.11 ACTIONS A, B, and C are added which, in this condition, would require an engineering evaluation to be performed to ensure continued operation is acceptable. This is an additional restriction on plant operation necessary to ensure the RCS is acceptable for continued operation.

TECHNICAL CHANGES - LESS RESTRICTIVE

"Generic"

LA.1 The specific requirements in CTS 4.4.6.1.1 and CTS 4.4.6.1.2 that operation be to the right of the limits lines of Figures 3.4.6.1-1 and 3.4.6.1-1a are proposed to be relocated to the Bases. These details are not necessary to ensure the P/T limits are met. The requirements to maintain the P/T limits in accordance with the Figures are still maintained in ITS 3.4.11, SR 3.4.11.1, and SR 3.4.11.2. Therefore, the relocated requirements are not required to be in the ITS to provide adequate protection of the public health and safety. Changes to the Bases will be controlled by the provisions of the proposed Bases Control Program described in Chapter 5 of the ITS.

LA.2 The detail in the ACTION of CTS 3.4.6.1 to perform an engineering evaluation to determine the effects of the out-of-limit condition on the structural integrity of the reactor coolant system is proposed to be relocated to the Bases. The requirements in proposed ITS 3.4.11 Required Actions A.2 and C.2 to determine RCS is acceptable for continued operation and the Condition A and C Note that the applicable action shall be completed if this Condition is entered ensures the current requirement is met. In addition, the Bases for these Required Actions indicates that an engineering evaluation shall be performed. As such, the relocated details are not required to be in the ITS to provide adequate protection of the public health and safety. Changes to the Bases will be controlled by the provisions of the proposed Bases Control Program described in Chapter 5 of the ITS.

LA.3 CTS 3.4.1.4 requires that an idle recirculation loop 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. The detail on where the temperature is monitored (e.g., space or line) to evaluate the temperature difference is proposed to be relocated to the Bases. The requirement in proposed ITS SR 3.4.11.3 to verify the difference between the bottom head coolant temperature and the reactor pressure (RPV) coolant temperature is sufficient to ensure that the differential temperature is met prior to a startup of a recirculation pump. The Bases for SR 3.4.11.3 indicates an acceptable method for evaluating the limit. As such, the relocated details are not required to be in the ITS to provide adequate protection of the public health and safety. Changes to the Bases will be controlled by the provisions of the proposed Bases Control Program described in Chapter 5 of the ITS.

TECHNICAL CHANGES - LESS RESTRICTIVE (continued)

LA.4 The details of CTS 3.4.1.4.b (and its associated Action) and CTS 4.4.1.4 relating to operational limits (maximum operating loop flow rate) during a return to two recirculation pump operation from single recirculation loop operation are proposed to be relocated to the UFSAR. The single loop flow rate is considered an operational limit since it is not directly related to the ability of the system to perform its safety analysis functions. The flow rate is limited only to restrict reactor vessel internals vibration to within acceptable limits during restart of the second pump. These requirements are oriented toward maintaining long term OPERABILITY of the recirculation loops and do not necessarily have an immediate impact on their OPERABILITY. As such, the relocated details are not required to be in the ITS to provide adequate protection of the public health and safety. Changes to the UFSAR will be controlled by the provisions of the 10 CFR 50.59.

"Specific"

None

RELOCATED SPECIFICATIONS

None

A.1 ITS 3.4.12

REACTOR COOLANT SYSTEM

REACTOR STEAM DOME

LIMITING CONDITION FOR OPERATION

LCO 3.4.12

3.4.6.2 The pressure in the reactor steam dome shall be less than or equal to 1020 psig. L.1

APPLICABILITY: OPERATIONAL CONDITION 1* and 2*. M.1

ACTION:

ACTION A — With the reactor steam dome pressure exceeding 1020 psig, reduce the pressure to less than 1020 psig within 15 minutes or be in at least HOT
ACTION B — SHUTDOWN within 12 hours.

SURVEILLANCE REQUIREMENTS

SR 3.4.12.1

4.4.6.2 The reactor steam dome pressure shall be verified to be less than or equal to 1020 psig at least once per 12 hours. L.1

~~*Not applicable during anticipated transients.~~ M.1

Page 1 of 2

ITS 3.4.12

A.1

REACTOR COOLANT SYSTEM

REACTOR STEAM DOME

LIMITING CONDITION FOR OPERATION

LCO 3.4.12 3.4.6.2 The pressure in the reactor steam dome shall be less than or equal to 1020 psig. L.1

APPLICABILITY: OPERATIONAL CONDITIONS 1~~*~~ and 2~~*~~. M.1

ACTION:

ACTION A With the reactor steam dome pressure exceeding 1020 psig, reduce the pressure to less than 1020 psig within 15 minutes or be in at least HOT SHUTDOWN within 12 hours.
ACTION B

SURVEILLANCE REQUIREMENTS

SR 3.4.12.1 4.4.6.2 The reactor steam dome pressure shall be verified to be less than or equal to 1020 psig at least once per 12 hours. L.1

M.1

~~*Not applicable during anticipated transients.~~

DISCUSSION OF CHANGES
ITS: 3.4.12 - REACTOR STEAM DOME PRESSURE

ADMINISTRATIVE

A.1 In the conversion of the LaSalle 1 and 2 current Technical Specifications (CTS) to the proposed plant specific Improved Technical Specifications (ITS), certain wording preferences or conventions are adopted that do not result in technical changes (either actual or interpretational). Editorial changes, reformatting, and revised numbering are adopted to make the ITS consistent with the BWR Standard Technical Specifications, NUREG-1434, Rev. 1 (i.e., the Improved Standard Technical Specifications (ISTS)).

TECHNICAL CHANGES - MORE RESTRICTIVE

M.1 The CTS 3.4.6.2 footnote * that states that the reactor steam dome pressure limit is not applicable during anticipated transients is deleted. The reactor steam dome pressure limit is provided to ensure the initial assumption of transient analyses is being met. The Required Actions of ITS 3.4.12 provide for prompt restoration of this initial assumption in the event a transient occurs causing reactor steam dome pressure to exceed the limit. This change represents an additional restriction on plant operation.

TECHNICAL CHANGES - LESS RESTRICTIVE

"Generic"

None

"Specific"

L.1 The reactor steam dome pressure limit of CTS 3.4.6.2 and CTS 4.4.6.2 (ITS 3.4.12 and SR 3.4.12.1) has been slightly increased to allow pressure to be equal to 1020 psig and still be within the limit. Currently, CTS 3.4.6.2 and CTS 4.4.6.2 require the reactor steam dome pressure to be less than 1020 psig. The safety analysis described in the UFSAR (overpressure protection analysis) assumes the initial reactor steam dome pressure is 1020 psig, not less than 1020 psig.

RELOCATED SPECIFICATIONS

None

Current Specification 3/4.4.8

R.1

3/4.4.8 STRUCTURAL INTEGRITY

LIMITING CONDITION FOR OPERATION

3.4.8 The structural integrity of ASME Code Class 1, 2 and 3 components shall be maintained in accordance with Specification 4.4.8.

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2, 3, 4 and 5.

ACTION:

a. With the structural integrity of any ASME Code Class 1 component(s) not conforming to the above requirements, restore the structural integrity of the affected component(s) to within its limit or isolate the affected component(s) prior to increasing the Reactor Coolant System temperature more than 50°F above the minimum temperature required by NDT considerations.

b. With the structural integrity of any ASME Code Class 2 component(s) not conforming to the above requirements, restore the structural integrity of the affected component(s) to within its limit or isolate the affected component(s) prior to increasing the Reactor Coolant System temperature above 200°F.

c. With the structural integrity of any ASME Code Class 3 component(s) not conforming to the above requirements, restore the structural integrity of the affected component(s) to within its limit or isolate the affected component(s) from service.

SURVEILLANCE REQUIREMENTS

4.4.8 No additional Surveillance Requirements other than those required by Specification 4.0.5.

Page 1 of 2

Current Specification 3/4.4.8

R.1

3/4.4.8 STRUCTURAL INTEGRITY

LIMITING CONDITION FOR OPERATION

3.4.8 The structural integrity of ASME Code Class 1, 2 and 3 components shall be maintained in accordance with Specification 4.4.8.

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2, 3, 4 and 5.

ACTION:

a. With the structural integrity of any ASME Code Class 1 component(s) not conforming to the above requirements, restore the structural integrity of the affected component(s) to within its limit or isolate the affected component(s) prior to increasing the Reactor Coolant System temperature more than 50°F above the minimum temperature required by NDT considerations.

b. With the structural integrity of any ASME Code Class 2 component(s) not conforming to the above requirements, restore the structural integrity of the affected component(s) to within its limit or isolate the affected component(s) prior to increasing the Reactor Coolant System temperature above 200°F.

c. With the structural integrity of any ASME Code Class 3 component(s) not conforming to the above requirements, restore the structural integrity of the affected component(s) to within its limit or isolate the affected component(s) from service.

SURVEILLANCE REQUIREMENTS

4.4.8 No additional Surveillance Requirements other than those required by Specification 4.0.5.

ADMINISTRATIVE

None

TECHNICAL CHANGES - MORE RESTRICTIVE

None

TECHNICAL CHANGES - LESS RESTRICTIVE

None

RELOCATED SPECIFICATIONS

R.1 The CTS 3/4.4.8 structural integrity inspections are provided to prevent long term component degradation and provide long term maintenance of acceptable structural conditions of the system. The associated inspections are not required to ensure immediate OPERABILITY of the system. Therefore, the requirements specified in CTS 3/4.4.8 did not satisfy the NRC Policy Statement Technical Specification screening criteria as documented in the Application of Selection Criteria to the LaSalle 1 and 2 Technical Specifications and have been relocated to the Technical Requirements Manual (TRM). The TRM will be incorporated by reference into the UFSAR at ITS implementation. Changes to the TRM will be controlled by the provisions of 10 CFR 50.59.

DISCUSSION OF CHANGES
ITS: SECTION 3.4 - REACTOR COOLANT SYSTEM BASES

The Bases of the current Technical Specifications for this section (B 3/4 4-1 through B 3/4 4-6) have been completely replaced by revised Bases reflecting the format and applicable content of the LaSalle 1 and 2 ITS Section 3.4, consistent with the BWR ISTS, NUREG-1434, Rev. 1. The revised Bases are as shown in the LaSalle 1 and 2 ITS Bases.

<CTS>

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.1 Recirculation Loops Operating

<LCO 3.4.1.1> LCO 3.4.1 Two recirculation loops within Region III of Figure 3.4.1-1 [1] with matched flows shall be in operation~~,~~.

<3.4.1 Act a>
<3.4.1 Act a.1.c)>
<3.4.1 Act a.1.d)>
<3.4.1 Act a.1.e)>
<LCO 3.4.1.3>
<LCO 3.4.1.5>
<LCO 3.4.1.5.b>

[2]

OR

One recirculation loop ~~may~~ shall [1] be in operation ~~provided~~ with the [1] following limits ~~are~~ applied when the associated LCO is applicable: [1]

a. LCO 3.2.1, "AVERAGE PLANAR LINEAR HEAT GENERATION RATE (APLHGR)," single loop operation limits [2] {specified in the COLR};

b. LCO 3.2.2, "MINIMUM CRITICAL POWER RATIO (MCPR)," single loop operation limits [2] {specified in the COLR}; ~~and~~ [1]

c. LCO 3.3.1.1, "Reactor Protection System (RPS) Instrumentation," Function 2.b (Average Power Range Monitors Flow Biased Simulated Thermal Power—~~High~~ Upscale [3]), Allowable Value of Table 3.3.1.1-1 is reset for single loop operation~~.~~; and [1]

d. LCO 3.3.2.1, "Control Rod Block Instrumentation," Function 1.a (Rod Block Monitor—Upscale), Allowable Value of Table 3.3.2.1-1, specified in the COLR, is reset for single loop operation. [1]

<Appl 3.4.1.1>
<Appl 3.4.1.3>
<Appl 3.4.1.5>

APPLICABILITY: MODES 1 and 2.

ACTIONS

<3.4.1.1 Act a.1>

CONDITION	REQUIRED ACTION	COMPLETION TIME
~~A.~~ G. [1] Requirements of the LCO not met for reasons other than Conditions A, C, D, and F [1].	~~A.1~~ G.1 [1] Satisfy the requirements of the LCO.	~~24~~ 12 hours [1]

~~Pending resolution of stability issue.~~ [1]

Insert 3.4.1-ACTIONS [1]

(continued)

<CTS>

INSERT 3.4.1 - ACTIONS

<LCO 3.4.1.5.c>
<4.4.1.5>

A. One or two recirculation loops operating within Region II of Figure 3.4.1-1.	A.1 -----------NOTE---------- Only applicable when 3 times baseline value is > 10% peak-to-peak value. ------------------------ Verify APRM and LPRM flux noise levels ≤ 3 times baseline.	45 minutes AND Once per 12 hours thereafter AND 45 minutes from discovery of Condition A concurrent with any THERMAL POWER increase of ≥ 5% RTP.
	AND	(continued)

<CTS>

INSERT 3.4.1 - ACTIONS

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. (continued) <LCO 3.4.1.5.c> <4.4.1.5>	A.2 -----------NOTE---------- Only applicable when 10% peak-to-peak value is $\geq$ 3 times baseline value. ------------------------ Verify APRM and LPRM flux noise levels $\leq$ 10% peak-to-peak.	45 minutes AND Once per 12 hours thereafter AND 45 minutes from discovery of Condition A concurrent with any THERMAL POWER increase of $\geq$ 5% RTP.
	AND A.3 Verify recirculation loop(s) are not operating in Region I of Figure 3.4.1-1.	Once per 12 hours
<3.4.1.5 Act b> B. Required Action A.1 or A.2 and associated Completion Time not met.	B.1 Satisfy the requirements of the LCO.	2 hours

(continued)

<CTS>

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
<3.4.1.5 Act a 3.4.1.5 Act a.1> C. One or two recirculation loops operating within Region I of Figure 3.4.1-1. OR Required Action A.3 and associated Completion Time not met.	C.1 Exit Region I of Figure 3.4.1-1.	2 hours
<3.4.1.1 Act b & Act b.2> <3.4.1.5 Act a.2 Act a.2.a & Act a.2.c> D. No recirculation loops in operation.	D.1 Verify APRM and LPRM flux noise levels $\leq$ 10% peak-to-peak. AND D.2 Reduce THERMAL POWER to < 36% RTP. AND D.3 Be in MODE 3.	Immediately 2 hours 12 hours
<3.4.1.5 Act a.2.b> <DOC M.3> E. Required Action B.1 or D.1 and associated Completion Time not met.	E.1 Place the mode switch in the shutdown position.	Immediately
<3.4.1.3 Act a & Act b> F. Recirculation loop flow mismatch not within limits.	F.1 Declare the recirculation loop with lower flow to be "not in operation."	2 hours

<CTS>

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
<3.4.1.1 Acta.2> 1 ~~B~~ H. Required Action and associated Completion Time of Condition ~~A~~ G not met. 1 1 ~~OR~~ ~~No recirculation loops in operation.~~	~~B~~ H.1 1 Be in MODE 3.	12 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
<LCO 3.4.1.3.a> <LCO 3.4.1.3.b> <4.4.1.3> SR 3.4.1.1	-------------------NOTE------------------- Not required to be performed until 24 hours after both recirculation loops are in operation. --- Verify recirculation loop jet pump flow mismatch with both recirculation loops in operation is: 2 a. ≤ 10% of rated core flow when operating at < 70% of rated core flow; and b. ≤ 5% of rated core flow when operating at ≥ 70% of rated core flow.	24 hours
<Doc M.4> SR 3.4.1.2	Verify operation is in Region III of Figure 3.4.1-1.	24 hours

1

1 Insert 3.4.1-Figure

⟨CTS⟩

⟨Figure 3.4.1.5-1⟩

INSERT 3.4.1-FIGURE

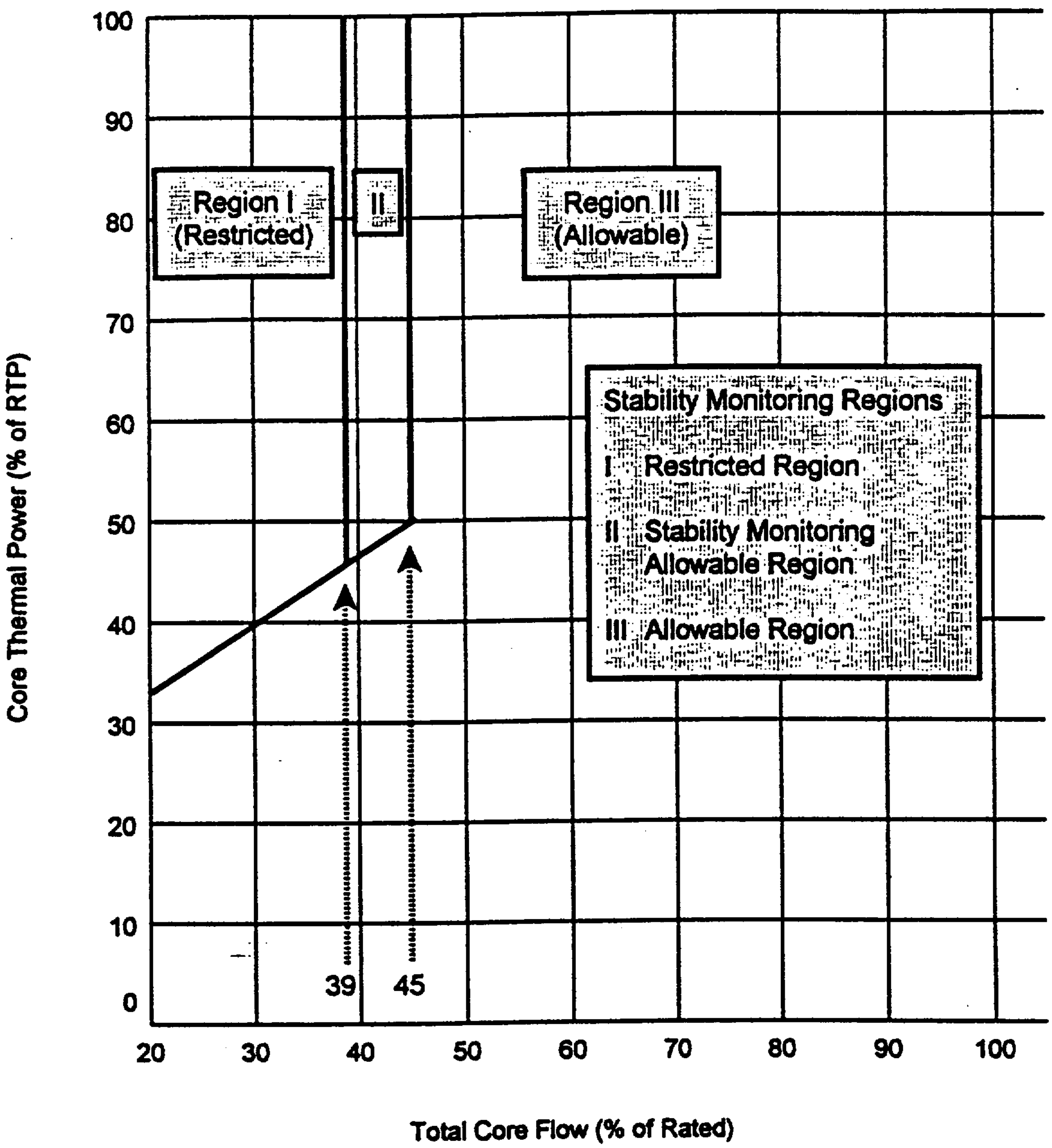


Figure 3.4.1-1 (Page 1 of 1)
Power versus Flow

JUSTIFICATION FOR DEVIATIONS FROM NUREG-1433, REVISION 1
ITS: 3.4.1 - RECIRCULATION LOOPS OPERATING

1. The "Recirculation Loops Operating" Specification has been revised to reflect Current Technical Specifications requirements related to core thermal hydraulic stability, except where justified in the Discussion of Changes. When ComEd completes resolution of the long-term stability issue, the ITS will be revised appropriately. The following requirements are renumbered to reflect this change.

2. The brackets have been removed and the proper plant specific information/value has been provided.

3. Changes have been made to reflect plant specific nomenclature.

<CTS>

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.2 Flow Control Valves (FCVs)

<DOC A.2> LCO 3.4.2 A recirculation loop FCV shall be OPERABLE in each operating recirculation loop.

<DOC A.2> APPLICABILITY: MODES 1 and 2.

ACTIONS

-----NOTE-----

<DOC A.2> Separate Condition entry is allowed for each FCV.

CONDITION	REQUIRED ACTION	COMPLETION TIME
<DOC A.2> A. One or two required FCVs inoperable.	A.1 Lock up the FCV.	4 hours
<DOC A.2> B. Required Action and associated Completion Time not met.	B.1 Be in MODE 3.	12 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
<4.4.1.1.a> SR 3.4.2.1 Verify each FCV fails "as is" on loss of hydraulic pressure at the hydraulic unit.	~~[18]~~ 24 months 1

(continued)

<CTS>

SURVEILLANCE REQUIREMENTS (continued)

	SURVEILLANCE	FREQUENCY
<4.4.1.1.b> SR 3.4.2.2	Verify average rate of each FCV movement is: 1 a. ≤ ~~[11]~~% of stroke per second for opening; and b. ≤ ~~[11]~~% of stroke per second for closing.	~~[18]~~ 24 months 1

JUSTIFICATION FOR DEVIATIONS FROM NUREG-1434, REVISION 1
ITS: 3.4.2 - FLOW CONTROL VALVES (FCVs)

1. The brackets have been removed and the proper plant specific information/value has been provided.

<CTS>

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.3 Jet Pumps

<LCO 3.4.1.2> LCO 3.4.3 All jet pumps shall be OPERABLE.

<Appl 3.4.1.2> APPLICABILITY: MODES 1 and 2.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
<3.4.1.2 Act> A. One or more jet pumps inoperable.	A.1 Be in MODE 3.	12 hours

<CTS>

SURVEILLANCE REQUIREMENTS

	SURVEILLANCE	FREQUENCY
<4.4.1.2.1 4.4.1.2.2> SR 3.4.3.1	-------------------NOTES------------------- 1. Not required to be performed until 4 hours after associated recirculation loop is in operation. 2. Not required to be performed until 24 hours after > 25% RTP. --- Verify at least two of the following criteria (a, b, and c) are satisfied for each operating recirculation loop: a. Recirculation loop drive flow versus flow control valve position differs by ≤ 10% from established patterns. b. ~~Recirculation loop drive~~ Indicated total core flow versus Calculated total core flow differs by ≤ 10% from established patterns. c. Each jet pump diffuser to lower plenum differential pressure differs by ≤ 20% from established patterns~~; or each jet pump flow differs by ≤ 10% from established patterns~~.	24 hours 1 3

~~Reviewer's Note: An acceptable option to these criteria for jet pump OPERABILITY can be found in the BWR/4 ITS, NUREG-1433.~~ 2

JUSTIFICATION FOR DEVIATIONS FROM NUREG-1434, REVISION 1
ITS: 3.4.3 - JET PUMPS

1. The proper LaSalle plant specific nomenclature/value has been provided.

2. This Reviewer's Note has been deleted. This Note provides the location of an alternative set of criteria that is not used at LaSalle. This is not meant to be retained in the final version of the plant specific submittal.

3. LaSalle does not have installed jet pump flow indicators for all jet pumps. Therefore, this method in NUREG-1434 SR 3.4.3.1.c will not be added.

<CTS>

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.4 Safety/Relief Valves (S/RVs)

17 S/RVs for Unit 1, and 12 [1]

for Unit 2,

<LCO 3.4.2> LCO 3.4.4 The safety function of [seven] S/RVs shall be OPERABLE,

~~**AND**~~

~~The relief function of [seven] additional S/RVs shall be OPERABLE.~~ [2]

<Appl 3.4.2> APPLICABILITY: MODES 1, 2, and 3.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
~~A. One [required] S/RV inoperable.~~	~~A.1 Restore [required] S/RV to OPERABLE status.~~	~~14 days~~ [3]
[3] A. ~~B. Required Action and associated Completion Time of Condition A not met.~~ ~~OR~~ <3.4.2 Acta> [1] One ~~[Two]~~ or more ~~[required]~~ S/RVs inoperable.	A ~~B~~.1 Be in MODE 3. **AND** [3] A ~~B~~.2 Be in MODE 4.	12 hours 36 hours

6 ---------- NOTE ---------- 5
Less than or equal to two required SRVs may be changed to a lower setpoint group.

<CTS>

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
<LCO 3.4.2> <LCO 3.4.2 fnote * LCO 3.4.2 fnote **> <DOC A.4> SR 3.4.4.1	Verify the safety function lift setpoints of the ~~[required]~~ S/RVs are as follows: 1 1 ~~Number of S/RVs: [8], [6], [6]~~ ~~Setpoint (psig): [1165 ± 34.9], [1180 ± 35.4], [1190 ± 35.7]~~ Following testing, lift settings shall be within ± 1%.	In accordance with the Inservice Testing Program ~~or [18] months~~ 1
SR 3.4.4.2	~~NOTE: Valve actuation may be excluded.~~ ~~Verify each [required] relief function S/RV actuates on an actual or simulated automatic initiation signal.~~	~~[18] months~~ 2
SR 3.4.4.3	~~NOTE: Not required to be performed until 12 hours after reactor steam pressure and flow are adequate to perform the test.~~ ~~Verify each [required] S/RV opens when manually actuated.~~	~~[18] months on a STAGGERED TEST BASIS for each valve solenoid~~ 4

Number of Unit 1 S/RVs	Setpoint (psig)
4	1205 ± 36.1
4	1195 ± 35.8
4	1185 ± 35.5
4	1175 ± 35.2
2	1150 ± 34.5

Number of Unit 2 S/RVs	Setpoint (psig)
2	1205 ± 36.1
3	1195 ± 35.8
2	1185 ± 35.5
4	1175 ± 35.2
2	1150 ± 34.5

JUSTIFICATION FOR DEVIATIONS FROM NUREG-1434, REVISION 1
ITS: 3.4.4 - SAFETY/RELIEF VALVES (S/RVs)

1. The brackets have been removed and the proper plant specific information/value has been provided.

2. The Current LaSalle 1 and 2 Licensing Basis does not include Technical Specification requirements for the relief mode of the S/RVs since the overpressure protection analysis does not assume the relief mode functions to mitigate an overpressurization event. Therefore, the relief mode requirements have been deleted.

3. This bracketed requirement (ISTS 3.4.4 ACTION A) has been deleted because it is not applicable to LaSalle. The Current LaSalle 1 and 2 Licensing Basis does not require an additional S/RV to meet the single failure criterion, thus the ISTS 3.4.4 ACTION A is not needed. The following Action has been renumbered to reflect this deletion. In addition, since ACTION A is not included, the bracketed first condition of ISTS ACTION B (ITS ACTION A) is not needed and has been deleted.

4. The Current LaSalle 1 and 2 Licensing Basis does not include Technical Specification requirements to manually actuate the S/RVs during MODES 1, 2, and 3. Post-maintenance testing provides adequate controls to ensure proper operation of the S/RVs during plant transients. The valves are bench tested in accordance with the applicable code as delineated in the Inservice Testing Program, which includes set pressure determination, seat leakage, and visual examination. Additional testing includes valve lift, opening response time, solenoid/air control valve electrical characteristics and pressure integrity, and air actuator pressure integrity and stroke capability. Stroke testing is also performed to verify relief mode functionality as part of post-maintenance testing. When the S/RVs are re-installed, post-maintenance tests are performed to verify the S/RVs have been properly reconnected electrically, pneumatically and mechanically. This includes cycling the valve actuator prior to plant startup. In addition, to avoid blockage of the valve discharge line administrative controls are in place to cover the valve discharge upon removal of the S/RV. This procedural control is reflected in LMP-GM-06, "Removal/Installation of Main Steam Safety Relief Valves."

5. The proposed Note to SR 3.4.4.1 allows two required S/RVs to be changed to a lower setpoint group. This Note is added to be consistent with the Current LaSalle 1 and 2 Licensing Basis reflected in the # Note of CTS 3.4.2. The Note is also consistent with proposed ISTS Generic Change TSTF-298.

6. Editorial change has been made to achieve consistency with the Writer's Guide.

<CTS>

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.5 RCS Operational LEAKAGE

<LCO 3.4.3.2> LCO 3.4.5 RCS operational LEAKAGE shall be limited to:

a. No pressure boundary LEAKAGE;

b. ≤ 5 gpm unidentified LEAKAGE; [and] 1

c. ≤ [30] 25 1 gpm total LEAKAGE averaged over the previous 24 hour period; and 1

d. ≤ 2 gpm increase in unidentified LEAKAGE within the previous [4] 24 1 hour period in MODE 1]. 1

<Appl 3.4.3.2> APPLICABILITY: MODES 1, 2, and 3.

ACTIONS

	CONDITION	REQUIRED ACTION	COMPLETION TIME
<3.4.3.2 Act b>	A. Unidentified LEAKAGE not within limit. **OR** Total LEAKAGE not within limit.	A.1 Reduce LEAKAGE to within limits.	4 hours
<3.4.3.1 Act e>	B. Unidentified LEAKAGE increase not within limit.	B.1 Reduce~~d~~ unidentified 3 2 LEAKAGE increase 3 to within limit. **OR**	4 hours (continued)

<CTS>

ACTIONS

	CONDITION	REQUIRED ACTION	COMPLETION TIME	
<3.4.3.2 Act e>	B. (continued) is not intergranular stress corrosion cracking susceptible material.	B.2 ~~Verify~~ Identify source of unidentified LEAKAGE increase ~~is not service sensitive type 304 or type 316 austenitic stainless steel~~.	4 hours	4
<3.4.3.2 Act a, 3.4.3.2 Act b, 3.4.3.2 Act e>	C. Required Action and associated Completion Time of Condition A or B not met. OR Pressure boundary LEAKAGE exists.	C.1 Be in MODE 3. AND C.2 Be in MODE 4.	12 hours 36 hours	

SURVEILLANCE REQUIREMENTS

	SURVEILLANCE		FREQUENCY
<4.4.3.2.1>	SR 3.4.5.1	Verify RCS unidentified and total LEAKAGE and unidentified LEAKAGE increase are within limits.	~~8~~ 12 hours 5

JUSTIFICATION FOR DEVIATIONS FROM NUREG-1434, REVISION 1
ITS: 3.4.5 - RCS OPERATIONAL LEAKAGE

1. The brackets have been removed and the proper plant specific information/value has been provided.

2. Typographical/grammatical error corrected.

3. Editorial change has been made to achieve consistency with the Writer's Guide.

4. The ISTS 3.4.5 Required Action B.2 requirement to verify the source of the unidentified LEAKAGE increase is not service sensitive type 304 or type 316 austenitic stainless steel has been changed in ITS 3.4.5 Required Action B.2 to only require identification of the source of the unidentified LEAKAGE increase is not intergranular stress corrosion cracking susceptible material. This change is consistent with the LaSalle CTS, as modified by changes for consistency with the Dresden and Quad Cities CTS.

5. The Surveillance Frequency has been extended from 8 hours to 12 hours consistent with Generic Letter 88-01, Supplement 1. The supplement allowed the Frequency to be once per shift, not to exceed 12 hours.

<CTS>

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.6 RCS Pressure Isolation Valve (PIV) Leakage

<LCO 3.4.3.2> LCO 3.4.6 The leakage from each RCS PIV shall be within limit.

<Appl 3.4.3.2> APPLICABILITY: MODES 1 and 2,
MODE 3, except valves in the residual heat removal (RHR) shutdown cooling flow path when in, or during the transition to or from, the shutdown cooling mode of operation. $\boxed{1}$

ACTIONS

-----NOTES-----

<DOC A.2>

1. Separate Condition entry is allowed for each flow path.

2. Enter applicable Conditions and Required Actions for systems made inoperable by PIVs.

CONDITION	REQUIRED ACTION	COMPLETION TIME
<3.4.3.2 Actc> A. One or more flow paths with leakage from one or more RCS PIVs not within limit.	-----NOTE----- Each valve used to satisfy Required Action A.1 and Required Action A.2 shall have been verified to meet SR 3.4.6.1 and be in the reactor coolant pressure boundary {for the high pressure portion of the system}. $\boxed{2}$ ----------	

(continued)

<CTS>

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
<3.4.3.2 Act c> A. (continued)	A.1	Isolate the high pressure portion of the affected system from the low pressure portion by use of one closed manual, deactivated automatic, or check valve.	4 hours
	<u>AND</u>		
	A.2	Isolate the high pressure portion of the affected system from the low pressure portion by use of a second closed manual, deactivated automatic, or check valve.	72 hours
<3.4.3.2 Act c> B. Required Action and associated Completion Time not met.	B.1	Be in MODE 3.	12 hours
	<u>AND</u>		
	B.2	Be in MODE 4.	36 hours

<CTS>

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
1 SR 3.4.6.1 <4.4.3.2.2>	---NOTE--- Only ~~Not~~ required to be performed in ~~MODE 3~~ MODES 1 and 2 [1] Verify equivalent leakage of each RCS PIV is ≤ 0.5 gpm per nominal inch of valve size up to a maximum of 5 gpm, at an RCS pressure ≥ ~~[1040]~~ 950 psig and ≤ ~~[1060]~~ 1050 psig.	~~[~~In accordance with the Inservice Testing Program ~~or [18] months]~~ [1]

JUSTIFICATION FOR DEVIATIONS FROM NUREG-1434, REVISION 1
ITS: 3.4.6 - RCS PRESSURE ISOLATION VALVE (PIV) LEAKAGE

1. Editorial changes have been made to achieve consistency with the Writer's Guide.

2. The brackets have been removed and the proper plant specific information/value has been provided.

<CTS>

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.7 RCS Leakage Detection Instrumentation

<LCO 3.4.3.1> LCO 3.4.7 The following RCS leakage detection instrumentation shall be OPERABLE:

a. Drywell floor drain sump flow [1] monitoring system; ~~{and}~~ [2]

b. One channel of either drywell atmospheric particulate or atmospheric gaseous monitoring system; ~~{~~and [2]

c. Drywell air cooler condensate flow rate monitoring system~~}~~. [2]

<Appl 3.4.3.1> APPLICABILITY: MODES 1, 2, and 3.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
<3.4.3.1 Act> <DOC L.2> A. Drywell floor drain sump flow [1] monitoring system inoperable.	~~NOTE: LCO 3.0.4 is not applicable.~~ [3] TSTF-60 not adopted A.1 Restore drywell floor drain sump flow [1] monitoring system to OPERABLE status.	30 days

(continued)

<CTS>

ACTIONS (continued)

	CONDITION	REQUIRED ACTION	COMPLETION TIME	
<3.4.3.1 Act>	B. Required drywell atmospheric monitoring system inoperable.	~~-----NOTE----- LCO 3.0.4 is not applicable.~~ 3 5		TSTF-60 not adopted
		B.1 Analyze grab samples of drywell atmosphere.	Once per 12 hours	
		~~AND~~		
		~~B.2 Restore required drywell atmospheric monitoring system to OPERABLE status.~~	~~30 days~~	5
<3.4.3.1 Act> <DOC M.1>	[C. Drywell air cooler condensate flow rate monitoring system inoperable.	-----NOTE----- Not applicable when the required drywell atmospheric monitoring system is inoperable.		
		C.1 Perform SR 3.4.7.1.	Once per 8 hours]	2
<3.4.3.1 Act> <DOC L.2>	[D. Required drywell atmospheric monitoring system inoperable. AND Drywell air cooler condensate flow rate monitoring system inoperable.	-----NOTE----- LCO 3.0.4 is not applicable. 3		TSTF-60 not adopted
		D.1 Restore required drywell atmospheric monitoring system to OPERABLE status.	30 days	
		OR	]	2

(continued)

<CTS>

ACTIONS

	CONDITION	REQUIRED ACTION	COMPLETION TIME
<3.4.3.1 Act> <Doc L.2>	D. (continued)	D.2 Restore drywell air cooler condensate flow rate monitoring system to OPERABLE status.	30 days 2
<3.4.3.1 Act>	E. Required Action and associated Completion Time of Condition A, B, C, or D not met. 2	E.1 Be in MODE 3. AND E.2 Be in MODE 4.	12 hours 36 hours
<Doc A.2>	F. All required leakage detection systems inoperable.	F.1 Enter LCO 3.0.3.	Immediately

<insert SR Note> 4

SURVEILLANCE REQUIREMENTS

	SURVEILLANCE		FREQUENCY
<4.4.3.1.a>	SR 3.4.7.1	Perform CHANNEL CHECK of required drywell atmospheric monitoring system.	12 hours
<4.4.3.1.a 4.4.3.1.b 4.4.3.1.c>	SR 3.4.7.2	Perform CHANNEL FUNCTIONAL TEST of required leakage detection instrumentation.	31 days
<4.4.3.1.a 4.4.3.1.b 4.4.3.1.c>	SR 3.4.7.3	Perform CHANNEL CALIBRATION of required leakage detection instrumentation.	~~[18]~~ 24 months 2

<CTS>

INSERT SR NOTE

-----NOTE-----

When a channel is placed in an inoperable status solely for performance of required Surveillances, entry into associated Conditions and Required Actions may be delayed for up to 6 hours provided the other required leakage detection instrumentation is OPERABLE.

<DOC L.3>

JUSTIFICATION FOR DEVIATIONS FROM NUREG-1434, REVISION 1
ITS: 3.4.7 - RCS LEAKAGE DETECTION INSTRUMENTATION

1. The proper LaSalle plant specific nomenclature/value has been provided.

2. The brackets have been removed and the proper plant specific information/value has been provided.

3. TSTF-60 moved the Notes in ISTS ACTIONS A, B, and D to the beginning of the ACTIONS Table. This implies the Note is applicable to all the ACTIONS. However, the Note "LCO 3.0.4 is not applicable" is not applicable to ISTS ACTIONS E and F, which require a plant shutdown. Therefore, to avoid confusion, TSTF-60 has not been incorporated.

4. A Note has been added to allow a channel to be inoperable for up to 6 hours solely for performance of required Surveillances, provided the other Leakage Detection System instrumentation is OPERABLE. This Note is similar to other Notes in the ITS, which allow channels that provide automatic actions to be inoperable for up to 6 hours. This instrumentation only provides indication, and the 6 hour allowance is not allowed unless the other channel is OPERABLE. This change has previously been approved at Georgia Power Company's Plant Hatch Units 1 and 2, in amendments 185 and 125, respectively, and in the ITS for Washington Public Power Supply System, Amendment 149.

5. The bracketed requirement (ISTS 3.4.7 Required Action B.2) to restore required drywell atmospheric monitoring systems to OPERABLE status within 30 days, is not applicable to LaSalle due to the existence of other methods of RCS leakage detection, and has been deleted. Additionally, the Note associated with ISTS 3.4.7 Required Action B stating that LCO 3.0.4 is not applicable has also been deleted because it is no longer required due to the deletion of Required Action B.2.

<CTS>

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.8 RCS Specific Activity

<LCO 3.4.5> LCO 3.4.8 The specific activity of the reactor coolant shall be limited to DOSE EQUIVALENT I-131 specific activity ≤ ~~[~~0.2~~]~~ μCi/gm. 2 1 – ≤

<Appl 3.4.5> APPLICABILITY: MODE 1,
MODES 2 and 3 with any main steam line not isolated.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
<3.4.5 Act a.1, 3.4.5 Act b> A. Reactor coolant specific activity 2 > ~~[~~0.2~~]~~ μCi/gm and ≤ 4.0 μCi/gm DOSE EQUIVALENT I-131.	-------------NOTE----------- LCO 3.0.4 is not applicable. ---------------------------	
	A.1 Determine DOSE EQUIVALENT I-131.	Once per 4 hours
	AND	
	A.2 Restore DOSE EQUIVALENT I-131 to within limits.	48 hours
<3.4.5 Act a.1, 3.4.5 Act b> B. Required Action and associated Completion Time of Condition A not met.	B.1 Determine DOSE EQUIVALENT I-131.	Once per 4 hours
OR	**AND**	
1 Reactor coolant ~~S~~specific activity 2 > ~~[~~4.0~~]~~ μCi/gm DOSE EQUIVALENT I-131.	B.2.1 Isolate all main steam lines.	12 hours
	OR	
		(continued)

<CTS>

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
<3.4.5 Act a.1> <3.4.5 Act b> B. (continued)	B.2.2.1 Be in MODE 3. **AND** B.2.2.2 Be in MODE 4.	12 hours 36 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
<4.4.5> <Table 4.4.5-1 Item 2> SR 3.4.8.1	------------------NOTE------------------ Only required to be performed in MODE 1. -- Verify reactor coolant DOSE EQUIVALENT I-131 specific activity is $\boxed{2}$ ≤ [0.2] μCi/gm.	7 days

JUSTIFICATION FOR DEVIATIONS FROM NUREG-1434, REVISION 1
ITS: 3.4.8 - RCS SPECIFIC ACTIVITY

1. Typographical/grammatical error corrected.

2. The brackets have been removed and the proper plant specific information/value has been provided.

<CTS>

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.9 Residual Heat Removal (RHR) Shutdown Cooling System—Hot Shutdown

<LCO 3.4.9.1> LCO 3.4.9 Two RHR shutdown cooling subsystems shall be OPERABLE, and, with no recirculation pump in operation, at least one RHR shutdown cooling subsystem shall be in operation.

---------------------------NOTES---------------------------

<3.4.9.1 fnote *> 1. Both RHR shutdown cooling subsystems and recirculation pumps may be ~~removed from~~ not in operation for up to 2 hours per 8 hour period. TSTF-153 2

<3.4.9.1 fnote #> 2. One RHR shutdown cooling subsystem may be inoperable for up to 2 hours for performance of Surveillances.

APPLICABILITY: MODE 3 with reactor ~~steam dome~~ vessel 3 pressure ~~≥~~ less than [the RHR cut in permissive pressure]. 1

ACTIONS

---------------------------NOTES---------------------------

<Doc L.2> 1. LCO 3.0.4 is not applicable.

<Doc A.3> 2. Separate Condition entry is allowed for each RHR shutdown cooling subsystem.

CONDITION	REQUIRED ACTION	COMPLETION TIME
<3.4.9.1 Act a> A. One or two RHR shutdown cooling subsystems inoperable.	A.1 Initiate action to restore RHR shutdown cooling subsystem to OPERABLE status. **AND**	Immediately (continued)

<CTS> ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
<3.4.9.1 Act a> A. (continued)	A.2 Verify an alternate method of decay heat removal is available for each inoperable RHR shutdown cooling subsystem. AND A.3 Be in MODE 4.	1 hour 24 hours
<3.4.9.1 Act b> <4.4.9.1> B. No RHR shutdown cooling subsystem in operation. AND No recirculation pump in operation.	B.1 Initiate action to restore one RHR shutdown cooling subsystem or one recirculation pump to operation. AND B.2 Verify reactor coolant circulation by an alternate method. AND B.3 Monitor reactor coolant temperature and pressure.	Immediately 1 hour from discovery of no reactor coolant circulation AND Once per 12 hours thereafter Once per hour

<CTS>

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
<4.4.9.1> SR 3.4.9.1 vessel 3	------------------NOTE------------------ Not required to be met until 2 hours after reactor ~~steam dome~~ vessel pressure is less than the RHR cut in permissive pressure. --	1
	Verify one RHR shutdown cooling subsystem or recirculation pump is operating.	12 hours

1. The brackets have been removed and the proper plant specific information/value has been provided.

2. TSTF-153 revised the RHR Shutdown Cooling System-Hot Shutdown LCO (ISTS LCO 3.4.9) Note 1, which provides an exception to the requirement for the required pump to be in operation, to provide a clarification of the intent of the Note consistent with the requirement being excepted. The justification for TSTF-153 described that the change was necessary to eliminate ambiguity that could lead to errors or improper enforcement. However, the change can now lead to a misinterpretation of the allowance of the Note. Specifically, the Note can now be interpreted as requiring the required subsystems or pumps to not be in operation for up to 2 hours per 8 hour period, i.e., they must be taken out of operation. The intent of the Note (as described in the associated Bases) is to allow (but not require) the required subsystems or pumps to not be in operation for up to 2 hours per 8 hour period. Therefore, the Note is revised to allow the subsystems or pumps to be "not in operation" for up to 2 hours per 8 hour period.

3. The proper LaSalle plant specific nomenclature has been provided.

<CTS>

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.10 Residual Heat Removal (RHR) Shutdown Cooling System—Cold Shutdown

<LCO 3.4.9.2> LCO 3.4.10 Two RHR shutdown cooling subsystems shall be OPERABLE, and, with no recirculation pump in operation, at least one RHR shutdown cooling subsystem shall be in operation.

-----NOTES-----

<3.4.9.2 fnote **> ~~1.~~ 2 [2] Both RHR shutdown cooling subsystems and recirculation pumps may be ~~removed from~~ not in operation for up to 2 hours per 8 hour period. TSTF 153 [1]

<3.4.9.2 fnote #> ~~2.~~ 3 One RHR shutdown cooling subsystem may be inoperable for up to 2 hours for the performance of Surveillances.

<3.4.9.2 fnote ##> 1. Not required to be met during hydrostatic testing. [2]

<Appl 3.4.9.2> APPLICABILITY: MODE 4.

ACTIONS

-----NOTE-----

<Doc A.3> Separate Condition entry is allowed for each RHR shutdown cooling subsystem.

CONDITION	REQUIRED ACTION	COMPLETION TIME
<3.4.9.2 Act a> A. One or two RHR shutdown cooling subsystems inoperable.	A.1 Verify an alternate method of decay heat removal is available for each inoperable RHR shutdown cooling subsystem.	1 hour AND Once per 24 hours thereafter

(continued)

<CTS>

ACTIONS (continued)

<3.4.9.2 Act b>
<4.4.9.2>

CONDITION	REQUIRED ACTION	COMPLETION TIME
B. No RHR shutdown cooling subsystem in operation. **AND** No recirculation pump in operation.	B.1 Verify reactor coolant circulating by an alternate method.	1 hour from discovery of no reactor coolant circulation **AND** Once per 12 hours thereafter
	AND	
	B.2 Monitor reactor coolant temperature and pressure.	Once per hour

SURVEILLANCE REQUIREMENTS

<4.4.9.2>

SURVEILLANCE		FREQUENCY
SR 3.4.10.1	Verify one RHR shutdown cooling subsystem or recirculation pump is operating.	12 hours

1. TSTF-153 revised the RHR Shutdown Cooling System-Cold Shutdown LCO (ISTS LCO 3.4.10) Note 1, which provides an exception to the requirement for the required pump to be in operation, to provide a clarification of the intent of the Note consistent with the requirement being excepted. The justification for TSTF-153 described that the change was necessary to eliminate ambiguity that could lead to errors or improper enforcement. However, the change can now lead to a misinterpretation of the allowance of the Note. Specifically, the Note can now be interpreted as requiring the required subsystems or pumps to not be in operation for up to 2 hours per 8 hour period, i.e., they must be taken out of operation. The intent of the Note (as described in the associated Bases) is to allow (but not require) the required subsystems or pumps to not be in operation for up to 2 hours per 8 hour period. Therefore, the Note is revised to allow the subsystems or pumps to be "not in operation" for up to 2 hours per 8 hour period.

2. Note 1 has been added to ISTS 3.4.10 (ITS 3.4.10) in order allow the performance of the hydrostatic test with both RHR shutdown cooling subsystems inoperable. This allowance in ITS 3.4.10 is necessary since ISTS 3.10.1, "Inservice Leak and Hydrostatic Testing Operation," has not been included in the LaSalle ITS due to the Technical Specifications Change Request submitted to the NRC for approval per ComEd letter dated February 29, 2000. The RHR Shutdown Cooling System is inoperable during hydrostatic testing since the system is not capable of circulating reactor coolant. The RHR Shutdown Cooling System is automatically isolated above the RHR cut-in permissive pressure. This isolation is necessary since the RHR Shutdown Cooling System is not designed to operate at the Reactor Coolant System pressure achieved during hydrostatic testing. This proposed Note is consistent with the ISTS 3.10.1 allowance to suspend the requirements of the RHR Shutdown Cooling System-Cold Shutdown LCO during hydrostatic testing. The subsequent notes have been renumbered to reflect this change.

<CTS>

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.11 RCS Pressure and Temperature (P/T) Limits

<LCO 3.4.6.1>
<LCO 3.4.11>

LCO 3.4.11 RCS pressure, RCS temperature, RCS heatup and cooldown rates, and the recirculation pump starting temperature requirements shall be maintained within ~~the~~ limits ~~specified in the PTLR~~. $\boxed{1}$

<Appl 3.4.6.1> APPLICABILITY: At all times.

ACTIONS

	CONDITION	REQUIRED ACTION	COMPLETION TIME
<3.4.6.1 Act a> <DOC M.2>	A. ---------NOTE--------- Required Action A.2 shall be completed if this Condition is entered. --------------------- Requirements of the LCO not met in ~~MODES~~ 1, 2, ~~and~~ or 3. $\boxed{2}$	A.1 Restore parameter(s) to within limits. AND A.2 Determine RCS is acceptable for continued operation.	30 minutes 72 hours
<3.4.6.1 Act> <DOC M.2>	B. Required Action and associated Completion Time of Condition A not met.	B.1 Be in MODE 3. AND B.2 Be in MODE 4.	12 hours 36 hours

(continued)

<CTS>

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
<3.4.6.1 Act> <DOC M.2> C. ----------NOTE---------- Required Action C.2 shall be completed if this Condition is entered. ------------------------ Requirements of the LCO not met in other than MODES 1, 2, and 3.	C.1 Initiate action to restore parameter(s) to within limits. AND C.2 Determine RCS is acceptable for operation.	Immediately Prior to entering MODE 2 or 3

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
<LCO 3.4.6.1> <LCO 3.4.6.1.a> <LCO 3.4.6.1.b> <LCO 3.4.6.1.c> <4.4.6.1> SR 3.4.11.1 --------------------NOTE-------------------- Only required to be performed during RCS heatup and cooldown operations, and RCS inservice leak and hydrostatic testing. --- ~~Verify~~ a. RCS pressure and RCS temperature ~~and RCS heatup and cooldown rates~~ are within the applicable limits specified in ~~the PTLR.~~ Figures 3.4.11-1, 3.4.11-2, 3.4.3.11-4, and 3.4.11-5; b. RCS heatup and cooldown rates are ≤ 100°F in any 1 hour period; and c. RCS temperature change during system leakage and hydrostatic testing is ≤ 20°F in any one hour period when the RCS pressure and RCS temperature are not within the limits of Figure 3.4.11-2 and 3.4.11.5, as applicable.	30 minutes
<LCO 3.4.6.1> <4.4.6.1.2> SR 3.4.11.2 Verify RCS pressure and RCS temperature are within the applicable criticality limits specified in ~~the PTLR~~ Figures 3.4.11-3, and 3.4.11-6.	Once within 15 minutes prior to control rod withdrawal for the purpose of achieving criticality

1

(continued)

⟨CTS⟩

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
⟨LCO 3.4.1.4⟩ ⟨Appl 3.4.1.4⟩ ⟨4.4.1.4⟩ SR 3.4.11.3	-------------------NOTE------------------- Only required to be met in MODES 1, 2, 3, and 4 ~~[with reactor steam dome pressure ≥ 25 psig]~~ during recirculation pump startup. TSTF 35 3 -- Verify the difference between the bottom head coolant temperature and the reactor pressure vessel (RPV) coolant temperature is ~~within the limits specified in the PTLR~~ ≤ 145°F 1	Once within 15 minutes prior to each startup of a recirculation pump
⟨LCO 3.4.1.4.a⟩ ⟨LCO 3.4.1.4.b⟩ ⟨Appl 3.4.1.4⟩ ⟨4.4.1.4⟩ SR 3.4.11.4	-------------------NOTE------------------- Only required to be met in MODES 1, 2, 3, and 4 during recirculation pump startup. -- Verify the difference between the reactor coolant temperature in the recirculation loop to be started and the RPV coolant temperature is ~~within the limits specified in the PTLR~~ ≤ 50°F 1	Once within 15 minutes prior to each startup of a recirculation pump
⟨LCO 3.4.6.1.d⟩ ⟨4.4.6.1.4 4.4.6.1.4.b⟩ SR 3.4.11.5	-------------------NOTE------------------- Only required to be performed when tensioning the reactor vessel head bolting studs. -- Verify reactor vessel flange and head flange temperatures are ~~within the limits specified in the PTLR~~ ≥ 72°F for Unit 1 and ≥ 86°F for Unit 2 1	30 minutes

TSTF-353 not adopted 4

(continued)

<CTS>

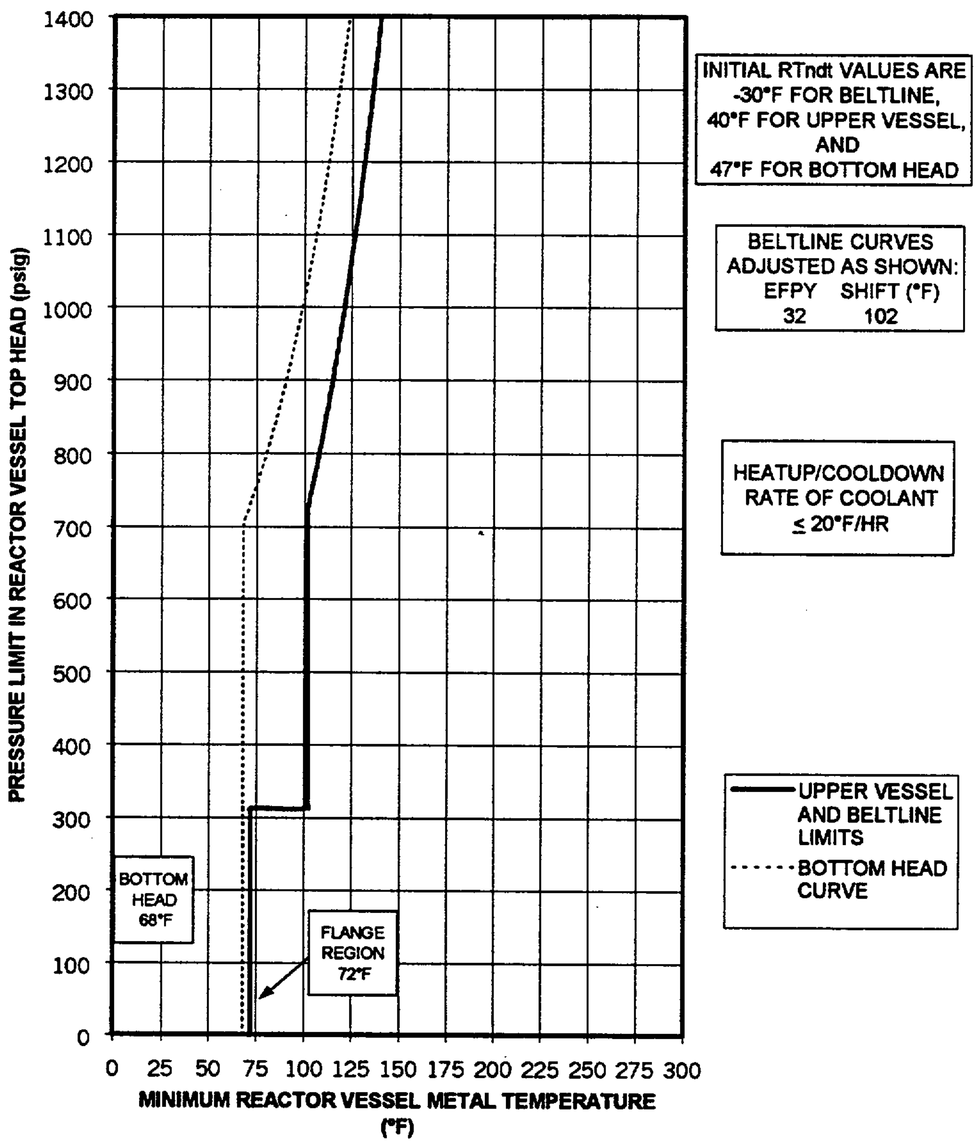
SURVEILLANCE REQUIREMENTS (continued)

	SURVEILLANCE	FREQUENCY
<LCO 3.4.6.1.d> <4.4.6.1.4 4.4.6.1.4.a.2> SR 3.4.11.6	-------------------NOTE------------------- Not required to be performed until 30 minutes after RCS temperature ≤ 80°F 85 for Unit 1 and ≤ 77°F for Unit 2 in MODE 4. $\boxed{1}$ --- Verify reactor vessel flange and head flange temperatures are ~~within the limits specified in the PTLR~~ ≥ 72°F for Unit 1 and ≥ 86°F for Unit 2 $\boxed{1}$.	30 minutes
<LCO 3.4.6.1.d> <4.4.6.1.4 4.4.6.1.4.a.1> SR 3.4.11.7	-------------------NOTE------------------- Not required to be performed until 12 hours after RCS temperature ≤ 100°F 92°F for Unit 1, and ≤ 106°F for Unit 2 $\boxed{1}$ in MODE 4. --- Verify reactor vessel flange and head flange temperatures are ~~within the limits specified in the PTLR~~ ≥ 72°F for Unit 1 and ≥ 86°F for Unit 2 $\boxed{1}$.	12 hours

Insert Figures 3.4.11-1, 3.4.11-2, 3.4.11-4, 3.4.11-5, and 3.4.11-6 $\boxed{1}$

⟨CTS⟩

⟨Figure 3.4.6.1-1⟩



P-T Curves for Hydrostatic or Leak Testing

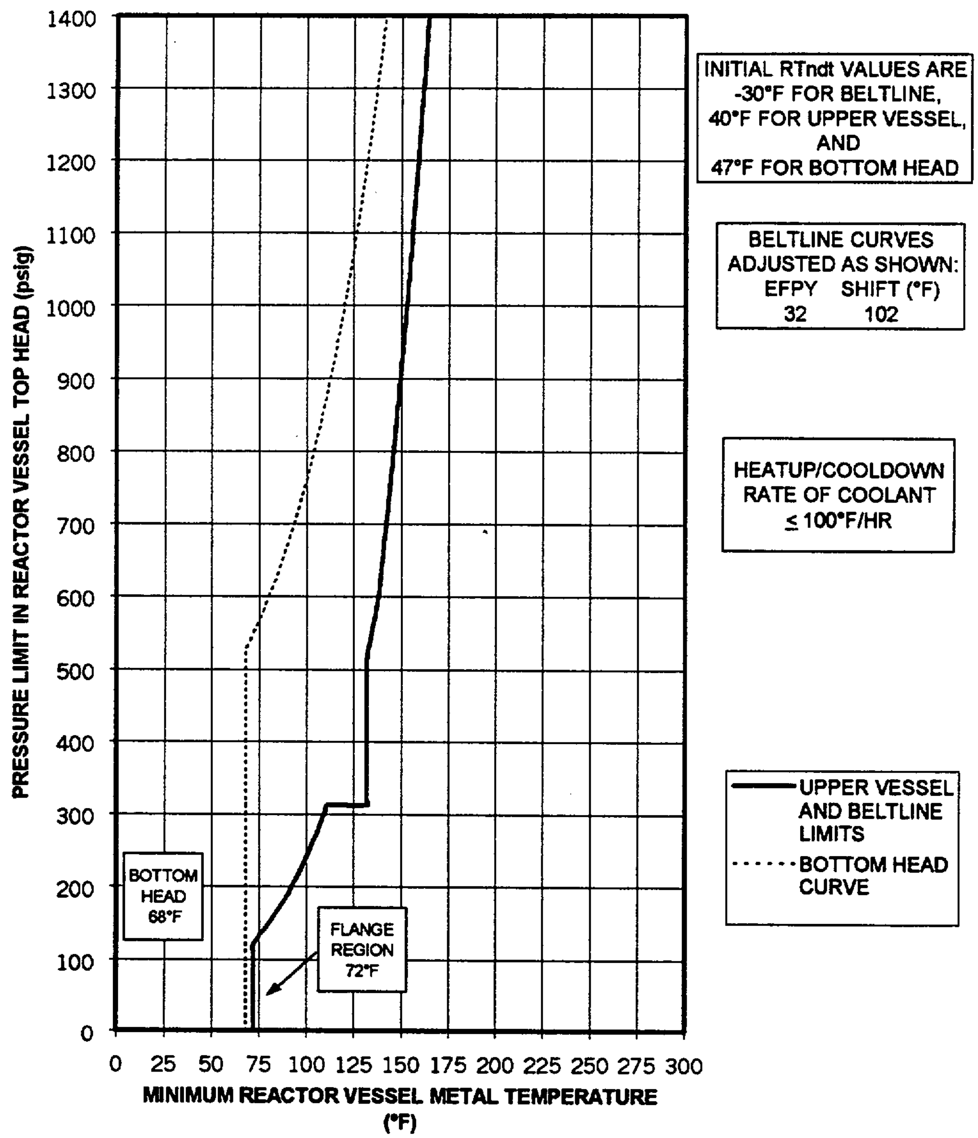
Figure 3.4.11-1 (Page 1 of 1)

Unit 1
Minimum Reactor Vessel Metal Temperature vs.
Reactor Vessel Pressure (Valid up to 32 EFPY)

<CTS>

<Figure 3.4.6.1-1a>

Insert Figure 3.4.11-2



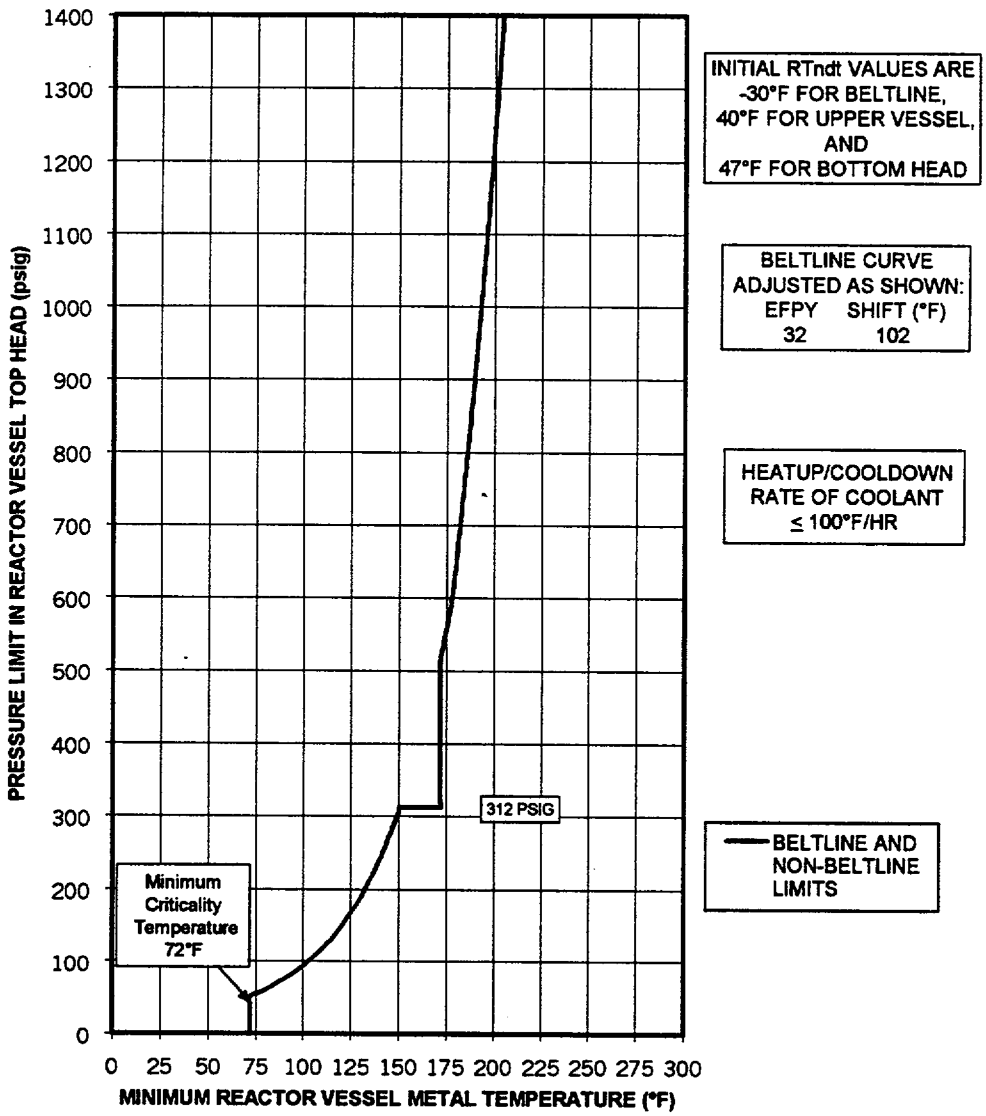
P-T Curves for Heatup by Non-Nuclear Means, Cooldown Following A Nuclear Shutdown and Low Power Physics Testing

Figure 3.4.11-2 (Page 1 of 1)

Unit 1
Minimum Reactor Vessel Metal Temperature vs.
Reactor Vessel Pressure (Valid up to 32 EFPY)

⟨CTS⟩

⟨Figure 3.4.6.1-1b⟩



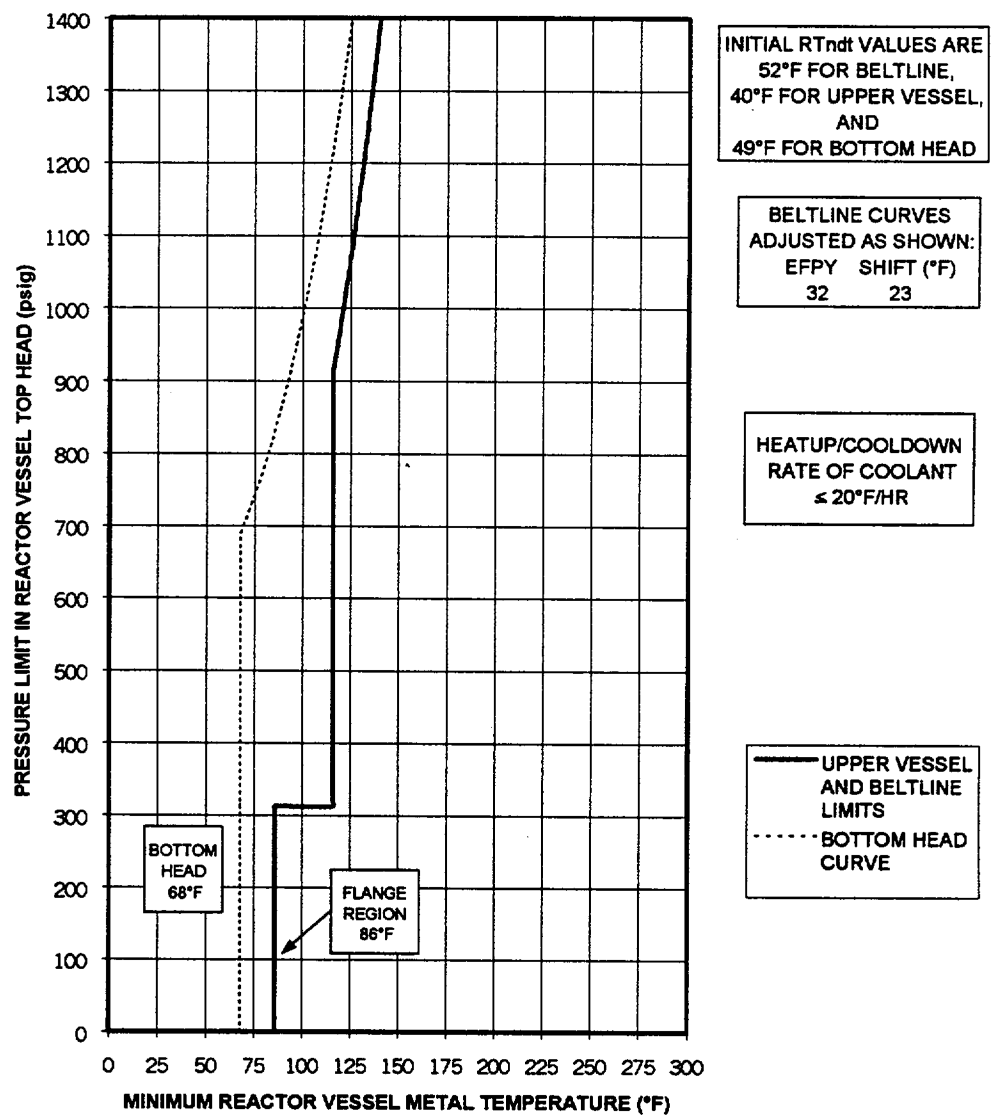
P-T Curves for Operation with a Core Critical
other than Low Power Physics Testing

Figure 3.4.11-3 (Page 1 of 1)

Unit 1
Minimum Reactor Vessel Metal Temperature vs.
Reactor Vessel Pressure (Valid up to 32 EFPY)

⟨CTS⟩

⟨Figure 3.4.6.1-1⟩

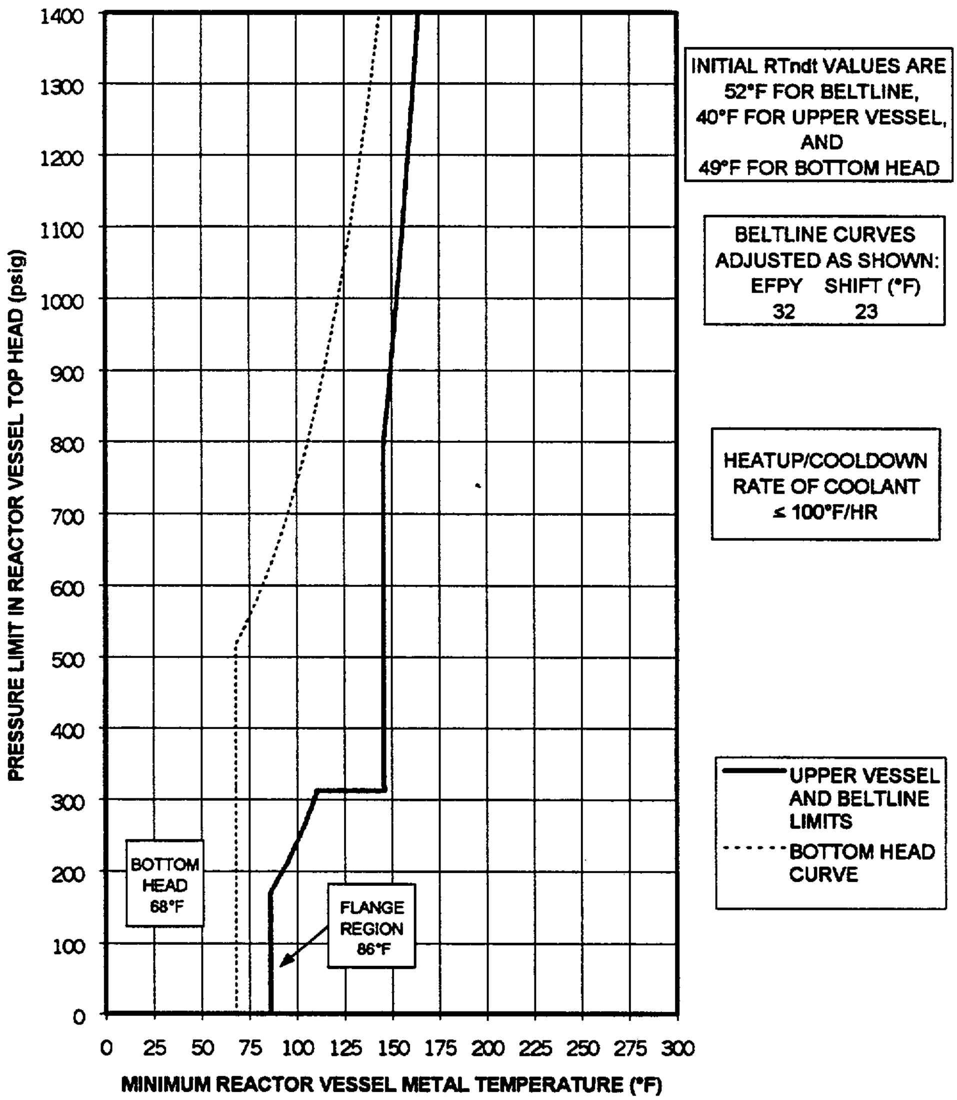


P-T Curves for Hydrostatic or Leak Testing

Figure 3.4.11-4 (Page 1 of 1)

Unit 2
Minimum Reactor Vessel Metal Temperature vs.
Reactor Vessel Pressure (Valid up to 32 EFPY)

Insert Figure 3.4.11-5



P-T Curves for Heatup by Non-Nuclear Means, Cooldown Following A Nuclear Shutdown and Low Power Physics Testing

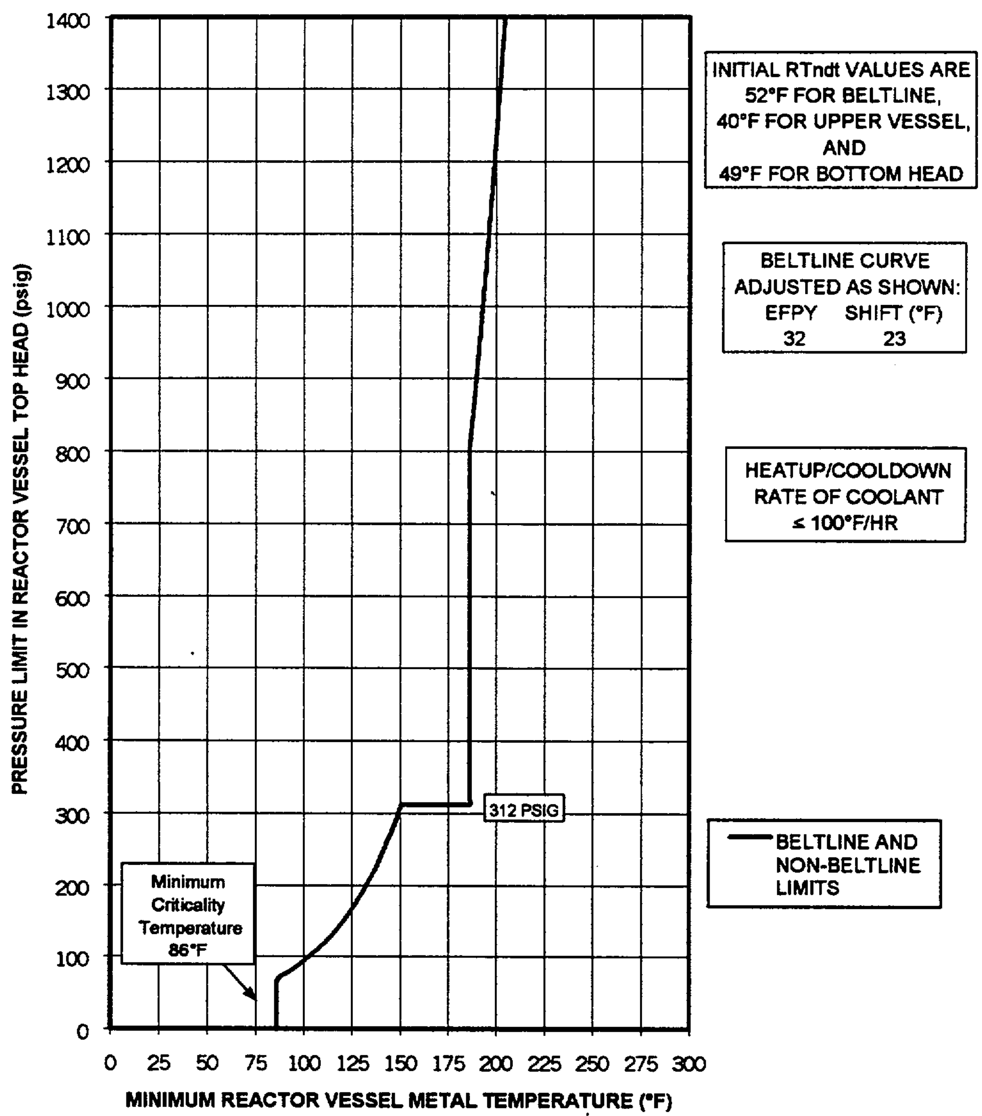
Figure 3.4.11-5 (Page 1 of 1)

Unit 2
Minimum Reactor Vessel Metal Temperature vs.
Reactor Vessel Pressure (Valid up to 32 EFPY)

⟨CTS⟩

⟨Figure 3.4.6.1-1b⟩

Insert Figure 3.4.11-6



P-T Curves for Operation with a Core Critical
other than Low Power Physics Testing

Figure 3.4.11-6 (Page 1 of 1)

Unit 2
Minimum Reactor Vessel Metal Temperature vs.
Reactor Vessel Pressure (Valid up to 32 EFPY)

JUSTIFICATION FOR DEVIATIONS FROM NUREG-1434, REVISION 1
ITS: 3.4.11 - RCS PRESSURE AND TEMPERATURE (P/T) LIMITS

1. The utilization of a Pressure and Temperature Limits Report (PTLR) requires the development, and NRC approval, of detailed methodologies for future revisions to P/T limits. At this time, LaSalle does not have the necessary methodologies submitted to the NRC for review and approval. Therefore, the proposed presentation removes references to the PTLR and proposes that the specific limits and curves be included in the P/T Limits Specification (ITS 3.4.11).

2. Editorial changes have been made to achieve consistency with the Writer's Guide.

3. The bracketed allowance has been deleted since it is not applicable to LaSalle.

4. TSTF-353 adds two bracketed Surveillances that verify coolant temperatures prior to increasing flow or power when in single loop operation. This TSTF has not been adopted since the Surveillances are not required in the current LaSalle 1 and 2 Technical Specifications (i.e., current licensing basis does not include these Surveillances). The coolant temperature verifications are only required in the CTS when starting an idle recirculation pump, and these verifications have been maintained in the ITS.

<CTS>

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.12 Reactor Steam Dome Pressure

1020 [1]

<LCO 3.4.6.2> LCO 3.4.12 The reactor steam dome pressure shall be ≤ ~~[1045]~~ 1020 psig.

<Appl 3.4.6.2> APPLICABILITY: MODES 1 and 2.

ACTIONS

	CONDITION	REQUIRED ACTION	COMPLETION TIME
<3.4.6.2 Act>	A. Reactor steam dome pressure not within limit.	A.1 Restore reactor steam dome pressure to within limit.	15 minutes
<3.4.6.2 Act>	B. Required Action and associated Completion Time not met.	B.1 Be in MODE 3.	12 hours

SURVEILLANCE REQUIREMENTS

	SURVEILLANCE	FREQUENCY
<4.4.6.2>	SR 3.4.12.1 Verify reactor steam dome pressure is ≤ ~~[1045]~~ 1020 psig. [1]	12 hours

JUSTIFICATION FOR DEVIATIONS FROM NUREG-1434, REVISION 1
ITS: 3.4.12 - REACTOR STEAM DOME PRESSURE

1. The brackets have been removed and the proper plant specific information/value has been provided.

B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.1 Recirculation Loops Operating

BASES

BACKGROUND

1

at a faster rate

2

The Reactor Coolant Recirculation System is designed to provide a forced coolant flow through the core to remove heat from the fuel. The forced coolant flow removes more heat from the fuel than would be possible with just natural circulation. The forced flow, therefore, allows operation at significantly higher power than would otherwise be possible. The recirculation system also controls reactivity over a wide span of reactor power by varying the recirculation flow rate to control the void content of the moderator. 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. Each external loop contains a two speed motor driven recirculation pump, a flow control valve, associated piping, jet pumps, valves, and instrumentation. The recirculation loops are part of the reactor coolant pressure boundary and are located inside the drywell structure. The jet pumps are reactor vessel internals.

2 1

The recirculated coolant consists of saturated water from the steam separators and dryers that has been subcooled by incoming feedwater. This water passes down the annulus between the reactor vessel wall and the core shroud. A portion of the coolant flows from the vessel, through the two external recirculation loops, and becomes the driving flow for the jet pumps. Each of the two external recirculation loops discharges high pressure flow into an external manifold, from which individual recirculation inlet lines are routed to the jet pump risers within the reactor vessel. The remaining portion of the coolant mixture in the annulus becomes the suction flow for the jet pumps. This flow enters the jet pump at suction inlets and is accelerated by the driving flow. The drive flow and suction flow are mixed in the jet pump throat section. The total flow then passes through the jet pump diffuser section into the area below the core (lower plenum), gaining sufficient head in the process to drive the required flow upward through the core.

1

and result in partial pressure recovery

(continued)

BASES

BACKGROUND (continued)

The subcooled water enters the bottom of the fuel channels and contacts the fuel cladding, where heat is transferred to the coolant. As it rises, the coolant begins to boil, creating steam voids within the fuel channel that continue until the coolant exits the core. Because of reduced moderation, the steam voiding introduces negative reactivity that must be compensated for to maintain or to increase reactor power. The recirculation flow control allows operators to increase recirculation flow and sweep some of the voids from the fuel channel, overcoming the negative reactivity void effect. Thus, the reason for having variable recirculation flow is to compensate for reactivity effects of boiling over a wide range of power generation (i.e., ~~55~~ approximately 65 to 100% RTP) without having to move control rods and disturb desirable flux patterns. <Insert Background> 1

Each recirculation loop is manually started from the control room. The recirculation flow control valves provide regulation of individual recirculation loop drive flows. The flow in each loop can be manually or automatically controlled.

APPLICABLE SAFETY ANALYSES

The operation of the Reactor ~~Coolant~~ Recirculation System is an initial condition assumed in the design basis loss of coolant accident (LOCA) (Ref. 1). 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 initial core flow decrease is rapid because the recirculation pump in the broken loop ceases to pump reactor coolant to the vessel almost immediately. The pump in the intact loop coasts down relatively slowly. This pump coastdown governs the core flow response for the next several seconds until the jet pump suction is uncovered (Ref. ~~1~~ 2). The analyses assume that both loops are operating at the same flow prior to the accident. However, the LOCA analysis was reviewed for the case with a flow mismatch between the two loops, with the pipe break assumed to be in the loop with the higher flow. While the flow coastdown and core response are potentially more severe in this assumed case (since the intact loop starts at a lower flow rate and the core response is the same as if both loops were operating at a lower flow rate), a small mismatch has been determined to be acceptable based on engineering judgement.

(continued)

INSERT BACKGROUND

In addition, the combination of core flow and THERMAL POWER is normally maintained such that core thermal-hydraulic oscillations do not occur. These oscillations can occur during two recirculation loop operation, single recirculation loop, and no recirculation loop operation. Plant procedures include requirements of this LCO as well as other vendor and NRC recommended requirements and actions to minimize the potential of core thermal-hydraulic oscillations.

BASES

APPLICABLE SAFETY ANALYSES (continued)

The recirculation system is also assumed to have sufficient flow coastdown characteristics to maintain fuel thermal margins during abnormal operational transients (Ref. 2), which are analyzed in Chapter 15 of the FSAR. U [1]

A plant specific LOCA 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 Emergency Core Cooling System response will provide adequate core cooling, provided the APLHGR requirements are modified accordingly (Ref. 3).

The transient analyses ~~of~~ in [2] Chapter 15 of the U [1] FSAR have also been performed for single recirculation loop operation (Ref. 3) and demonstrate sufficient flow coastdown characteristics to maintain fuel thermal margins during the abnormal operational transients analyzed provided the MCPR [4] and the Rod Block Monitor (RBM) requirements are modified. During single recirculation loop operation, modification to the Reactor Protection System average power range monitor (APRM) ~~instrument setpoints~~ Allowable Values [3] is also required to account for the different relationships between recirculation drive flow and reactor core flow. The APLHGR and MCPR ~~setpoints~~ limits for single loop operation are specified in the COLR. [3] The Rod Block Monitor - Upscale Allowable Value is specified in the COLR. [4] The APRM ~~f~~Flow ~~b~~Biased ~~s~~Simulated ~~t~~Thermal ~~p~~Power [3] - Upscale Allowable Value ~~setpoint~~ is in LCO 3.3.1.1, "Reactor Protection System (RPS) Instrumentation." [3] Insert ASA [1]

Recirculation loops operating satisfies Criterion 2 of ~~the NRC Policy Statement~~ 10 CFR 50.36 (c)(2)(ii). [1]

LCO

Two recirculation loops are normally [3] required to be in operation with their flows matched within the limits specified in SR 3.4.1.1 to ensure that during a LOCA caused by a break of the piping of one recirculation loop the assumptions of the LOCA analysis are satisfied. With the limits specified in SR 3.4.1.1 not met, the recirculation loop with the lower flow must be considered not in operation. With only one recirculation loop in operation, modifications to the required APLHGR limits (LCO 3.2.1, "AVERAGE PLANAR LINEAR HEAT GENERATION RATE (APLHGR)"), MCPR limits (LCO 3.2.2, [4] "MINIMUM CRITICAL POWER RATIO (MCPR)"), ~~and~~ APRM Flow Biased Simulated Thermal Power [3] - Upscale Allowable Value ~~High setpoint~~ (LCO 3.3.1.1), and the Rod Block Monitor - Upscale Allowable Value (LCO 3.3.2.1) [4] ~~may~~ must [3] be

(continued)

INSERT ASA

Safety analyses performed in References 1, 2, and 3 implicitly assume core conditions are stable. However, during operation at the high power/low flow region of the operating domain, a small probability of limit cycle neutron flux oscillations exists depending on combinations of operating conditions (e.g., power shape, bundle power, and bundle flow).

General Electric (GE) Service Information Letter (SIL) No. 380 (Ref. 4) addressed boiling instability and made several recommendations. In this SIL, the power/flow operating map was divided into several regions of varying concern. It also discussed the objectives and philosophy of "detect and suppress."

NRC Generic Letter 86-02 (Ref. 5) discussed both the GE and Siemens stability methodology and stated that due to uncertainties, 10 CFR 50, Appendix A, General Design Criteria (GDC) 10 and 12 could not be met using available analytical procedures on a BWR. The Generic Letter discussed SIL 380 and stated that GDC 10 and 12 could be met by imposing SIL 380 recommendations in operating regions of potential instability. The NRC concluded that regions of potential instability constituted decay ratios of 0.8 and greater by the GE methodology and 0.75 by the Siemens methodology. Figure 3.4.1-1 was generated as an interim solution to provide an increased margin of safety until the investigation is completed (Ref. 6).

BASES

LCO (continued)

applied to allow continued operation consistent with the assumptions of Reference 3. <Insert LCO> 1

APPLICABILITY

In MODES 1 and 2, requirements for operation of the Reactor ~~Coolant~~ Recirculation System are necessary since there is considerable energy in the reactor core and the limiting design basis transients and accidents are assumed to occur. 2

In MODES 3, 4, and 5, the consequences of an accident are reduced and the coastdown characteristics of the recirculation loops are not important.

ACTIONS

<Insert ACTIONS> 4

With both recirculation loops operating but the flows not matched, the flows must be matched within 2 hours. If matched flows are not restored, the recirculation loop with lower flow must be declared "not in operation" as required by Required Action F.1.

~~A.1~~ F.1 and G.1 4

With the requirements of the LCO not met ~~the recirculation loops~~ for reasons other than Conditions A, C, D, and F (e.g., one loop is "not in operation") 4 compliance with the LCO must be restored ~~to operation with matched flows~~ within ~~24~~ 12 4 hours. A recirculation loop is considered not in operation when the pump in that loop is idle or when the mismatch between total jet pump ~~flows of the two loops is greater than required limits.~~ ~~The loop with the lower flow must be considered not in operation.~~ for greater than 2 hours (i.e., Required Action F.1 has been taken) 4 Should a LOCA occur with one recirculation loop not in operation, the core flow coastdown and resultant core response may not be bounded by the LOCA analyses. Therefore, only a limited time is allowed to restore the inoperable loop to operating status.

Alternatively, if the single loop requirements of the LCO are applied to the APLHGR and MCPR 3 operating limits and RPS ~~setpoints~~ and RBM Allowable Values 4, operation with only one recirculation loop would satisfy the requirements of the LCO and the initial conditions of the accident sequence.

The ~~24~~ 2 hour and 12 4 hour Completion ~~Time is~~ Times are based on the low probability of an accident occurring during this time period, on a reasonable time to complete the Required Action, and on frequent core monitoring by operators allowing abrupt changes in core flow conditions to be quickly detected.

This Required Action does not require tripping the recirculation pump in the lowest flow loop when the mismatch between total jet pump flows of the two loops is greater than the required limits. However, in cases where large 4

(continued)

INSERT LCO

In addition, during two-loop and single-loop operation, the combination of core flow and THERMAL POWER must be in Region III of Figure 3.4.1-1 to ensure core thermal-hydraulic oscillations do not occur.

INSERT ACTIONS

A.1, A.2, and A.3

With one or two recirculation loops in operation in Region II of Figure 3.4.1-1, the plant is operating in a region where the potential for thermal-hydraulic oscillations exists. To ensure oscillations are not occurring, APRM and LPRM neutron flux noise levels must be verified to be less than or equal to the larger of either 3 times the baseline noise levels or 10% peak-to-peak (Required Action A.1 and A.2) when Region II is entered. For the LPRM neutron flux noise verification, detector levels A and C of one LPRM string per core octant plus detector levels A and C of one LPRM string in the center region of the core should be monitored. Prompt action to monitor APRM and LPRM neutron flux noise levels should be taken to ensure oscillations are not occurring.

The 45 minute Completion Time of Required Actions A.1 and A.2 provides a reasonable time to stabilize operation in Region II and verify the neutron flux noise levels are within limits. A verification of the APRM and LPRM neutron flux noise levels once per 12 hours following the initial verification provides frequent periodic information of neutron flux noise levels to verify stable steady state operation. Also, a verification of neutron flux noise levels after any THERMAL POWER increase of $\geq$ 5% RTP while in Region II provides indication of operational stability following a potential for change of the thermal-hydraulic properties of the system.

In addition, a verification that one or both recirculation loops are not operating within Region I of Figure 3.4.1-1 (Required Action A.3) is required to be performed once per 12 hours. The Completion Time of once per 12 hours is reasonable based on operating experience and the operator's knowledge of reactor status, including changes in reactor power and core flow.

INSERT ACTIONS (CONTINUED)

B.1

If evidence of approaching reactor instability occurs (i.e., APRM or LPRM neutron flux noise levels exceed the associated limit of Required Actions A.1 or A.2, as applicable) while operating in Region II of Figure 3.4.1-1, prompt action should be taken to restore the APRM or LPRM neutron flux noise levels to within the associated limit or exit Region II of Figure 3.4.1-1. This may be accomplished by either increasing core flow by recirculation loop flow control valve manipulation or reduction of THERMAL POWER by control rod insertion. The 2 hour Completion Time is reasonable to restore plant parameters in an orderly manner and without challenging plant systems.

C.1

With one or both recirculation loops in operation in Region I of Figure 3.4.1-1, the plant is operating in a region where the potential for thermal-hydraulic oscillations is increased and sufficient margin may not be available for operator response to suppress potential thermal-hydraulic oscillations. As a result, prompt action should be taken to exit Region I of Figure 3.4.1-1. This may be accomplished by either increasing core flow by recirculation loop flow control valve manipulation or reduction of THERMAL POWER by control rod insertion. The 2 hour Completion Time is reasonable to restore plant parameters in an orderly manner and without challenging plant systems.

INSERT ACTIONS (CONTINUED)

D.1, D.2, and D.3

With no recirculation loops in service, the probability of thermal-hydraulic oscillations is greatly increased. Therefore, prompt action should be taken to ensure oscillations are not occurring by verifying APRM and LPRM neutron flux noise levels are ≤ 10% peak-to-peak. If neutron flux noise levels are discovered to be > 10% peak-to-peak at anytime while in this Condition, Condition E must be immediately entered.

Also, prompt action should be taken to reduce THERMAL POWER low enough to avoid the region of potential instability in natural circulation (i.e., reduce THERMAL POWER below 36% RTP). The 2 hour Completion Time provides are reasonable time to restore operation to Region III of Figure 3.4.1-1.

In addition, with no recirculation loops in operation, plant operation is not allowed to continue in MODE 1 or 2. Therefore, the unit is required to be brought to a MODE in which the LCO does not apply. The allowed Completion Time of 12 hours is reasonable, based on operating experience, to reach MODE 3 in an orderly manner and without challenging plant systems.

E.1

In the event no recirculation loops are in operation and evidence is indicated of approaching reactor instability (i.e., APRM or LPRM neutron flux noise levels exceed the associated limit) or APRM or LPRM neutron flux noise levels cannot be restored within 2 hours while in Region II of Figure 3.4.1-1, action must be immediately initiated to eliminate the potential for a thermal-hydraulic instability event. As such, the reactor mode switch must be immediately placed in the shutdown position.

BASES

F.1 and G.1 [4]

ACTIONS

A.1 (continued)

flow mismatches occur, low flow or reverse flow can occur in the low flow loop jet pumps, causing vibration of the jet pumps. If zero or reverse flow is detected, the condition should be alleviated by changing flow control valve position to re-establish forward flow or by tripping the pump. [4]

H [4]

B.1

If [4]

~~With no recirculation loops in operation, or~~ the Required Action and associated Completion Time of Condition ~~A~~ G not ~~met~~ is met, the unit is required to be brought to a MODE in which the LCO does not apply. To achieve this status, the plant must be brought to MODE 3 within 12 hours. In this condition, the recirculation loops are not required to be operating because of the reduced severity of DBAs and minimal dependence on the recirculation loop coastdown characteristics. The allowed Completion Time of 12 hours is reasonable, based on operating experience, to reach MODE 3 from full power conditions in an orderly manner and without challenging plant systems. [4]

SURVEILLANCE REQUIREMENTS

SR 3.4.1.1

APLHGR and [4]

This SR ensures the recirculation loop flows are within the allowable limits for mismatch. At low core flow (i.e., < ~~[~~70~~]~~% of rated core flow), the MCPR requirements provide larger margins to the fuel cladding integrity Safety Limit such that the potential adverse effect of early boiling transition during a LOCA is reduced. A larger flow mismatch can therefore be allowed when core flow is < ~~[~~70~~]~~% of rated core flow. The recirculation loop jet pump flow, as used in this Surveillance, is the summation of the flows from all of the jet pumps associated with a single recirculation loop. [5]

[3]

The mismatch is measured in terms of percent of rated core flow. If the flow mismatch exceeds the specified limits, the loop with the lower flow is considered ~~inoperable~~ not in operation. This SR is not required when both loops are not in operation since the mismatch limits are meaningless during single loop or natural circulation operation. The Surveillance must be

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.4.1.1 (continued)

performed within 24 hours after both loops are in operation. The 24 hour Frequency is consistent with the Frequency for jet pump OPERABILITY verification and has been shown by operating experience to be adequate to detect off normal jet pump loop flows in a timely manner.

REFERENCES

1 [5]

1. UFSAR, Section ~~6.3.3.4.~~ 6.3 and 15.6.5.
2. UFSAR, ~~Section~~ ~~5.5.1.4.~~ Appendix G.3.1.2.
~~3. Plant specific analysis for single loop operation.~~

<Insert SR 3.4.1.2> 4

1

4. GE Service Information Letter (SIL) No. 380, "BWR Core Thermal Hydraulic Stability," Revision 1, February 10, 1984.
5. NRC Generic Letter 86-02, "Technical Resolution of Generic Issue B-19, Thermal Hydraulic Stability," January 22, 1986.
6. NRC Safety Evaluation supporting Amendment No. 60 to Facility Operating License No. NPF-11, and Amendment No. 40 to Facility Operating License No. NPF-18, Commonwealth Edison Company, LaSalle County Station, Units 1 and 2, dated September 7, 1988.

5

3. UFSAR, Section 6.B.

INSERT SR 3.4.1.2

SR 3.4.1.2

The SR ensures the combination of core flow and THERMAL POWER are within the appropriate limits to prevent inadvertent entry into a region of potential thermal-hydraulic instability. At low recirculation loop flow and high reactor power, the reactor exhibits increased susceptibility to thermal-hydraulic instability. Figure 3.4.1-1 is based on guidance provided in References 4 and 5. The 24 hour Frequency is based on operating experience and the operator's knowledge of the reactor status, including significant changes in THERMAL POWER and core flow.

1. Changes have been made (additions, deletions, and/or changes to the NUREG) to reflect the plant specific nomenclature, number, reference, system description, or analysis description.

2. Editorial change made for enhanced clarity or to be consistent with similar statements in other places in the Bases.

3. Changes have been made to more closely match the LCO requirement.

4. Changes have been made to reflect those changes made to the Specification. The following requirements have been renumbered, where applicable, to reflect the changes.

5. The brackets have been removed and the proper plant specific information/value has been provided.

B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.2 Flow Control Valves (FCVs)

BASES

BACKGROUND

The Reactor Coolant Recirculation System is described in the Background section of the Bases for LCO 3.4.1, "Recirculation Loops Operating," which discusses the operating characteristics of the system and how this affects the design basis transient and accident analyses. The ~~jet pumps and the~~ FCVs are part of the Reactor ~~Coolant~~ Recirculation System. ~~The jet pumps are described in the Bases for LCO 3.4.3, "Jet Pumps."~~ 1 2

The Recirculation Flow Control System consists of the electronic and hydraulic components necessary for the positioning of the two hydraulically actuated FCVs. The recirculation loop flow rate can be rapidly changed within the expected flow range, in response to rapid changes in system demand. Limits on the system response are required to minimize the impact on core flow response during certain accidents and transients. Solid state control logic will generate an FCV "motion inhibit" signal in response to any one of several hydraulic power unit or analog control circuit failure signals. The "motion inhibit" signal causes hydraulic power unit shutdown and hydraulic isolation such that the FCVs fail "as is."

APPLICABLE SAFETY ANALYSES

The FCV stroke rate is limited to ≤ 11% per second in the opening and closing directions on a control signal failure of maximum demand. This stroke rate is an assumption of the analysis of the recirculation flow control failures on decreasing and increasing flow (Refs. 1 and 2). The closure of a recirculation FCV concurrent with a loss of coolant accident (LOCA) ~~has been analyzed and found to be acceptable for a maximum closure rate of 11% of strokes/per second (Ref. 3)~~. 1

1 Insert ASA

Flow control valves satisfy Criterion 2 of ~~the NRC Policy Statement~~ 10 CFR 50.36(c)(2)(ii).

was analyzed during initial licensing and found to be acceptable for a maximum closure rate of 11% of stroke per second, since this event involves multiple failures

---

(continued)

<u>INSERT ASA</u>

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 initial core flow decrease is rapid because the recirculation pump in the broken loop ceases to pump almost immediately since it has lost suction. The pump in the intact loop coasts down relatively slowly. This pump coastdown governs the core flow response for the next several seconds (Ref. 3), because the FCV is assumed to fail "as is" due to a motion inhibit as a result of a high drywell pressure interlock. In addition,

BASES (continued)

LCO

An FCV in each operating recirculation loop must be OPERABLE to ensure that the assumptions of the design basis transient and accident analyses are satisfied.

APPLICABILITY

In MODES 1 and 2, the FCVs are required to be OPERABLE, since during these conditions there is considerable energy in the reactor core, and the limiting design basis transients and accidents are assumed to occur. In MODES 3, 4, and 5, the consequences of a transient or accident are reduced and OPERABILITY of the flow control valves is not important.

ACTIONS

A Note has been provided to modify the ACTIONS related to FCVs. Section 1.3, Completion Times, specifies once a Condition has been entered, subsequent divisions, subsystems, components or variables expressed in the Condition, discovered to be inoperable or not within limits, will not result in separate entry into the Condition. Section 1.3 also specifies Required Actions of the Condition continue to apply for each additional failure, with Completion Times based on initial entry into the Condition. However, the Required Actions for inoperable FCVs provide appropriate compensatory measures for separate inoperable FCVs. As such, a Note has been provided that allows separate Condition entry for each inoperable FCV. [1]

Could result in a more severe coolant flow decrease transient. Both conditions could

A.1

With one or two required FCVs inoperable, the assumptions of the design basis transient and accident analyses may not be met and the inoperable FCV must be returned to OPERABLE status or hydraulically locked within 4 hours.

[1]

Insert ACTION A.1

[1] Opening an FCV faster than the limit could result in a more severe flow runout transient, resulting in violation of the Safety Limit MCPR. Closing an FCV faster than the limit assumed in the LOCA analysis (~~Refs. 1 and 2~~) could affect the recirculation flow coastdown, resulting in higher peak clad temperatures. Therefore, if an FCV is inoperable ~~due to stroke times faster than the limits~~, deactivating the valve will essentially lock the valve in position, which ③-[1] [4]

(continued)

INSERT ACTION A.1

The FCVs are designed to lockup (high drywell pressure interlock) under LOCA conditions. When the FCVs "lock-up", the recirculation flow coastdown is adequate and the resulting calculated clad temperatures are acceptable. In addition, it has been calculated with the FCVs closing at the specified limit, the resulting calculated clad temperatures will also be acceptable. Closing an FCV faster than the limit

BASES

ACTIONS

A.1 (continued)

will prohibit the FCV from adversely affecting the DBA and transient analyses. Continued operation is allowed in this Condition.

The 4 hour Completion Time is a reasonable time period to complete the Required Action, while limiting the time of operation with an inoperable FCV.

B.1

If the FCVs are not deactivated ("locked up") within the associated Completion Time, the unit must be brought to a MODE in which the LCO does not apply. To achieve this status, the unit must be brought to at least MODE 3 within 12 hours. This brings the unit to a condition where the flow coastdown characteristics of the recirculation loop are not important. The allowed Completion Time of 12 hours is reasonable, based on operating experience, to reach MODE 3 from full power conditions in an orderly manner and without challenging unit systems.

SURVEILLANCE REQUIREMENTS

SR 3.4.2.1

4-way 1

Hydraulic power unit pilot operated isolation valves located between the servo valves and the common "open" and "close" lines are required to close in the event of a loss of hydraulic pressure. When closed, these valves inhibit FCV motion by blocking hydraulic pressure from the servo valve to the common open and close lines as well as to the alternate subloop. This Surveillance verifies FCV lockup on a loss of hydraulic pressure.

24 3

The ~~18~~ month Frequency is based on the need to perform this Surveillance under the conditions that apply during a plant outage and the potential for an unplanned transient if the Surveillance were performed with the reactor at power. Operating experience has shown these components usually pass the SR when performed at the ~~18~~ month Frequency. Therefore, the Frequency was concluded to be acceptable from a reliability standpoint.

24

(continued)

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.4.2.2

This SR ensures the overall average rate of FCV movement at all positions is maintained within the analyzed limits.

24 3

The ~~[18]~~ month Frequency is based on the need to perform this Surveillance under the conditions that apply during a plant outage and the potential for an unplanned transient if the Surveillance were performed with the reactor at power. Operating experience has shown these components usually pass the SR when performed at the ~~[18]~~ 24 month Frequency. Therefore, the Frequency was concluded to be acceptable from a reliability standpoint.

REFERENCES

U 1

1. UFSAR, Section ~~[~~15.3.2~~]~~. 3
2. UFSAR, Section ~~[~~15.4.5~~]~~.
3. ~~[Plant specific Safety Evaluation Report.]~~ UFSAR, Appendix G. 3

JUSTIFICATION FOR DEVIATIONS FROM NUREG-1434, REVISION 1
ITS BASES: 3.4.2 - FLOW CONTROL VALVES (FCVs)

1. Changes have been made (additions, deletions, and/or changes to the NUREG) to reflect the plant specific nomenclature, number, reference, system description, or analysis description.

2. The Specification deals with the flow control valves and there is no reason to reference the jet pumps. Therefore, the reference to jet pumps has been deleted. This concept (not referencing in a "subcomponent" Bases the other "subcomponents" of the associated system) is consistent with other sections of the ITS Bases.

3. The brackets have been removed and the proper plant specific information/value has been provided.

4. Changes have been made to more closely match the LCO requirement.

B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.3 Jet Pumps

BASES

BACKGROUND

$\boxed{1}$

The Reactor ~~Coolant~~ Recirculation System is described in the Background section of the Bases for LCO 3.4.1, "Recirculation Loops Operating," which discusses the operating characteristics of the system and how these characteristics affect the Design Basis Accident (DBA) analyses.

$\boxed{1}$

The jet pumps are part of the Reactor ~~Coolant~~ Recirculation System and are designed to provide forced circulation through the core to remove heat from the fuel. The jet pumps are located in the annular region between the core shroud and the vessel inner wall. Because the jet pump suction elevation is at two thirds core height, the vessel can be reflooded and coolant level maintained at two thirds core height even with the complete break of the recirculation loop pipe that is located below the jet pump suction elevation. 10 $\boxed{1}$

Each reactor coolant recirculation loop contains ~~12~~ jet pumps. Recirculated coolant passes down the annulus between the reactor vessel wall and the core shroud. A portion of the coolant flows from the vessel, through the two external recirculation loops, and becomes the driving flow for the jet pumps. Each of the two external recirculation loops discharges high pressure flow into an external manifold from which individual recirculation inlet lines are routed to the jet pump risers within the reactor vessel. The remaining portion of the coolant mixture in the annulus becomes the suction flow for the jet pumps. This flow enters the jet pump at suction inlets and is accelerated by the drive flow. The drive flow and suction flow are mixed in the jet pump throat section. The total flow then passes through the jet pump diffuser section into the area below the core (lower plenum), gaining sufficient head in the process to drive the required flow upward through the core.

APPLICABLE SAFETY ANALYSES

Jet pump OPERABILITY is an explicit assumption in the design basis loss of coolant accident (LOCA) analysis evaluated in Reference 1.

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

The capability of reflooding the core to two-thirds core height is dependent upon the structural integrity of the jet pumps. If the structural system, including the beam holding a jet pump in place, fails, jet pump displacement and performance degradation could occur, resulting in an increased 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. } 3

Jet pumps satisfy Criterion 2 3 of ~~the NRC Policy Statement.~~ 10CFR50.36(c)(2)(ii) } 1

LCO

The structural failure of any of the jet pumps could cause significant degradation in the ability of the jet pumps to allow reflooding to two thirds core height during a LOCA. OPERABILITY of all jet pumps is required to ensure that operation of the Reactor Coolant Recirculation System will be consistent with the assumptions used in the licensing basis analysis (Ref. 1). 1

APPLICABILITY

In MODES 1 and 2, the jet pumps are required to be OPERABLE since there is a large amount of energy in the reactor core and since the limiting DBAs are assumed to occur in these MODES. This is consistent with the requirements for operation of the Reactor Coolant Recirculation System (LCO 3.4.1). 1

In MODES 3, 4, and 5, the Reactor Coolant Recirculation System is not required to be in operation, and when not in operation sufficient flow is not available to evaluate jet pump OPERABILITY.

ACTIONS

A.1

An inoperable jet pump to can increase the blowdown area and reduce the capability of reflooding during a design basis LOCA. If one or more of the jet pumps are inoperable, the plant must be brought to a MODE in which the LCO does not apply. To achieve this status, the plant must be brought to MODE 3 within 12 hours. The allowed Completion Time of } 2

(continued)

BASES

ACTIONS

A.1 (continued)

12 hours is reasonable, based on operating experience, to reach MODE 3 from full power conditions in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.4.3.1

This SR is designed to detect significant degradation in jet pump performance that precedes jet pump failure (Ref. 2). This SR is required to be performed only when the loop has forced recirculation flow since surveillance checks and measurements can only be performed during jet pump operation. The jet pump failure of concern is a complete mixer displacement due to jet pump beam failure. Jet pump plugging is also of concern since it adds flow resistance to the recirculation loop. Significant degradation is indicated if the specified criteria confirm unacceptable deviations from established patterns or relationships. The allowable deviations from the established patterns have been developed based on the variations experienced at plants during normal operation and with jet pump assembly failures (Refs. 2 and 3). Since refueling activities (fuel assembly replacement or shuffle, as well as any modifications 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 re-established 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 baselining new "established patterns", engineering judgement of the daily surveillance results is used to detect significant abnormalities which could indicate a jet pump failure.

2 Any two of the three

3

1 Insert SR 3.4.3.1-1

The recirculation flow control valve (FCV) operating characteristics (loop flow versus FCV position) are determined by the flow resistance from the loop suction through the jet pump nozzles. A change in the relationship indicates a flow restriction, loss in pump hydraulic performance, leak, or new flow path between the recirculation pump discharge and jet pump nozzle. For this criterion, the loop flow versus FCV position relationship must be verified.

1 characteristics

4 may

(continued)

1 When both recirculation loops are operating, the established FCV position should include the loop flow characteristics for two recirculation loop operation. When only one recirculation loop is operating, the established FCV position should include the loop flow characteristics for single loop operation.

INSERT SR 3.4.3.1-1

In addition, during two recirculation loop operation, the jet pump SR should be performed with balanced recirculation loop drive flows (drive flow mismatch less than 5%) to ensure an accurate indication of jet pump performance.

BASES

1

SURVEILLANCE REQUIREMENTS

either the established THERMAL POWER-core flow relationship or the core plate differential pressure-core flow relationship.

SR 3.4.3.1 (continued) calculated 1

Total core flow can be determined from ~~measurements of the recirculation loop drive flows~~. Once this relationship has been established, increased or reduced total core flow ~~for the same recirculation loop drive flow~~ may be an indication of failures in one or several jet pumps.

5 indicated

from the calculated total core flow

Individual jet pumps in a recirculation loop typically do not have the same flow. The unequal flow is due to the drive flow manifold, which does not distribute flow equally to all risers. The ~~flow for~~ jet pump diffuser to lower plenum differential pressure~~s~~ pattern or relationship of one jet pump to the loop average is repeatable. An appreciable change in this relationship is an indication that increased (or reduced) resistance has occurred in one of the jet pumps. ~~This may be indicated by an increase in the relative flow for a jet pump that has experienced beam cracks.~~ 6

5 5

1

When determing calculated total core flow in single recirculation loop operation using the core plate differential pressure- core flow relationship, the calculated total core flow value should be derived using the established core plate differential pressure - core flow relationship for two recirculation loop operation.

The deviations from normal are considered indicative of a potential problem in the recirculation drive flow or jet pump system (Ref. 2). Normal flow ranges and established jet pump ~~flow and~~ differential pressure patterns are established by plotting historical data as discussed in Reference 2. 5

The 24 hour Frequency has been shown by operating experience to be adequate to verify jet pump OPERABILITY and is consistent with the Frequency for recirculation loop OPERABILITY verification.

7

until 24 hours after

This SR is modified by two Notes. Note 1 allows this Surveillance not to be performed ~~until 4 hours~~ after the associated recirculation loop is in operation, since these checks can only be performed during jet pump operation. The ~~4~~ hours is an acceptable time to establish conditions appropriate for data collection and evaluation.

exceeds 7

Note 2 allows this SR not to be performed ~~when~~ THERMAL POWER ~~is ≤~~ 25% RTP. During low flow conditions, jet pump noise approaches the threshold response of the associated flow instrumentation and precludes the collection of repeatable and meaningful data.

The 24 hours is an acceptable time to establish conditions appropriate to perform this SR. 2

(continued)

BASES (continued)

8

REFERENCES

1 u
1. FSAR, Section ~~6.3~~ 5. and Appendices G2.2.2 and G2.2.3.

2. GE Service Information Letter No. 330, June 9, 19~~80~~. 8 3

3. NUREG/CR-3052, "Closeout of IE Bulletin 80-07: BWR Jet Pump Assembly Failure," November 1984. 2

JUSTIFICATION FOR DEVIATIONS FROM NUREG-1434, REVISION 1
ITS BASES: 3.4.3 - JET PUMPS

1. Changes have been made (additions, deletions, and/or changes to the NUREG) to reflect the plant specific nomenclature, number, reference, system description, licensing basis description, or analysis description.

2. Editorial change made for enhanced clarity or to be consistent with similar statements in other places in the Bases.

3. Typographical/grammatical error corrected.

4. The word "may" has been added since a change in the described relationship may be due to other factors.

5. Changes have been made to reflect those changes made to the Specification.

6. This statement has been deleted since it is misleading; an increase in flow could be indicative of other problems.

7. Changes have been made to more closely match the LCO requirements.

8. The brackets have been removed and the proper plant-specific information/value has been provided.

B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.4 Safety/Relief Valves (S/RVs)

BASES

BACKGROUND

The American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code (Ref. 1) 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/relief valves (S/RVs) are selected such that peak pressure in the nuclear system will not exceed the ASME Code limits for the reactor coolant pressure boundary (RCPB).

The S/RVs are located on the main steam lines between the reactor vessel and the first isolation valve within the drywell. Each S/RV discharges steam through a discharge line to a point below the minimum water level in the suppression pool.

2

(However, for the purposes of this LCO, only the safety mode is required)

The S/RVs can actuate by either of two modes: the safety mode or the relief mode. In the safety mode (or spring mode of operation), the direct action of the steam pressure in the main steam lines will act against a spring loaded disk that will pop open when the valve inlet pressure exceeds the spring force. In the relief mode (or power actuated mode of operation), a pneumatic piston ~~or~~ / cylinder and mechanical linkage assembly are used to open the valve by overcoming the spring force, even with the valve inlet pressure equal to 0 psig. The pneumatic operator is arranged so that its malfunction will not prevent the valve disk from lifting if steam inlet pressure reaches the spring lift set pressures. In the relief mode, valves may be opened manually or automatically at the selected preset pressure. ~~Six of the S/RVs providing the relief function also provide the low-low set relief function specified in LCO 3.6.1.6, "Low-Low Set (LLS) Valves."~~ 1 ~~Eight~~ seven 1 of the S/RVs that provide the Safety and 1 relief function are part of the Automatic Depressurization System specified in LCO 3.5.1, "ECCS—Operating." The instrumentation associated with the relief valve function ~~and low-low set relief function is discussed in the Bases for LCO 3.3.6.5, "Relief and Low-Low Set (LLS) Instrumentation," and instrumentation~~ 1 for the ADS function is discussed in LCO 3.3.5.1, "Emergency Core Cooling Systems (ECCS) Instrumentation."

(continued)

BASES (continued)

APPLICABLE SAFETY ANALYSES

The overpressure protection system must accommodate the most severe pressure transient. Evaluations have determined that the most severe transient is the closure of all main steam isolation valves (MSIVs) followed by reactor scram on high neutron flux (i.e., failure of the direct scram associated with MSIV position) (Ref. 2). For the purpose of the analyses, ~~six~~ [3] ten of the S/RVs are assumed to operate in ~~the relief mode, and seven in~~ [1] the safety mode. The analysis results demonstrate that the design S/RV capacity is capable of maintaining reactor pressure below the ASME Code limit of 110% of vessel design pressure (110% x 1250 psig = 1375 psig). This LCO helps to ensure that the acceptance limit of 1375 psig is met during the design basis event.

From an overpressure standpoint, the design basis events are bounded by the MSIV closure with flux scram event described above. [1] <Insert ASA> Reference 3 discusses additional events that are expected to actuate the S/RVs.

S/RVs satisfy Criterion 3 of ~~the NRC Policy Statement~~ [1] 10 CFR 50.36(c)(2)(ii).

LCO

The safety function of ~~seven~~ [4] 17 S/RVs for Unit 1, and 12 S/RVs for Unit 2 [4] is required to be OPERABLE ~~in the safety mode, and an additional seven S/RVs (other than the seven S/RVs that satisfy the safety function) must be OPERABLE in the relief mode~~ [4]. The requirements of this LCO are applicable only to the capability of the S/RVs to mechanically open to relieve excess pressure when the lift setpoint is exceeded (safety mode). In Reference 2, an evaluation was performed to establish the parametric relationship between the peak vessel pressure and the number of OPERABLE S/RVs. The results show that with a minimum of ~~seven~~ [4] 17 S/RVs for Unit 1, and 12 S/RVs in the safety mode ~~and six S/RVs in the relief mode~~ [1] OPERABLE, the ASME Code limit of 1375 psig is not exceeded.

The S/RV [2] safety setpoints are established to ensure the ASME Code limit on peak reactor pressure is satisfied. The ASME Code specifications require the lowest safety valve be set at or below vessel design pressure (1250 psig) and the highest safety valve be set so the total accumulated pressure does not exceed 110% of the design pressure for [2] overpressurization conditions. The transient evaluations in Reference 3 [2] involving the safety mode are based on these setpoints, but also include the additional uncertainties of ±~~1%~~ [1] 3% of the nominal setpoint to account for potential setpoint drift to provide an added degree of conservatism.

(continued)

INSERT ASA

For other pressurization events, such as a turbine trip or generator load rejection with Main Turbine Bypass System failure, the S/RVs are assumed to function. The opening of the valves during the pressurization event mitigates the increase in reactor vessel pressure, which affects the MINIMUM CRITICAL POWER RATIO (MCPR) during these events. The number of S/RVs required to mitigate these events is bounded by the number required to be OPERABLE by the LCO.

BASES

LCO (continued)

Operation with fewer valves OPERABLE than specified, or with setpoints outside the ASME limits, could result in a more severe reactor response to a transient than predicted, possibly resulting in the ASME Code limit on reactor pressure being exceeded.

APPLICABILITY

In MODES 1, 2, and 3, the specified number of S/RVs must be OPERABLE since there may be considerable energy in the reactor core and the limiting design basis transients are assumed to occur. The S/RVs may be required to provide pressure relief to ~~discharge energy from the core until such time that the Residual Heat Removal (RHR) System is capable of dissipating the heat.~~

limit peak reactor pressure. 2

In MODE 4, decay heat is low enough for the RHR System to provide adequate cooling, and reactor pressure is low enough that the overpressure limit is unlikely to be approached by assumed operational transients or accidents. In MODE 5, the reactor vessel head is unbolted or removed and the reactor is at atmospheric pressure. The S/RV function is not needed during these conditions.

ACTIONS

4

~~A.1~~

~~With the safety function of one [required] S/RV inoperable, the remaining OPERABLE S/RVs are capable of providing the necessary overpressure protection. Because of additional design margin, the ASME Code limits for the RCPB can also be satisfied with two S/RVs inoperable. However, the overall reliability of the pressure relief system is reduced because additional failures in the remaining OPERABLE S/RVs could result in failure to adequately relieve pressure during a limiting event. For this reason, continued operation is permitted for a limited time only.~~

~~The 14 day Completion Time to restore the inoperable required S/RVs to OPERABLE status is based on the relief capability of the remaining S/RVs, the low probability of an event requiring S/RV actuation, and a reasonable time to complete the Required Action.~~

(continued)

The S/RVs are required to be OPERABLE to limit peak pressure in the main steam lines and maintain reactor pressure within acceptable limits during events that cause rapid pressurization, so that MCPR is not exceeded. 1

BASES

ACTIONS (continued)

A.1 and A.2 [4]

With less than the minimum number of required S/RVs OPERABLE, a transient may result in the violation of the ASME Code limit on reactor pressure. ~~[If the inoperable required S/RV cannot be restored to OPERABLE status within the associated Completion Time of Required Action A.1] or if~~ [5] [3] one ~~[two]~~ or more [required] S/RVs are inoperable, the plant must be brought to a MODE in which the LCO does not apply. To achieve this status, the plant must be brought to at least MODE 3 within 12 hours and to MODE 4 within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.4.4.1

This Surveillance demonstrates that the [required] S/RVs [3] will open at the pressures assumed in the safety analysis of Reference 2. The demonstration of the S/RV safety function lift settings must be performed during shutdown, since this is a bench test[,] and in accordance with the Inservice Testing Program[.] The lift setting pressure shall correspond to ambient conditions of the valves at nominal operating temperatures and pressures. The S/RV setpoint is ± [3]% for OPERABILITY; however, the valves are reset to ± 1% during the Surveillance to allow for drift. [3]

~~The [18 month] Frequency was selected because this Surveillance must be performed during shutdown conditions and is based on the time between refuelings.~~ [4]

<Insert SR 3.4.4.1> [4]

~~SR 3.4.4.2~~ [4]

~~The [required] relief function S/RVs are required to actuate automatically upon receipt of specific initiation signals. A system functional test is performed to verify the mechanical portions of the automatic relief function operate as designed when initiated either by an actual or simulated initiation signal. The LOGIC SYSTEM FUNCTIONAL TEST in SR 3.3.6.5.4 overlaps this SR to provide complete testing of the safety function.~~

(continued)

[4]

A Note is provided to allow up to two of the required 17 S/RVs for Unit 1, and 12 S/RVs for Unit 2, to be physically replaced with S/RVs with lower setpoints. This provides operational flexibility which maintains the assumptions in the over-pressure protection analysis.

INSERT SR 3.4.4.1

The Frequency is specified in the Inservice Testing Program which requires the valves be subjected to a bench test during refueling outages. The Frequency is acceptable based on industry standards and operating history.

BASES

SURVEILLANCE REQUIREMENTS

SR 3.4.4.2 (continued)

The [18 month] Frequency is based on the need to perform this Surveillance under the conditions that apply during a plant outage and the potential for an unplanned transient if the Surveillance were performed with the reactor at power. Operating experience has shown these components usually pass the SR when performed at the [18 month] Frequency. Therefore, the Frequency was concluded to be acceptable from a reliability standpoint.

4

This SR is modified by a Note that excludes valve actuation. This prevents an RPV pressure blowdown.

SR 3.4.4.3

A manual actuation of each [required] S/RV is performed to verify that, mechanically, the valve is functioning properly and no blockage exists in the valve discharge line. This can be demonstrated by the response of the turbine control valves or bypass valves, by a change in the measured steam flow, or 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. Also, adequate steam flow must be passing through the main turbine or turbine bypass valves to continue to control reactor pressure when the S/RVs divert steam flow upon opening. Sufficient time is therefore allowed after the required pressure and flow are achieved to perform this test. Adequate pressure at which this test is to be performed is 950 psig (the pressure recommended by the valve manufacturer). Adequate steam flow is represented by [at least 1.25 turbine bypass valves open, or total steam flow $\geq 10^6$ lb/hr]. Plant 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. Therefore, this SR is modified by a Note that states the Surveillance is not required to be performed until 12 hours after reactor steam pressure and flow are adequate to perform the test. The 12 hours allowed for manual actuation after the required pressure is reached is sufficient to achieve stable conditions for testing and provides a reasonable time to complete the SR. If the valve fails to actuate due only to the failure of the solenoid but

4

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.4.4.3 (continued) 4

is capable of opening on overpressure, the safety function of the S/RV is considered OPERABLE.

The [18] month on a STAGGERED TEST BASIS Frequency ensures that each solenoid for each S/RV is alternately tested. The 18 month Frequency was developed based on the S/RV tests required by the ASME Boiler and Pressure Vessel Code, Section XI (Ref. 1). Operating experience has shown that these components usually pass the Surveillance when performed at the 18 month Frequency. Therefore, the Frequency was concluded to be acceptable from a reliability standpoint.

REFERENCES

1. ASME, Boiler and Pressure Vessel Code, Section III.

1 2. U FSAR, Section ~~[5.2.5.5.3]~~. 5.2.2.1.3 3

3. U FSAR, ~~Section~~ Chapter 1 [15]. 3

JUSTIFICATION FOR DEVIATIONS FROM NUREG-1434, REVISION 1
ITS BASES: 3.4.4 - SAFETY/RELIEF VALVES (S/RVs)

1. Changes have been made (additions, deletions, and/or changes to the NUREG) to reflect the plant specific nomenclature, number, reference, system description, or analysis description.

2. Editorial change made for enhanced clarity or to be consistent with similar statements in other places in the Bases.

3. The brackets have been removed and the proper plant specific information/value has been provided.

4. Changes have been made to reflect those changes made to the Specification. The following requirements have been renumbered, where applicable, to reflect the changes.

5. The bracketed requirement/information has been deleted because it is not applicable to LaSalle. The following requirements have been renumbered, where applicable, to reflect the changes.

B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.5 RCS Operational LEAKAGE

BASES

BACKGROUND

The RCS includes systems and components that contain or transport the coolant to or from the reactor core. The pressure containing components of the RCS and the portions of connecting systems out to and including the isolation valves define the reactor coolant pressure boundary (RCPB). The joints of the RCPB components are welded or bolted.

During plant life, the joint and valve interfaces can produce varying amounts of reactor coolant LEAKAGE, through either normal operational wear or mechanical deterioration. Limits on RCS operational LEAKAGE are required to ensure appropriate action is taken before the integrity of the RCPB is impaired. This LCO specifies the types and limits of LEAKAGE.

This protects the RCS pressure boundary described in 10 CFR 50.2, 10 CFR 50.55a(c), and GDC 55 of 10 CFR 50, Appendix A (Refs. 1, 2, and 3). 1

The safety significance of leaks from the RCPB varies widely depending on the source, rate, and duration. Therefore, detection of LEAKAGE in the drywell is necessary. Methods for quickly separating the identified LEAKAGE from the unidentified LEAKAGE are necessary to provide the operators quantitative information to permit them to take corrective action should a leak occur detrimental to the safety of the facility or the public.

A limited amount of leakage inside the drywell is expected from auxiliary systems that cannot be made 100% leaktight. Leakage from these systems should be detected and isolated from the drywell atmosphere, if possible, so as not to mask RCS operational LEAKAGE detection.

This LCO deals with protection of the RCPB from degradation and the core from inadequate cooling, in addition to preventing the accident analyses radiation release assumptions from being exceeded. The consequences of violating this LCO include the possibility of a loss of coolant accident.

(continued)

BASES (continued)

APPLICABLE SAFETY ANALYSES

The allowable RCS operational LEAKAGE limits are based on the predicted and experimentally observed behavior of pipe cracks. The normally expected background LEAKAGE due to equipment design and the detection capability of the instrumentation for determining system LEAKAGE were also considered. The evidence from experiments suggests, for LEAKAGE even greater than the specified unidentified LEAKAGE limits, the probability is small that the imperfection or crack associated with such LEAKAGE would grow rapidly.

The unidentified LEAKAGE flow limit allows time for corrective action before the RCPB could be significantly compromised. The 5 gpm limit is a small fraction of the calculated flow from a critical crack in the primary system piping. Crack behavior from experimental programs (Refs. 4 and 5) shows leak rates of hundreds of gallons per minute will precede crack instability (Ref. 6).

The low limit on increase in unidentified LEAKAGE assumes a failure mechanism of intergranular stress corrosion cracking (IGSCC) that produces tight cracks. This flow increase limit is capable of providing an early warning of such deterioration.

No applicable safety analysis assumes the total LEAKAGE limit. The total LEAKAGE limit considers RCS inventory makeup capability and drywell ~~floor~~ sump capacity. 2

RCS operational LEAKAGE satisfies Criterion 2 of ~~the NRC Policy Statement~~ 10 CFR 50.36(c)(2)(ii). 2 2

LCO

RCS operational LEAKAGE shall be limited to:

a. Pressure Boundary LEAKAGE

No pressure boundary LEAKAGE is allowed, being indicative of material degradation. LEAKAGE of this type is unacceptable as the leak itself could cause further deterioration, resulting in higher LEAKAGE. Violation of this LCO could result in continued degradation of the RCPB. LEAKAGE past seals and gaskets is not pressure boundary LEAKAGE.

(continued)

BASES

LCO (continued)

b. Unidentified LEAKAGE

flow

Five gpm of unidentified LEAKAGE is allowed as a reasonable minimum detectable amount that the drywell air monitoring, drywell sump level monitoring, and drywell air cooler condensate flow rate monitoring equipment can detect within a reasonable time period. Violation of this LCO could result in continued degradation of the RCPB. 2

atmospheric

c. Total LEAKAGE

The total LEAKAGE limit is based on a reasonable minimum detectable amount. The limit also accounts for LEAKAGE from known sources (identified LEAKAGE). Violation of this LCO indicates an unexpected amount of LEAKAGE and, therefore, could indicate new or additional degradation in an RCPB component or system.

3

d. Unidentified LEAKAGE Increase

24

An unidentified LEAKAGE increase of > 2 gpm within the previous ~~4~~ hour period indicates a potential flaw in the RCPB and must be quickly evaluated to determine the source and extent of the LEAKAGE. The increase is measured relative to the steady state value; temporary changes in LEAKAGE rate as a result of transient conditions (e.g., startup) are not considered. As such, the 2 gpm increase limit is only applicable in MODE 1 when operating pressures and temperatures are established. Violation of this LCO could result in continued degradation of the RCPB.

APPLICABILITY

In MODES 1, 2, and 3, the RCS operational LEAKAGE LCO applies because the potential for RCPB LEAKAGE is greatest when the reactor is pressurized.

In MODES 4 and 5, RCS operational LEAKAGE limits are not required since the reactor is not pressurized and stresses in the RCPB materials and potential for LEAKAGE are reduced.

(continued)

BASES (continued)

ACTIONS

A.1

With RCS unidentified or total LEAKAGE greater than the limits, actions must be taken to reduce the leak. Because the LEAKAGE limits are conservatively below the LEAKAGE that would constitute a critical crack size, 4 hours is allowed to reduce the LEAKAGE rates before the reactor must be shut down. If an unidentified LEAKAGE has been identified and quantified, it may be reclassified and considered as identified LEAKAGE. However, the total LEAKAGE limit would remain unchanged.

B.1 and B.2

24 3

An unidentified LEAKAGE increase of > 2 gpm within a ~~4~~ hour period is an indication of a potential flaw in the RCPB and must be quickly evaluated. Although the increase does not necessarily violate the absolute unidentified LEAKAGE limit, certain susceptible components must be determined not to be the source of the LEAKAGE increase within the required Completion Time. For an unidentified LEAKAGE increase greater than required limits, an alternative to reducing LEAKAGE increase to within limits (i.e., reducing the leakage rate such that the current rate is less than the "2 gpm increase in the previous ~~4~~ hours" limit; either by isolating the source or other possible methods) is to ~~evaluate RCS type 304 and type 316 austenitic stainless steel piping that is subject to high stress or that contains relatively stagnant or intermittent flow fluids and determine it is not the source of the increased LEAKAGE. This type of piping is very susceptible to IGSCC.~~ 3

4 24

3 identify the source of the unidentified leakage increase is not material susceptible to IGSCC.

The 4 hour Completion Time is needed to properly reduce the LEAKAGE increase or ~~verify~~ identify 3 the source before the reactor must be shut down.

C.1 and C.2

If any Required Action and associated Completion Time of Condition A or B is not met or if pressure boundary LEAKAGE exists, the plant must be brought to a MODE in which the LCO does not apply. To achieve this status, the plant must be brought to MODE 3 within 12 hours and to MODE 4 within 36 hours. The allowed Completion Times are reasonable,

(continued)

BASES

ACTIONS

C.1 and C.2 (continued)

based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.4.5.1

The RCS LEAKAGE is monitored by a variety of instruments designed to provide alarms when LEAKAGE is indicated and to quantify the various types of LEAKAGE. Leakage detection instrumentation is discussed in more detail in the Bases for LCO 3.4.7, "RCS Leakage Detection Instrumentation." Sump level and flow rate are typically monitored to determine actual LEAKAGE rates. However, any method may be used to quantify LEAKAGE ~~within the guidelines of Reference 7~~ provided the method has suitable sensitivity to satisfy the requirements of LCO 3.4.5. [6] In conjunction with alarms and other administrative controls, ~~an~~ a ~~8~~ 12 hour [3] Frequency for this Surveillance is appropriate for identifying changes in LEAKAGE and for tracking required trends (Ref. ~~8~~ 7). [2]

REFERENCES

1. 10 CFR 50.2.
2. 10 CFR 50.55a(c).
3. 10 CFR 50, Appendix A, GDC 55.
4. GEAP-5620, "Failure Behavior in ASTM A106B Pipes Containing Axial Through-Wall Flaws," April 1968. [2]
5. NUREG-~~75~~/0677, "Investigation and Evaluation of Cracking in Austenitic Stainless Steel Piping of Boiling Water Reactor Plants," October 1975. [5]
6. [2] ~~FSAR~~ U, Section ~~[~~5.2.5.5.~~3~~2. [4]
7. ~~Regulatory Guide 1.45.~~ [2]
~~8.~~ 7. Generic Letter 88-01, Supplement 1, February 1992. [1]

JUSTIFICATION FOR DEVIATIONS FROM NUREG-1434, REVISION 1
ITS BASES: 3.4.5 - RCS OPERATIONAL LEAKAGE

1. Editorial change made for enhanced clarity or to be consistent with similar statements in other places in the Bases.

2. Changes have been made (additions, deletions, and/or changes to the NUREG) to reflect the plant specific nomenclature, number, reference, system description, or analysis description.

3. Changes have been made to reflect those changes made to the Specification. The following requirements have been renumbered, where applicable, to reflect the changes.

4. The brackets have been removed and the proper plant specific information/value has been provided.

5. Typographical/grammatical error corrected.

6. The LaSalle design includes LEAKAGE measurement methods with suitable sensitivity to meet the requirements of LCO 3.4.5, but which do not satisfy the requirements of Regulatory Guide 1.45. The Bases description of methods that may be used to quantify RCS leakage in accordance with ITS SR 3.4.5.1 has been expanded to allow credit for these methods consistent with current plant practice.

B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.6 RCS Pressure Isolation Valve (PIV) Leakage

BASES

BACKGROUND

RCS PIVs are defined as any two normally closed valves in series within the reactor coolant pressure boundary (RCPB). 1 The function of RCS PIVs is to separate the high pressure RCS from an attached low pressure system. This protects the RCS pressure boundary described in 10 CFR 50.2, 10 CFR 50.55a(c), and GDC 55 of 10 CFR 50, Appendix A (Refs. 1, 2, and 3). PIVs are designed to meet the requirements of Reference 4. During their lives, these valves can produce varying amounts of reactor coolant leakage through either normal operational wear or mechanical deterioration.

The RCS PIV LCO allows RCS high pressure operation when leakage through these valves exists in amounts that do not compromise safety. The PIV leakage limit applies to each individual valve. Leakage through these valves is not included in any allowable LEAKAGE specified in LCO 3.4.5, "RCS Operational LEAKAGE."

2

Although this specification provides a limit on allowable PIV leakage rate, its main purpose is to prevent overpressure failure of the low pressure portions of connecting systems. The leakage limit is an indication that the PIVs between the RCS and the connecting systems are degraded or degrading. PIV leakage could lead to overpressure of the low pressure piping or components. Failure consequences could be a loss of coolant accident (LOCA) outside of containment, an unanalyzed accident which could degrade the ability for low pressure injection.

A study (Ref. 5) evaluated various PIV configurations to determine the probability of intersystem LOCAs. This study concluded that periodic leakage testing of the PIVs can substantially reduce intersystem LOCA probability.

PIVs are provided to isolate the RCS from the following 3 typically connected systems:

a. Residual Heat Removal (RHR) System;

b. Low Pressure Core Spray System;

(continued)

BASES

BACKGROUND (continued)

c. High Pressure Core Spray System; and

d. Reactor Core Isolation Cooling System.

4

The PIVs are listed in ~~Reference 6~~. the Technical Requirements Manual (Ref. 6)

APPLICABLE SAFETY ANALYSES

Reference 5 evaluated various PIV configurations, leakage testing of the valves, and operational changes to determine the effect on the probability of intersystem LOCAs. This study concluded that periodic leakage testing of the PIVs can substantially reduce the probability of an intersystem LOCA.

4 reactor coolant pressure boundary

PIV leakage is not considered in any Design Basis Accident analyses. This Specification provides for monitoring the condition of the RCPB to detect PIV degradation that has the potential to cause a LOCA outside of containment. 4 P RCS PIV leakage satisfies Criterion 2 of ~~the NRC Policy Statement~~. 10 CFR 50.36(c)(2)(ii) 3

LCO

RCS PIV leakage is leakage into closed systems connected to the RCS. Isolation valve leakage is usually on the order of drops per minute. Leakage that increases significantly suggests that something is operationally wrong and corrective action must be taken. Violation of this LCO could result in continued degradation of a PIV, which could lead to overpressurization of a low pressure system and the loss of the integrity of a fission product barrier.

The LCO PIV leakage limit is 0.5 gpm per nominal inch of valve size with a maximum limit of 5 gpm (Ref. 4).

Reference 7 permits leakage testing at a lower pressure differential than between the specified maximum RCS pressure and the normal pressure of the connected system during RCS operation (the maximum pressure differential). The observed rate may be adjusted to the maximum pressure differential by assuming leakage is directly proportional to the pressure differential to the one-half power.

(continued)

BASES (continued)

APPLICABILITY

In MODES 1, 2, and 3, this LCO applies because the PIV leakage potential is greatest when the RCS is pressurized. In MODE 3, valves in the RHR flowpath are not required to meet the requirements of this LCO when in, or during transition to or from, the RHR shutdown cooling mode of operation.

In MODES 4 and 5, leakage limits are not provided because the lower reactor coolant pressure results in a reduced potential for leakage and for a LOCA outside the containment. Accordingly, the potential for the consequences of reactor coolant leakage is far lower during these MODES.

ACTIONS

The ACTIONS are modified by two Notes. Note 1 has been provided to modify the ACTIONS related to RCS PIV flow paths. Section 1.3, Completion Times, specifies once a Condition has been entered, subsequent divisions, subsystems, components or variables expressed in the Condition, discovered to be inoperable or not within limits, will not result in separate entry into the Condition. Section 1.3 also specifies Required Actions of the Condition continue to apply for each additional failure, with Completion Times based on initial entry into the Condition. 2 However, the Required Actions for the Condition of RCS PIV leakage limits exceeded provide appropriate compensatory 5 ~~measures for separate~~ affected RCS PIV flow paths. As such, a Note has been provided that allows separate Condition entry for each affected RCS PIV flow path. Note 2 requires an evaluation of affected systems if a PIV is inoperable. The leakage may have affected system OPERABILITY, or isolation of a leaking flow path with an alternate valve may have degraded the ability of the interconnected system to perform its safety function. As a result, the applicable Conditions and Required Actions for systems made inoperable by PIVs must be entered. This ensures appropriate remedial actions are taken, if necessary, for the affected systems.

A.1 and A.2

If leakage from one or more RCS PIVs is not within limit, the flow path must be isolated by at least one closed

(continued)

BASES

ACTIONS

A.1 and A.2 (continued) 4

manual, deactivated, automatic, or check valve within 4 hours. Required Action A.1 and Required Action A.2 are modified by a Note stating that the valves used for isolation must meet the same leakage requirements as the PIVs and must be on the RCPB {for the high pressure portion of the system.} 5

Four hours provides time to reduce leakage in excess of the allowable limit and to isolate the flow path if leakage cannot be reduced while corrective actions to reseat the leaking PIVs are taken. The 4 hours allows time for these actions and restricts the time of operation with leaking valves.

Required Action A.2 specifies that the double isolation barrier of two valves be restored by closing another valve qualified for isolation or restoring one leaking PIV. The 72 hour Completion Time after exceeding the limit considers the time required to complete the Required Action, the low probability of a second valve failing during this time period, and the low probability of a pressure boundary rupture of the low pressure ECCS piping when overpressurized to reactor pressure (Ref. 7).

B.1 and B.2

If leakage cannot be reduced or the system isolated, the plant must be brought to a MODE in which the LCO does not apply. To achieve this status, the plant must be brought to MODE 3 within 12 hours and to MODE 4 within 36 hours. This action may reduce the leakage and also reduces the potential for a LOCA outside the containment. The Completion Times are reasonable, based on operating experience, to achieve the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.4.6.1

Performance of leakage testing on each RCS PIV is required to verify that leakage is below the specified limit and to identify each leaking valve. The leakage limit of 0.5 gpm

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.4.6.1 (continued)

per inch of nominal valve diameter up to 5 gpm maximum applies to each valve. Leakage testing requires a stable pressure condition. For the two PIVs tested in series, the leakage requirement applies to each valve individually and not to the combined leakage across both valves ~~. If the PIVs are not individually leakage tested, one valve may have failed completely and not be detected if the other valve in series meets the leakage requirement. In this situation, the protection provided by redundant valves would be lost.~~ (i.e., the leakage acceptance criteria is the criteria for one valve to account for the condition where all of the leakage is through one valve) 4

4

As stated in the LCO section of the Bases, the test pressure may be at a lower pressure than the maximum pressure differential (at the maximum pressure of 1050 psig) provided the observed leakage rate is adjusted in accordance with Reference 4. 3

6 The ~~18 month~~ Frequency required by the Inservice Testing Program is within the ASME Code, Section XI, Frequency requirement and is based on the need to perform this Surveillance under the conditions that apply during an outage and the potential for an unplanned transient if the Surveillance were performed with the reactor at power.

4 ~~Therefore,~~ This SR is modified by a Note that states the leakage Surveillance is ~~not~~ only required to be performed in ~~MODE 3~~ MODES 1 and 2 6. Entry into MODE 3 is permitted for leakage testing at high differential pressures with stable conditions not possible in the lower MODES.

REFERENCES

1. 10 CFR 50.2.
2. 10 CFR 50.55a(c).
3. 10 CFR 50, Appendix A, GDC 55.
4. ASME, Boiler and Pressure Vessel Code, Section XI.
5. NUREG-0677, "The Probability of Intersystem LOCA: Impact due to Leak Testing and Operational Changes," May 1980. 4
6. ~~FSAR, Section [ ]~~. Technical Requirements Manual. 3
7. NEDC-31339, "BWR Owners Group Assessment of Emergency Core Cooling System Pressurization in Boiling Water Reactors," November 1986. 4

JUSTIFICATION FOR DEVIATIONS FROM NUREG-1434, REVISION 1
ITS BASES: 3.4.6 - RCS PRESSURE ISOLATION VALVE (PIV) LEAKAGE

1. The Bases "definition" of RCS PIVs has been deleted since it is not an accurate presentation. Many pairs of in-series, normally closed valves are within the RCPB at LaSalle, but are not classified as RCS PIVs.

2. Typographical/grammatical error corrected.

3. Changes have been made (additions, deletions, and/or changes to the NUREG) to reflect the plant specific nomenclature, number, reference, system description, or analysis description.

4. Editorial change made for enhanced clarity or to be consistent with similar statements in other places in the Bases.

5. The brackets have been removed and the proper plant specific information/value has been provided.

6. Changes have been made to reflect those changes made to the Specification. The following requirements have been renumbered, where applicable, to reflect the changes.

B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.7 RCS Leakage Detection Instrumentation

BASES

BACKGROUND

GDC 30 of 10 CFR 50, Appendix A (Ref. 1), requires means for detecting and, to the extent practical, identifying the location of the source of RCS LEAKAGE. Regulatory Guide 1.45 (Ref. 2) describes acceptable methods for selecting leakage detection systems.

Limits on LEAKAGE from the reactor coolant pressure boundary (RCPB) are required so that appropriate action can be taken before the integrity of the RCPB is impaired (Ref. 2). Leakage detection systems for the RCS are provided to alert the operators when leakage rates above normal background levels are detected and also to supply quantitative measurement of rates. The Bases for LCO 3.4.5, "RCS Operational LEAKAGE," discuss the limits on RCS LEAKAGE rates.

Systems for separating the LEAKAGE of an identified source from an unidentified source are necessary to provide prompt and quantitative information to the operators to permit them to take immediate corrective action.

1

1 flow rate

1 flow

drywell cooler drain flow rate,

1

subsystems

LEAKAGE from the RCPB inside the drywell is detected by at least one of ~~two or~~ three independently monitored variables, such as sump level changes and drywell gaseous and particulate radioactivity levels. The primary means of quantifying LEAKAGE in the drywell is the drywell floor drain sump monitoring system.

The drywell floor drain sump monitoring system monitors the LEAKAGE collected in the floor drain sump. This unidentified LEAKAGE consists of LEAKAGE from control rod drives, valve flanges or packings, floor drains, the Closed Cooling Water System, and drywell air cooling unit condensate drains, and any LEAKAGE not collected in the drywell equipment drain sump. The drywell floor drain sump has ~~transmitters that supply level indications~~ in the main control room.

1

a weir level transmitter that supplies floor drain sump fill-up rate flow indication

The floor drain sump level indicators have switches that start and stop the sump pumps when required. A timer starts each time the sump is pumped down to the low level setpoint.

(continued)

BASES

BACKGROUND (continued)

~~If the sump fills to the high level setpoint before the timer ends, an alarm sounds in the control room, indicating a LEAKAGE rate into the sump in excess of a preset limit. A second timer starts when the sump pumps start on high level. Should this timer run out before the sump level reaches the low level setpoint, an alarm is sounded in the control room indicating a LEAKAGE rate into the sump in excess of a preset limit. A flow indicator in the discharge line of the drywell floor drain sump pumps provides flow indication in the control room.~~ 1

1 INSERT B3.4.7

The drywell air monitoring systems continuously monitor the drywell atmosphere for airborne particulate and gaseous radioactivity. A sudden increase of radioactivity, which may be attributed to RCPB steam or reactor water LEAKAGE, is annunciated in the control room. The drywell atmosphere particulate and gaseous radioactivity monitoring systems are not capable of quantifying leakage rates, but are sensitive enough to indicate increased LEAKAGE rates of 1 gpm within 1 hour. Larger changes in LEAKAGE rates are detected in proportionally shorter times (Ref. 3).

1

[Condensate from four of the six drywell coolers is routed to the drywell floor drain sump and is monitored by a flow transmitter that provides indication and alarms in the control room. This drywell air cooler condensate flow rate monitoring system serves as an added indicator, but not quantifier, of RCS unidentified LEAKAGE.] 2

APPLICABLE SAFETY ANALYSES

A threat of significant compromise to the RCPB exists if the barrier contains a crack that is large enough to propagate rapidly. LEAKAGE rate limits are set low enough to detect the LEAKAGE emitted from a single crack in the RCPB (Refs. 4 and 5). Each of the leakage detection systems inside the drywell is designed with the capability of detecting LEAKAGE less than the established LEAKAGE rate limits and providing appropriate alarm of excess LEAKAGE in the control room.

A control room alarm allows the operators to evaluate the significance of the indicated LEAKAGE and, if necessary, shut down the reactor for further investigation and corrective action. The allowed LEAKAGE rates are well below the rates predicted for critical crack sizes (Ref. 6).

(continued)

Insert B 3.4.7

The floor drain sump has level switches that start and stop the sump pumps when required. The sump pump which is selected Lead starts on high level in the sump. The other pump starts, and a control room alarm is annunciated, if the sump level reaches the high-high level. The pumps stop when low level is reached in the sump. A timer starts each time the first sump pump starts. A second timer starts when the pump is stopped. If the pump takes longer than a given time to pump down the sump, or if the pump starts too soon after the previous pumpdown, an alarm is sounded in the control room indicating a higher than normal sump fill up rate. A flow monitor in the discharge line of the drywell floor drain sump pumps provides a flow input to a flow totalizer which is indicated in the control room. This flow totalizer indication can be used to quantify the amount of sump inputs.

BASES

APPLICABLE SAFETY ANALYSES (continued)

Therefore, these actions provide adequate response before a significant break in the RCPB can occur.

RCS leakage detection instrumentation satisfies Criterion 1 of ~~the NRC Policy Statement~~ 10 CFR 50.36(c)(2)(ii). 1

LCO

The drywell floor drain sump flow monitoring system is required to quantify the unidentified LEAKAGE from the RCS. Thus, for the system to be considered OPERABLE, ~~either the flow monitoring or the sump level monitoring portion~~ the floor drain sump fillup rate monitor portion of the system must be OPERABLE. The other monitoring systems provide early alarms to the operators so closer examination of other detection systems will be made to determine the extent of any corrective action that may be required. With the leakage detection systems inoperable, monitoring for LEAKAGE in the RCPB is degraded. 1 1

APPLICABILITY

In MODES 1, 2, and 3, leakage detection systems are required to be OPERABLE to support LCO 3.4.5. This Applicability is consistent with that for LCO 3.4.5.

ACTIONS

A.1

With the drywell floor drain sump flow monitoring system inoperable, no other form of sampling can provide the equivalent information to quantify leakage. However, the drywell atmospheric activity monitor [and the drywell air cooler condensate flow rate monitor] will provide indications of changes in leakage. 1 2

With the drywell floor drain sump flow monitoring system inoperable, but with RCS unidentified and total LEAKAGE being determined every ~~8~~ 12 hours (SR 3.4.5.1), operation may continue for 30 days. The 30 day Completion Time of Required Action A.1 is acceptable, based on operating experience, considering the multiple forms of leakage detection that are still available. Required Action A.1 is modified by a Note that states that the provisions of LCO 3.0.4 are not applicable. As a result, a MODE change is allowed when the drywell floor drain sump flow monitoring system 1 4 1

(continued)

BASES

ACTIONS

A.1 (continued)

is inoperable. This allowance is provided because other instrumentation is available to monitor RCS leakage.

B.1 ~~and B.2~~ 4

(i.e. the required drywell atmospheric monitoring system) 3

With both gaseous and particulate drywell atmospheric monitoring channels inoperable~~,~~ grab samples of the drywell atmosphere shall be taken and analyzed to provide periodic leakage information. ~~[Provided a sample is obtained and analyzed every 12 hours, the plant may be operated for up to 30 days to allow restoration of at least one of the required monitors.]~~ 5 [Provided a sample is obtained and analyzed every 12 hours, the plant may continue operation since at least one other form of drywell leakage detection (i.e., air cooler condensate flow rate monitor) is available.] 2

The 12 hour interval provides periodic information that is adequate to detect LEAKAGE. 1 ~~The 30 day Completion Time for restoration recognizes that at least one other form of leakage detection is available.~~ 4

~~The Required Actions are modified by a Note that states that the provisions of LCO 3.0.4 are not applicable. As a result, a MODE change is allowed when both the gaseous and particulate primary containment atmospheric monitoring channels are inoperable. This allowance is provided because other instrumentation is available to monitor RCS leakage.~~ 4

[C.1

With the required drywell air cooler condensate flow rate monitoring system inoperable, SR 3.4.7.1 is performed every 8 hours to provide periodic information of activity in the drywell at a more frequent interval than the routine Frequency of SR 3.4.7.1. The 8 hour interval provides periodic information that is adequate to detect LEAKAGE and recognizes that other forms of leakage detection are available. However, this Required Action is modified by a Note that allows this action to be not applicable if the required drywell atmospheric monitoring system is inoperable. Consistent with SR 3.0.1, Surveillances are not required to be performed on inoperable equipment.] 2

(continued)

BASES

ACTIONS (continued)

D.1 and D.2

With both the gaseous and particulate drywell atmospheric monitor channels and the drywell air cooler condensate flow rate monitor inoperable, the only means of detecting LEAKAGE is the drywell floor drain sump flow monitor. This Condition 3 does not provide the required diverse means of leakage detection. The Required Action is to restore either of the inoperable monitors to OPERABLE status within 30 days to regain the intended leakage detection diversity. The 30 day Completion Time ensures that the plant will not be operated in a degraded configuration for a lengthy time period. The Required Actions are modified by a Note that states that the provisions of LCO 3.0.4 are not applicable. As a result, a MODE change is allowed when both the gaseous and particulate primary containment atmospheric monitoring channels and air cooler condensate flow rate are inoperable. This allowance is provided because other instrumentation is available to monitor RCS leakage. 1 2

E.1 and E.2

If any Required Action of Condition A, B, C, or D cannot be met within the associated Completion Time, the plant must be brought to a MODE in which the LCO does not apply. To achieve this status, the plant must be brought to at least MODE 3 within 12 hours and to MODE 4 within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions in an orderly manner and without challenging plant systems. 2

F.1

With all required monitors inoperable, no required automatic means of monitoring LEAKAGE are available, and immediate plant shutdown in accordance with LCO 3.0.3 is required.

SURVEILLANCE REQUIREMENTS

SR 3.4.7.1 INSERT SR 4

This SR requires the performance of a CHANNEL CHECK of the required drywell atmospheric monitoring system. The check gives reasonable confidence that the channel is operating

(continued)

INSERT SR

The Surveillances are modified by a Note to indicate that when a channel is placed in an inoperable status solely for performance of required Surveillances, entry into associated Conditions and Required Actions may be delayed for up to 6 hours, provided the other required instrumentation (the drywell sump flow monitoring system, drywell atmospheric monitoring channel, or the drywell air cooler condensate flow monitoring system, as applicable) is OPERABLE. Upon completion of the Surveillance, or expiration of the 6 hour allowance, the channel must be returned to OPERABLE status or the applicable Condition entered and Required Actions taken. The 6 hour testing allowance is acceptable since it does not significantly reduce the probability of properly monitoring RCS leakage.

BASES

SURVEILLANCE REQUIREMENTS

SR 3.4.7.1 (continued)

properly. The Frequency of 12 hours is based on instrument reliability and is reasonable for detecting off normal conditions.

SR 3.4.7.2

This SR requires the performance of a CHANNEL FUNCTIONAL TEST of the required RCS leakage detection instrumentation. The test ensures that the monitors can perform their function in the desired manner. The test also verifies the alarm setpoint and relative accuracy of the instrument string. The Frequency of 31 days considers instrument reliability, and operating experience has shown it proper for detecting degradation.

1 function

SR 3.4.7.3

This SR requires the performance of a CHANNEL CALIBRATION of the required RCS leakage detection instrumentation channels. The calibration verifies the accuracy of the instrument string, including the instruments located inside the drywell. The Frequency of [18] months is a typical refueling cycle and considers channel reliability. Operating experience has proven this Frequency is acceptable.

24 2

REFERENCES

1. 10 CFR 50, Appendix A, GDC 30.
2. Regulatory Guide 1.45, May 1973.
3. FSAR, Section [5.2.5.2]. 1.1 2
4. GEAP-5620, April 1968. 1 "Failure Behavior in ASTM A106B Pipes Containing Axial Through-Wall Flaws,"
5. NUREG-75/067, October 1975.
6. FSAR, Section [5.2.5.5.2]. 2 2

4 1

"Investigation and Evaluation of Cracking in Austenitic Stainless Steel Piping of Boiling Water Reactor Plants," 1

JUSTIFICATION FOR DEVIATIONS FROM NUREG-1434, REVISION 1
ITS BASES: 3.4.7 - RCS LEAKAGE DETECTION INSTRUMENTATION

1. Changes have been made (additions, deletions, and/or changes to the NUREG) to reflect the plant specific nomenclature, number, reference, system description, or analysis description.

2. The brackets have been removed and the proper plant specific information/value has been provided.

3. Editorial change made for enhanced clarity or to be consistent with similar statements in other places in the Bases.

4. Changes have been made to reflect those changes made to the Specification. The following requirements have been renumbered, where applicable, to reflect the changes.

5. The bracketed requirement/information has been deleted because it is not applicable to LaSalle Units 1 and 2.

B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.8 RCS Specific Activity

BASES

BACKGROUND

During circulation, the reactor coolant acquires radioactive materials due to release of fission products from fuel leaks into the coolant and activation of corrosion products in the reactor coolant. These radioactive materials in the coolant can plate out in the RCS, and, at times, an accumulation will break away to spike the normal level of radioactivity. The release of coolant during a Design Basis Accident (DBA) could send radioactive materials into the environment.

Limits on the maximum allowable level of radioactivity in the reactor coolant are established to ensure, in the event of a release of any radioactive material to the environment during a DBA, radiation doses are maintained within the limits of 10 CFR 100 (Ref. 1).

This LCO contains iodine specific activity limits. The iodine isotopic activities per gram of reactor coolant are expressed in terms of a DOSE EQUIVALENT I-131. The allowable levels are intended to limit the 2 hour radiation dose to an individual at the site boundary to a small fraction of the 10 CFR 100 limit.

APPLICABLE SAFETY ANALYSES

Analytical methods and assumptions involving radioactive material in the primary coolant are presented in the U FSAR (Ref. 2). The specific activity in the reactor coolant (the source term) is an initial condition for evaluation of the consequences of an accident due to a main steam line break (MSLB) outside containment. No fuel damage is postulated in the MSLB accident, and the release of radioactive material to the environment is assumed to end when the main steam isolation valves (MSIVs) close completely. 1

This MSLB release forms the basis for determining offsite doses (Ref. 2). The limits on the specific activity of the primary coolant ensure that the 2 hour thyroid and whole body doses at the site boundary, resulting from an MSLB

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

outside containment during steady state operation, will not exceed 10% of the dose guidelines of 10 CFR 100.

The ~~limits~~ on specific activity ~~are~~ is a ~~values~~ value from ~~a parametric~~ evaluation of typical site locations. ~~These~~ This ~~limits~~ limit ~~are~~ is conservative because the evaluation considered more restrictive parameters than for a specific site, such as the location of the site boundary and the meteorological conditions of the site. 2

RCS specific activity satisfies Criterion 2 of ~~the NRC Policy Statement~~ 10 CFR 50.36 (c)(2)(ii). 1

LCO

The specific iodine activity is limited to ≤ [0.2] μCi/gm DOSE EQUIVALENT I-131. This limit ensures the source term assumed in the safety analysis for the MSLB is not exceeded, so any release of radioactivity to the environment during an MSLB is less than a small fraction of the 10 CFR 100 limits. 3

APPLICABILITY

In MODE 1, and MODES 2 and 3 with any main steam line not isolated, limits on the primary coolant radioactivity are applicable since there is an escape path for release of radioactive material from the primary coolant to the environment in the event of an MSLB outside of primary containment.

In MODES 2 and 3 with the main steam lines isolated, such limits do not apply since an escape path does not exist. In MODES 4 and 5, no limits are required since the reactor is not pressurized and the potential for leakage is reduced.

ACTIONS

A note to the Required Action of Condition A excludes the MODE change restriction of LCO 3.0.4. This exception allows entry into the applicable MODE(S) while relying on the ACTIONS even though the ACTIONS may eventually require plant shutdown. This exception is acceptable due to the significant conservatism incorporated into the specific activity limit, the low probability of an event which is limiting due to exceeding this limit, and the ability to 4

Move to Insert A.1 and A.2, next page TSTF-137

(continued)

BASES

ACTIONS (continued)

restore transient specific activity excursions while the plant remains at, or proceeds to power operation.

move to Insert A.1 and A.2 this page

TSTF-137

A.1 and A.2

When the reactor coolant specific activity exceeds the LCO DOSE EQUIVALENT I-131 limit, but is ≤ 4.0 μCi/gm, samples must be analyzed for DOSE EQUIVALENT I-131 at least once every 4 hours. In addition, the specific activity must be restored to the LCO limit within 48 hours. The Completion Time of once every 4 hours is based on the time needed to take and analyze a sample. The 48 hour Completion Time to restore the activity level provides a reasonable time for temporary coolant activity increases (iodine spikes or crud bursts) to be cleaned up with the normal processing systems.

<Insert A.1 and A.2>

B.1, B.2.1, B.2.2.1, and B.2.2.2

If the DOSE EQUIVALENT I-131 cannot be restored to ≤ [0.2] μCi/gm within 48 hours, or if at any time it is > [4.0] μCi/gm, it must be determined at least every 4 hours and all the main steam lines must be isolated within 12 hours. Isolating the main steam lines precludes the possibility of releasing radioactive material to the environment in an amount that is more than a small fraction of the requirements of 10 CFR 100 during a postulated MSLB accident. 3

Alternately, the plant can be brought to MODE 3 within 12 hours and to MODE 4 within 36 hours. This option is provided for those instances when isolation of main steam lines is not desired (e.g., due to the decay heat loads). In MODE 4, the requirements of the LCO are no longer applicable.

The Completion Time of once every 4 hours is the time needed to take and analyze a sample. The 12 hour Completion Time is reasonable, based on operating experience, to isolate the main steam lines in an orderly manner and without challenging plant systems. Also, the allowed Completion Times for Required Actions B.2.2.1 and B.2.2.2 for bringing the plant to MODES 3 and 4 are reasonable, based on

(continued)

BASES

ACTIONS

B.1, B.2.1, B.2.2.1, and B.2.2.2 (continued)

operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.4.8.1

This Surveillance is performed to ensure iodine remains within limit during normal operation. The 7 day Frequency is adequate to trend changes in the iodine activity level.

This SR is modified by a Note that requires this Surveillance to be performed only in MODE 1 because the level of fissION products generated in other MODES is much less.

REFERENCES

1. 10 CFR 100.11, ~~1973~~. 4

1 2. u FSAR, Section ~~15.1.40~~. 15.6.4.5 3

JUSTIFICATION FOR DEVIATIONS FROM NUREG-1434, REVISION 1
ITS BASES: 3.4.8 - RCS SPECIFIC ACTIVITY

1. Changes have been made (additions, deletions, and/or changes to the NUREG) to reflect the plant specific nomenclature, number, reference, system description, or analysis description.

2. Changes have been made to more closely match the LCO requirements.

3. The brackets have been removed and the proper plant specific information/value has been provided.

4. Editorial change made for enhanced clarity or to be consistent with similar statements in other places in the Bases.

B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.9 Residual Heat Removal (RHR) Shutdown Cooling System—Hot Shutdown

BASES

BACKGROUND

Irradiated fuel in the shutdown reactor core generates heat during the decay of fission products and increases the temperature of the reactor coolant. This decay heat must be removed to reduce the temperature of the reactor coolant to ≤ 200°F. ~~This decay heat removal is~~ in preparation for performing refueling or maintenance operations, or ~~for~~ ~~keeping~~ the reactor in the Hot Shutdown condition. 1

the decay heat must be removed for

1 maintaining

Cold Shutdown

The two redundant, manually controlled shutdown cooling subsystems of the RHR System provide decay heat removal. Each loop consists of a motor driven pump, two heat exchangers ~~in series~~, and associated piping and valves. Both loops have a common suction from the same recirculation loop. Each pump discharges the reactor coolant, after circulation through the respective heat exchanger, to the reactor via ~~separate feedwater lines or to the reactor via the LPCI injection path.~~ The RHR heat exchangers transfer heat to the Standby Service Water System (LCO 3.7.1, "[Standby] Service Water ([SSW)] System ~~and [Ultimate Heat Sink (UHS)]~~"). a 2

2

2 the associated recirculation loop.

3

Residual Heat Removal

(RHRSW)

APPLICABLE SAFETY ANALYSES

Decay heat removal by the RHR System in the shutdown cooling mode is not required for mitigation of any event or accident evaluated in the safety analyses. Decay heat removal is, however, an important safety function that must be accomplished or core damage could result. Although the RHR Shutdown Cooling System does not meet a specific criterion ~~of the NRC Policy Statement, it was identified in the NRC Policy Statement as a significant contributor to risk reduction. Therefore, the RHR Shutdown Cooling System is retained as a Technical Specification.~~ P 1

4 of 10 CFR 50.36 (c)(2)(ii)

4

LCO

Two RHR shutdown cooling subsystems are required to be OPERABLE, and, when no recirculation pump is in operation, one shutdown cooling subsystem must be in operation. An OPERABLE RHR shutdown cooling subsystem consists of one OPERABLE RHR pump, two heat exchangers ~~in series~~, and the

2

One

(continued)

BASES

LCO (continued)

associated piping and valves. 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. In MODE 3, one RHR shutdown cooling subsystem can provide the required cooling, but two subsystems are required to be OPERABLE to provide redundancy. Operation of one subsystem can maintain or reduce the reactor coolant temperature as required.
1 However, to ensure adequate core flow to allow for accurate average reactor coolant temperature monitoring, nearly continuous operation is required.

not be in operation TSTF-153

Note 1 permits both RHR shutdown cooling subsystems and recirculation pumps to ~~be shut down~~ for a period of 2 hours in an 8 hour period. Note 2 allows one RHR shutdown cooling subsystem to be inoperable for up to 2 hours for performance of surveillance tests. These tests may be on the affected RHR System or on some other plant system or component that necessitates placing the RHR System in an inoperable status during the performance. This is permitted because the core heat generation can be low enough and the heatup rate slow enough to allow some changes to the RHR subsystems or other operations requiring RHR flow interruption and loss of redundancy.

APPLICABILITY

vessel 5

In MODE 3 with reactor ~~steam dome~~ pressure below the RHR cut-in 1 permissive pressure (i.e., the actual pressure at which the interlock resets) the RHR Shutdown Cooling System ~~may~~ be operated in the shutdown cooling mode to remove decay heat to reduce or maintain coolant temperature. Otherwise, a recirculation pump is required to be in operation.

5 must be OPERABLE and one RHR shutdown cooling subsystem shall

5 vessel

In MODES 1 and 2, and in MODE 3 with reactor ~~steam dome~~ pressure greater than or equal to the RHR cut-in permissive 1 pressure, this LCO is not applicable. Operation of the RHR System in the shutdown cooling mode is not allowed above this pressure because the RCS pressure may exceed the design pressure of the shutdown cooling piping. Decay heat removal at reactor pressures greater than or equal to the RHR cut-in 1 permissive pressure is typically accomplished by condensing the steam in the main condenser. Additionally, in MODE 2
5 below this pressure, the OPERABILITY requirements for the 5 Emergency Core Cooling Systems (ECCS) (LCO 3.5.1,

With an RHR shutdown cooling subsystem not in operation

(continued)

BASES

APPLICABILITY (continued)

"ECCS—Operating") do not allow placing the RHR shutdown cooling subsystem into operation.

The requirements for decay heat removal in MODES 4 and 5 are discussed in LCO 3.4.10, "Residual Heat Removal (RHR) Shutdown Cooling System—Cold Shutdown"; LCO 3.9.8, "Residual Heat Removal (RHR)—High Water Level"; and LCO 3.9.9, "Residual Heat Removal (RHR)—Low Water Level."

ACTIONS

5

A Note to the ACTIONS excludes the MODE change restriction of LCO 3.0.4. This exception allows entry into the applicable MODE(S) while relying on the ACTIONS even though the ACTIONS may eventually require plant shutdown. This exception is acceptable due to the redundancy of the OPERABLE subsystems, the low pressure at which the plant is operating, the low probability of an event occurring during operation in this condition, and the availability of alternate methods of decay heat removal capability.

A second Note has been provided to modify the ACTIONS related to RHR shutdown cooling subsystems. Section 1.3, Completion Times, specifies once a Condition has been entered, subsequent divisions, subsystems, components or variables expressed in the Condition, discovered to be inoperable or not within limits, will not result in separate entry into the Condition. Section 1.3 also specifies Required Actions of the Condition continue to apply for each additional failure, with Completion Times based on initial entry into the Condition. However, the Required Actions for inoperable shutdown cooling subsystems provide appropriate compensatory measures for separate inoperable shutdown cooling subsystems. As such, a Note has been provided that allows separate Condition entry for each inoperable RHR shutdown cooling subsystem.

<u>A.1, A.2, and A.3</u> 5

With one required RHR shutdown cooling subsystem inoperable for decay heat removal, except as permitted by LCO Note 2, the inoperable subsystem must be restored to OPERABLE status without delay. In this condition, the remaining OPERABLE subsystem can provide the necessary decay heat removal. The overall reliability is reduced, however, because a single

(continued)

BASES

ACTIONS

A.1, A.2, and A.3 (continued)

failure in the OPERABLE subsystem could result in reduced RHR shutdown cooling capability. Therefore an alternate method of decay heat removal must be provided.

With both RHR shutdown cooling subsystems inoperable, an alternate method of decay heat removal must be provided in addition to that provided for the initial RHR shutdown cooling subsystem inoperability. This re-establishes backup decay heat removal capabilities, similar to the requirements of the LCO. The 1 hour Completion Time is based on the decay heat removal function and the probability of a loss of the available decay heat removal capabilities.

Condensate/Feed and Main Steam Systems 2

The required cooling capacity of the alternate method should be ensured by verifying (by calculation or demonstration) its capability to maintain or reduce temperature. Decay heat removal by ambient losses can be considered as, or contributing to, the alternate method capability. Alternate methods that can be used include (but are not limited to) the Spent Fuel Pool Cooling System or the Reactor Water Cleanup System.

(by itself or using feed and bleed in combination with the Control Rod Drive System or Condensate/Feed System), and a combination of an ECCS pump and S/RVs 2

However, due to the potentially reduced reliability of the alternate methods of decay heat removal, it is also required to reduce the reactor coolant temperature to the point where MODE 4 is entered.

B.1, B.2, and B.3

With no RHR shutdown cooling subsystem and no recirculation pump in operation, except as is permitted by LCO Note 1, reactor coolant circulation by the RHR shutdown cooling subsystem or one recirculation pump must be restored without delay.

Until RHR or recirculation pump operation is re-established, an alternate method of reactor coolant circulation must be placed into service. This will provide the necessary circulation for monitoring coolant temperature. The 1 hour Completion Time is based on the coolant circulation function and is modified such that the 1 hour is applicable

(continued)

BASES

ACTIONS

B.1, B.2, and B.3 (continued)

separately for each occurrence involving a loss of coolant circulation. Furthermore, verification of the functioning of the alternate method must be reconfirmed every 12 hours thereafter. This will provide assurance of continued temperature monitoring capability.

During the period when the reactor coolant is being circulated by an alternate method (other than by the required RHR shutdown cooling subsystem or recirculation pump), the reactor coolant temperature and pressure must be periodically monitored to ensure proper function of the alternate method. The once per hour Completion Time is deemed appropriate.

SURVEILLANCE REQUIREMENTS

SR 3.4.9.1

This Surveillance verifies that one RHR shutdown cooling subsystem or recirculation pump is in operation and circulating reactor coolant. The required flow rate is determined by the flow rate necessary to provide sufficient decay heat removal capability. The Frequency of 12 hours is sufficient in view of other visual and audible indications available to the operator for monitoring the RHR subsystem in the control room.

This Surveillance is modified by a Note allowing sufficient time to align the RHR System for shutdown cooling operation after clearing the pressure interlock that isolates the system, or for placing a recirculation pump in operation. The Note takes exception to the requirements of the Surveillance being met (i.e., forced coolant circulation is not required for this initial 2 hour period), which also allows entry into the Applicability of this Specification in accordance with SR 3.0.4 since the Surveillance will not be "not met" at the time of entry into the Applicability.

REFERENCES

None.

1. Editorial change made for enhanced clarity or to be consistent with similar statements in other places in the Bases.

2. Changes have been made (additions, deletions, and/or changes to the NUREG) to reflect the plant specific nomenclature, number, reference, system description, or analysis description.

3. The brackets have been removed and the proper plant specific information/value has been provided.

4. The proper 10 CFR 50.36(c)(2)(ii) criterion has been used. The current wording was developed prior to the issuance of the change to 10 CFR 50.36, which uses Criterion 4 for the current words of the NUREG.

5. Changes have been made to more closely match the LCO requirements.

B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.10 Residual Heat Removal (RHR) Shutdown Cooling System—Cold Shutdown

BASES

BACKGROUND

Irradiated fuel in the shutdown reactor core generates heat during the decay of fission products and increases the temperature of the reactor coolant. This decay heat must be removed to maintain the temperature of the reactor coolant at ≤ 200°F. ~~This decay heat removal is~~ in preparation for performing refueling ~~or~~ maintenance operations, or ~~for~~ ~~keeping~~ the reactor in the Cold Shutdown condition.

1 the decay heat must be removed for

1 maintaining

The two redundant, manually controlled shutdown cooling subsystems of the RHR System provide decay heat removal. Each loop consists of a motor driven pump, ~~two~~ heat exchanger~~s in series~~, and associated piping and valves. Both loops have a common suction from the same recirculation loop. Each pump discharges the reactor coolant, after circulation through the respective heat exchanger, to the reactor via ~~separate feedwater lines or to the reactor via the LPCI injection path.~~ The RHR heat exchangers transfer heat to the ~~Standby~~ Service Water System.

a 2

2

2 the associated recirculation loop.

Residual Heat Removal (RHRSW) 2

APPLICABLE SAFETY ANALYSES

Decay heat removal by the RHR System in the shutdown cooling mode is not required for mitigation of any event or accident evaluated in the safety analyses. Decay heat removal is, however, an important safety function that must be accomplished or core damage could result. ~~Although~~ the RHR Shutdown Cooling System ~~does not~~ meets ~~a specific~~ criterion ~~of the NRC Policy Statement, it was identified in the NRC Policy Statement as a significant contributor to risk reduction. Therefore, the RHR Shutdown Cooling System is retained as a Technical Specification.~~

P 3

4 of 10 CFR 50.36 (c)(2)(ii)

LCO

Two RHR shutdown cooling subsystems are required to be OPERABLE, and, when no recirculation pump is in operation, one RHR shutdown cooling subsystem must be in operation. An OPERABLE RHR shutdown cooling subsystem consists of one OPERABLE RHR pump, ~~two~~ heat exchanger~~s in series~~, and the associated piping and valves. Each shutdown cooling

2 one 2

(continued)

2 the necessary portions of the RHRSW system and Ultimate Heat Sink capable of providing cooling to the heat exchanger,

BASES

LCO (continued)

subsystem is considered OPERABLE if it can be manually aligned (remote or local) in the shutdown cooling mode for removal of decay heat. In MODE 4, one RHR shutdown cooling subsystem can provide the required cooling, but two subsystems are required to be OPERABLE to provide redundancy. Operation of one subsystem can maintain and reduce the reactor coolant temperature as required. ~~However, t~~ To ensure adequate core flow to allow for accurate average reactor coolant temperature monitoring, nearly continuous operation is required. 1

INSERT Note 1

Note ~~1~~ 2 permits both RHR shutdown cooling subsystems and recirculation pumps to ~~be shut down~~ not be in operation for a period of 2 hours in an 8 hour period. Note ~~2~~ 3 allows one RHR shutdown cooling subsystem to be inoperable for up to 2 hours for performance of surveillance tests. These tests may be on the affected RHR System or on some other plant system or component that necessitates placing the RHR System in an inoperable status during the performance. This is permitted because the core heat generation can be low enough and the heatup rate slow enough to allow some changes to the RHR subsystems or other operations requiring RHR flow interruption and loss of redundancy. 5 TSTF-153

APPLICABILITY

In MODE 4, the RHR Shutdown Cooling 2 System ~~may~~ must be OPERABLE and one RHR shutdown cooling subsystem shall be operated in the shutdown cooling mode to remove decay heat to maintain coolant temperature below 200°F. ~~Otherwise~~ With an RHR shutdown cooling subsystem not in operation, a recirculation pump is required to be in operation. 4

In MODES 1 and 2, and in MODE 3 with reactor ~~steam dome~~ vessel 2 pressure greater than or equal to the RHR cut~~-~~in permissive pressure, this LCO is not applicable. Operation of the RHR System in the shutdown cooling mode is not allowed above this pressure because the RCS pressure may exceed the design pressure of the shutdown cooling piping. Decay heat removal at reactor pressures greater than or equal to the RHR cut~~-~~in permissive pressure is typically accomplished by condensing the steam in the main condenser. Additionally, in MODE 2 below this pressure, the OPERABILITY requirements for the Emergency Core Cooling Systems (ECCS) (LCO 3.5.1, "ECCS—Operating") do not allow placing the RHR shutdown cooling subsystem into operation. 2

(continued)

Insert Note 1

Note 1 allows both RHR shutdown cooling subsystems to be inoperable during hydrostatic testing. This is necessary since the RHR Shutdown Cooling System is not designed to operate at the Reactor Coolant System pressures achieved during hydrostatic testing. This is acceptable since adequate reactor coolant circulation will be achieved by operation of a reactor recirculation pump and since systems are available to control reactor coolant temperature.

BASES

APPLICABILITY (continued)

2

The requirements for decay heat removal in MODE 3 below the cut in permissive pressure and in MODE 5 are discussed in LCO 3.4.9, "Residual Heat Removal (RHR) Shutdown Cooling System—Hot Shutdown"; LCO 3.9.8, "Residual Heat Removal (RHR)—High Water Level"; and LCO 3.9.9, "Residual Heat Removal (RHR)—Low Water Level."

ACTIONS

A Note has been provided to modify the ACTIONS related to RHR shutdown cooling subsystems. Section 1.3, Completion Times, specifies once a Condition has been entered, subsequent divisions, subsystems, components or variables expressed in the Condition, discovered to be inoperable or not within limits, will not result in separate entry into the Condition. Section 1.3 also specifies Required Actions of the Condition continue to apply for each additional failure, with Completion Times based on initial entry into the Condition. However, the Required Actions for inoperable shutdown cooling subsystems provided appropriate compensatory measures for separate inoperable shutdown cooling subsystems. As such, a Note has been provided that allows separate Condition entry for each inoperable RHR shutdown cooling subsystem.

A.1 4 Notes 1 and 3 5

With one of the two ~~required~~ RHR shutdown cooling subsystems inoperable except as permitted by LCO ~~Note 2~~, the remaining subsystem is capable of providing the required decay heat removal. However, the overall reliability is reduced. Therefore, an alternate method of decay heat removal must be provided. With both RHR shutdown cooling subsystems inoperable, an alternate method of decay heat removal must be provided in addition to that provided for the initial RHR shutdown cooling subsystem inoperability. This re-establishes backup decay heat removal capabilities, similar to the requirements of the LCO. The 1 hour Completion Time is based on the decay heat removal function and the probability of a loss of the available decay heat removal capabilities. Furthermore, verification of the functional availability of these alternate method(s) must be reconfirmed every 24 hours thereafter. This will provide assurance of continued heat removal capability.

(continued)

BASES

ACTIONS

A.1 (continued)

The required cooling capacity of the alternate method should be ensured by verifying (by calculation or demonstration) its capability to maintain or reduce temperature. Decay heat removal by ambient losses can be considered as, or contributing to, the alternate method capability. Alternate methods that can be used include (but are not limited to) the ~~Spent Fuel Pool Cooling System or~~ the Reactor Water Cleanup System.

1

Condensate/Feed and Main Steam Systems,

2

(by itself or using Feed and bleed in combination with the Control Rod Drive System or Condensate/Feed System) and a combination of an ECCS pump and S/RVs

B.1 and B.2

Notes 1 and 2

5

With no RHR shutdown cooling subsystem and no recirculation pump in operation, except as is permitted by LCO ~~Note 1~~, and until RHR or recirculation pump operation is re-established, an alternate method of reactor coolant circulation must be placed into service. This will provide the necessary circulation for monitoring coolant temperature. The 1 hour Completion Time is based on the coolant circulation function and is modified such that the 1 hour is applicable separately for each occurrence involving a loss of coolant circulation. Furthermore, verification of the functioning of the alternate method must be reconfirmed every 12 hours thereafter. This will provide assurance of continued temperature monitoring capability.

1

subsystem

During the period when the reactor coolant is being circulated by an alternate method (other than by the required RHR shutdown cooling system or recirculation pump), the reactor coolant temperature and pressure must be periodically monitored to ensure proper function of the alternate method. The once per hour Completion Time is deemed appropriate.

SURVEILLANCE REQUIREMENTS

SR 3.4.10.1

This Surveillance verifies that one RHR shutdown cooling subsystem or recirculation pump is in operation and circulating reactor coolant. The required flow rate is determined by the flow rate necessary to provide sufficient decay heat removal capability. The Frequency of 12 hours is

(continued)

BASES

SURVEILLANCE REQUIREMENTS	SR 3.4.10.1 (continued) sufficient in view of other visual and audible indications available to the operator for monitoring the RHR subsystem in the control room.
REFERENCES	None.

1. Editorial change made for enhanced clarity or to be consistent with similar statements in other places in the Bases.

2. Changes have been made (additions, deletions, and/or changes to the NUREG) to reflect the plant specific nomenclature, number, reference, system description, or analysis description.

3. The proper 10 CFR 50.36(c)(2)(ii) criterion has been used. The current wording was developed prior to the issuance of the change to 10 CFR 50.36, which uses Criterion 4 for the current words of the NUREG.

4. Changes have been made to more closely match the LCO requirements.

5. Changes have been made to reflect those changes made to the Specification. The following requirements have been renumbered, where applicable, to reflect the changes.

B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.11 RCS Pressure and Temperature (P/T) Limits

BASES

BACKGROUND

All components of the RCS are designed to withstand effects of cyclic loads due to system pressure and temperature changes. These loads are introduced by startup (heatup) and shutdown (cooldown) operations, power transients, and reactor trips. This LCO limits the pressure and temperature changes during RCS heatup and cooldown, within the design assumptions and the stress limits for cyclic operation.

Specification

(and criticality)

The PTLR contains P/T limit curves for heatup, cooldown, and inservice leak and hydrostatic testing, and ~~data for~~ the maximum rate of change of reactor coolant temperature. The heatup curve provides limits for both heatup and criticality.

also limits

1

The P/T limit curves are applicable for 32 effective full power years.

Each P/T limit curve defines an acceptable region for normal operation. The usual use of the curves is operational guidance during heatup or cooldown maneuvering, when pressure and temperature indications are monitored and compared to the applicable curve to determine that operation is within the allowable region.

The LCO establishes operating limits that provide a margin to brittle failure of the reactor vessel and piping of the reactor coolant pressure boundary (RCPB). The vessel is the component most subject to brittle failure. Therefore, the LCO limits apply mainly to the vessel.

10 CFR 50, Appendix G (Ref. 1), requires the establishment of P/T limits for material fracture toughness requirements of the RCPB materials. Reference 1 requires an adequate margin to brittle failure during normal operation, anticipated operational occurrences, and system hydrostatic tests. It mandates the use of the American Society of Mechanical Engineers (ASME) Code, Section III, Appendix G (Ref. 2).

The actual shift in the RT_{NDT} of the vessel material will be established periodically by removing and evaluating the irradiated reactor vessel material specimens, in accordance with ASTM E 185 (Ref. 3) and 10 CFR 50, Appendix H (Ref. 4). The operating P/T limit curves will be adjusted,

(continued)

BASES

The non-nuclear heatup and cooldown curve 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.

2

BACKGROUND (continued)

as necessary, based on the evaluation findings and the recommendations of Reference 5.

The P/T limit curves are composite curves established by superimposing limits derived from stress analyses of those portions of the reactor vessel and head that are the most restrictive. At any specific pressure, temperature, and temperature rate of change, one location within the reactor vessel will dictate the most restrictive limit. Across the span of the P/T limit curves, different locations are more restrictive, and, thus, the curves are composites of the most restrictive regions.

~~The heatup curve represents a different set of restrictions than the cooldown curve because the directions of the thermal gradients through the vessel wall are reversed. The thermal gradient reversal alters the location of the tensile stress between the outer and inner walls.~~

4 P/T

The criticality limits include the Reference 1 requirement that they be at least 40°F above the non-critical heatup curve or the cooldown curve and not lower than the minimum permissible temperature for the inservice leak and hydrostatic testing. 2

The consequence of violating the LCO limits is that the RCS has been operated under conditions that can result in brittle failure of the RCPB, possibly leading to a nonisolable leak or loss of coolant accident. In the event these limits are exceeded, an evaluation must be performed to determine the effect on the structural integrity of the RCPB components. The ASME Code, Section XI, Appendix E (Ref. 6), provides a recommended methodology for evaluating an operating event that causes an excursion outside the limits.

APPLICABLE SAFETY ANALYSES

The P/T limits are not derived from Design Basis Accident (DBA) analyses. They are prescribed during normal operation to avoid encountering pressure, temperature, and temperature rate of change conditions that might cause undetected flaws to propagate and cause nonductile failure of the RCPB, a condition that is unanalyzed. Reference 7 ~~establishes the methodology for determining the P/T limits~~ approved the curves and limits required by this Specification. Since the P/T limits are not derived from any DBA, there are no acceptance limits related to the P/T limits. Rather, the P/T limits

2

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

are acceptance limits themselves since they preclude operation in an unanalyzed condition.

RCS P/T limits satisfy Criterion 2 of ~~the NRC Policy Statement~~ 10 CFR 50.36(c)(2)(ii). [2]

LCO

The elements of this LCO are:

a. RCS pressure and temperature ~~, and~~ heatup and ~~or~~ cooldown rates are within the limits specified in ~~the PTLR~~ Figures 3.4.11-1, 3.4.11-2, 3.4.11-3, 3.4.11-4, 3.4.11-5, and 3.4.11-6, and the RCS temperature change during system leakage and hydrostatic testing is ≤ 20°F in any 1 hour period when the RCS temperature and pressure are not within the limits of Figure 3.4.11-2, or 3.4.11-5, as applicable, during RCS heatup, cooldown, and inservice leak and hydrostatic testing are ≤ 100°F in any 1 hour period;

b. The temperature difference between the reactor vessel bottom head coolant and the reactor pressure vessel (RPV) coolant is ~~within the limit of the PTLR~~ ≤ 145°F during recirculation pump startup ~~, and during increases in THERMAL POWER or loop flow while operating at low THERMAL POWER or loop flow~~ in MODES 1, 2, 3, and 4;

c. The temperature difference between the reactor coolant in the respective recirculation loop and in the reactor vessel ~~meets the limit of the PTLR~~ is ≤ 50°F during recirculation pump startup ~~, and during increases in THERMAL POWER or loop flow while operating at low THERMAL POWER or loop flow~~ in MODES 1, 2, 3, and 4;

d. RCS pressure and temperature are within the applicable criticality limits specified in ~~the PTLR~~ Figures 3.4.11-3, and 3.4.11-6, prior to achieving criticality; and

e. The reactor vessel flange and the head flange temperatures are ~~within the limits of the PTLR~~ ≥ 72°F for Unit 1 and ≥ 86°F for Unit 2 when tensioning the reactor vessel head bolting studs and when the reactor head is tensioned. [1]

These limits define allowable operating regions and permit a large number of operating cycles while also providing a wide margin to nonductile failure.

The rate of change of temperature limits control the thermal gradient through the vessel wall and are used as inputs for calculating the heatup, cooldown, and inservice leak and hydrostatic testing P/T limit curves. Thus, the LCO for the rate of change of temperature restricts stresses caused by

(continued)

BASES

LCO (continued)

thermal gradients and also ensures the validity of the P/T limit curves.

Violation of the limits places the reactor vessel outside of the bounds of the stress analyses and can increase stresses in other RCS components. The consequences depend on several factors, as follows:

a. The severity of the departure from the allowable operating pressure temperature regime or the severity of the rate of change of temperature;

b. The length of time the limits were violated (longer violations allow the temperature gradient in the thick vessel walls to become more pronounced); and

c. The existences, sizes, and orientations of flaws in the vessel material. 3

APPLICABILITY

The potential for violating a P/T limit exists at all times. For example, P/T limit violations could result from ambient temperature conditions that result in the reactor vessel metal temperature being less than the minimum allowed temperature for boltup. Therefore, this LCO is applicable even when fuel is not loaded in the core.

ACTIONS

A.1 and A.2

Operation outside the P/T limits while in MODE 1, 2, or 3 must be corrected so that the RCPB is returned to a condition that has been verified by stress analyses.

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 engineering evaluation is required to determine if RCS operation can continue. The evaluation must verify the RCPB integrity remains acceptable and must be completed if continued operation is desired. Several methods may be used, including comparison with 4

(continued)

BASES

ACTIONS

A.1 and A.2 (continued)

pre-analyzed transients in the stress analyses, new analyses, or inspection of the components. 3

ASME Code, Section XI, Appendix E (Ref. 6), may be used to support the evaluation. However, its use is restricted to evaluation of the vessel beltline.

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.

Condition A is modified by a Note requiring Required Action A.2 be completed whenever the Condition is entered. The Note emphasizes the need to perform the evaluation of the effects of the excursion outside the allowable limits. Restoration alone per Required Action A.1 is insufficient because higher than analyzed stresses may have occurred and may have affected the RCPB integrity.

B.1 and B.2

If a Required Action and associated Completion Time of Condition A are not met, the plant must be brought to a lower MODE because either the RCS remained in an unacceptable P/T region for an extended period of increased stress, or a sufficiently severe event caused entry into an unacceptable region. Either possibility indicates a need for more careful examination of the event, best accomplished with the RCS at reduced pressure and temperature. With the reduced pressure and temperature conditions, the possibility of propagation of undetected flaws is decreased.

Pressure and temperature are reduced by bringing the plant to at least MODE 3 within 12 hours and to MODE 4 within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

(continued)

BASES

ACTIONS (continued)

C.1 and C.2

Operation outside the P/T limits in other than MODES 1, 2, and 3 (including defueled conditions) must be corrected so that the RCPB is returned to a condition that has been verified by stress analyses. The Required Action must be initiated without delay and continued until the limits are restored.

4 engineering

Besides restoring the P/T limit parameters to within limits, an evaluation is required to determine if RCS operation is allowed. This evaluation must verify that the RCPB integrity is acceptable and must be completed before approaching criticality or heating up to > 200°F. Several methods may be used, including comparison with pre-analyzed transients, new analyses, or inspection of the components. ASME Section XI, Appendix E (Ref. 6), may be used to support the evaluation; however, its use is restricted to evaluation of the beltline.

<Insert C.1 and C.2> 5

SURVEILLANCE REQUIREMENTS

SR 3.4.11.1

1

Verification that operation is within PTLR limits is required every 30 minutes when RCS pressure and temperature conditions are undergoing planned changes. This Frequency is considered reasonable in view of the control room indication available to monitor RCS status. Also, since temperature rate of change limits are specified in hourly increments, 30 minutes permits assessment and correction of minor deviations.

1 The limits of Figures 3.4.11-1, 3.4.11-2, 3.4.11-3, 3.4.11-4, 3.4.11-5, and 3.4.11-6 are met when operation is to the right of the applicable curve

Surveillance for heatup, cooldown, or inservice leakage and hydrostatic testing may be discontinued when the criteria given in the relevant plant procedure for ending the activity are satisfied.

2

This SR has been modified by a Note that requires this Surveillance to be performed only during system heatup and cooldown operations and inservice leakage and hydrostatic testing.

(continued)

INSERT C.1 AND C.2

Condition C is modified by a Note requiring Required Action C.2 be completed whenever the Condition is entered. The Note emphasizes the need to perform the evaluation of the effects of the excursion outside the allowable limits. Restoration alone per Required Action C.1 is insufficient because higher than analyzed stresses may have occurred and may have affected the RCPB integrity.

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.4.11.2

A separate limit is used when the reactor is approaching criticality. Consequently, the RCS pressure and temperature must be verified within the appropriate limits before withdrawing control rods that will make the reactor critical.

The limits of Figures 3.4.11-3 and 3.4.11-6 are met when operation is to the right of the applicable curve.

1

Performing the Surveillance within 15 minutes before control rod withdrawal for the purpose of achieving criticality provides adequate assurance that the limits will not be exceeded between the time of the Surveillance and the time of the control rod withdrawal.

TSTF-353 changes not adopted

SR 3.4.11.3 and SR 3.4.11.4

Differential temperatures within the applicable PTLR limits 1 ensure that thermal stresses resulting from the startup of an idle recirculation pump will not exceed design allowances. In addition, compliance with these limits ensures that the assumptions of the analysis for the startup of an idle recirculation loop (Ref. 8) are satisfied.

2

An acceptable means of demonstrating compliance with the temperature differential requirement in SR 3.4.11.3 is to compare temperatures of the reactor pressure vessel steam space coolant and the bottom head drain line coolant.

Performing the Surveillance within 15 minutes before starting the idle recirculation pump provides adequate assurance that the limits will not be exceeded between the time of the Surveillance and the time of the idle pump start.

An acceptable means of demonstrating compliance with the temperature differential requirement in SR 3.4.11.4 is to compare the temperatures of the operating recirculation loop and the idle loop.

and SR 3.4.11.4 have 1

SR 3.4.11.3 ~~has~~ been modified by a Note that requires the Surveillance to be met only in MODES 1, 2, 3, and 4 ~~with reactor steam dome pressure ≥ 25 psig~~. In MODE 5, the overall stress on limiting components is lower; therefore, ΔT limits are not required.

6

during a recirculation pump startup since this is when the stresses occur.

SR 3.4.11.5, SR 3.4.11.6, and SR 3.4.11.7

Limits on the reactor vessel flange and head flange temperatures are generally bounded by the other P/T limits

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.4.11.5, SR 3.4.11.6, and SR 3.4.11.7 (continued)

during system heatup and cooldown. However, operations approaching MODE 4 from MODE 5 and in MODE 4 with RCS temperature less than or equal to certain specified values require assurance that these temperatures meet the LCO limits.

every 30 minutes thereafter 5

within 5

The flange temperatures must be verified to be above the limits ~~30 minutes before and~~ while tensioning the vessel head bolting studs to ensure that once the head is tensioned the limits are satisfied. When in MODE 4 with RCS temperature ≤ ~~80~~ 77 °F for Unit 1 and ≤ 91°F for Unit 2 1, 30 minute checks of the flange temperatures are required because of the reduced margin to the limits. When in MODE 4 with RCS temperature ≤ ~~100~~ 92 °F for Unit 1 and ≤ 106°F for Unit 2 1, monitoring of the flange temperature is required every 12 hours to ensure the temperatures are within the limits specified ~~in the PTLR~~. 1

The 30 minute Frequency reflects the urgency of maintaining the temperatures within limits, and also limits the time that the temperature limits could be exceeded. The 12 hour Frequency is reasonable based on the rate of temperature change possible at these temperatures.

Insert SR 3.4.11.5, SR 3.4.11.6 and SR 3.4.11.7 5

REFERENCES

1. 10 CFR 50, Appendix G.

2. ASME, Boiler and Pressure Vessel Code, Section III, Appendix G.

3. ASTM E 185~~-82, July 1982~~. 2

4. 10 CFR 50, Appendix H.

5. Regulatory Guide 1.99, Revision 2, May 1988.

6. ASME, Boiler and Pressure Vessel Code, Section XI, Appendix E.

7. ~~NEDO-21778-A, December 1978.~~

8. U FSAR, Section ~~15.1.26~~ 15.4.4. 6

2 NRC Safety Evaluation supporting Amendment No. 71 to Facility Operating License No. NPF-11 and Amendment No. 55 to Facility Operating License No. NPF-18 — LaSalle County Station, Units 1 and 2, dated January 16, 1990. 2

INSERT SR 3.4.11.5, SR 3.4.11.6, AND SR 3.4.11.7

SR 3.4.11.5 is modified by a Note that requires the Surveillance to be performed only when tensioning the reactor vessel head bolting studs. SR 3.4.11.6 is modified by a Note that requires the Surveillance to be initiated 30 minutes after RCS temperature $\leq$ 77°F for Unit 1 and $\leq$ 91°F for Unit 2 in MODE 4, SR 3.4.11.7 is modified by a Note that requires the Surveillance to be initiated 12 hours after RCS temperature $\leq$ 92°F for Unit 1 and $\leq$ 106°F for Unit 2 in MODE 4. The Notes contained in these SRs are necessary to specify when the reactor vessel flange and head flange temperatures are required to be verified to be within the specified limits.

JUSTIFICATION FOR DEVIATIONS FROM NUREG-1434, REVISION 1
ITS BASES: 3.4.11 - RCS PRESSURE AND TEMPERATURE (P/T) LIMITS

1. Changes have been made to reflect those changes made to the Specification. The following requirements have been renumbered, where applicable, to reflect the changes.

2. Changes have been made (additions, deletions, and/or changes to the NUREG) to reflect the plant specific nomenclature, number, reference, system description, or analysis description.

3. Typographical/grammatical error corrected.

4. Editorial change made for enhanced clarity or to be consistent with similar statements in other places in the Bases.

5. Changes have been made to more closely match the LCO requirements.

6. The brackets have been removed and the proper plant specific information/value has been provided.

B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.12 Reactor Steam Dome Pressure

BASES

BACKGROUND 1

The reactor steam dome pressure is an assumed initial condition of Design Basis Accidents (DBAs) and transients~~.~~ ~~and is also~~ an assumed value in the determination of compliance with reactor pressure vessel overpressure protection criteria~~.~~ and is also

APPLICABLE SAFETY ANALYSES

The reactor steam dome pressure of ≤ ~~[1045]~~ 1020 4 psig is an initial condition of the vessel overpressure protection analysis of Reference 1. This analysis assumes an initial maximum reactor steam dome pressure and evaluates the response of the pressure relief system, primarily the safety/relief 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. Reference 2 also assumes an initial reactor steam dome pressure for the analysis of DBAs and transients used to determine the limits for fuel cladding integrity MCPR (see Bases for LCO 3.2.2, "MINIMUM CRITICAL POWER RATIO (MCPR)") and 1% cladding plastic strain (see Bases for LCO 3.2.1, "AVERAGE PLANAR LINEAR HEAT GENERATION RATE (APLHGR)~~)~~". 2 , and LCO 3.2.3, "LINEAR HEAT GENERATION RATE (LHGR)

2 Insert Applicable Safety Analyses

Reactor steam dome pressure satisfies the requirements of Criterion 2 of ~~the NRC Policy Statement~~ 10 CFR 50.36(c)(2)(ii). 2

LCO 4

The specified reactor steam dome pressure limit of ≤ ~~[1045]~~ 1020 psig ensures the plant is operated within the assumptions of the ~~transient analyses~~ reactor overpressure analysis 2. Operation above the limit may result in a transient response more severe than analyzed.

APPLICABILITY

In MODES 1 and 2, the reactor steam dome pressure is required to be less than or equal to the limit. In these

(continued)

INSERT APPLICABLE SAFETY ANALYSES

The nominal reactor operating pressure is approximately 1005 psig. Transient analyses typically use the nominal or a design dome pressure as input to the analysis. Small deviations (5 to 10 psi) from the nominal pressure are not expected to change most of the transient analyses results. However, sensitivity studies for fast pressurization events (main turbine generator load rejection without bypass, turbine trip without bypass, and feedwater controller failure) indicate that the delta-CPR may increase for lower initial pressures. Therefore, the fast pressurization events have considered a bounding initial pressure based on a typical operating range to assure a conservative delta-CPR and operating limit.

BASES

APPLICABILITY (continued)

2

events that may challenge the overpressure limits are possible.

MODES, the reactor may be generating significant steam, and ~~the DBAs and transients are bounding~~.

In MODES 3, 4, and 5, the limit is not applicable because the reactor is shut down. In these MODES, the reactor pressure is well below the required limit, and no anticipated events will challenge the overpressure limits.

ACTIONS

A.1

With the reactor steam dome pressure greater than the limit, prompt action should be taken to reduce pressure to below the limit and return the reactor to operation within the bounds of the analyses. The 15 minute Completion Time is reasonable considering the importance of maintaining the pressure within limits. This Completion Time also ensures that the probability of an accident while pressure is greater than the limit is minimal. ~~If the operator is unable to restore the reactor steam dome pressure to below the limit, then the reactor should be brought to MODE 3 to be within the assumptions of the transient analyses.~~

3

B.1

If the reactor steam dome pressure cannot be restored to within the limit within the associated Completion Time, the plant must be brought to a MODE in which the LCO does not apply. To achieve this status, the plant must be brought to at least MODE 3 within 12 hours. The allowed Completion Time of 12 hours is reasonable, based on operating experience, to reach MODE 3 from full power conditions in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.4.12.1

1020 4

2

vessel overpressure protection analysis is

Verification that reactor steam dome pressure is ≤ ~~[1045]~~ psig ensures that the initial condition~~s~~ of the ~~DBAs and transients are~~ met. Operating experience has shown the 12 hour Frequency to be sufficient for identifying trends and verifying operation within safety analyses assumptions.

2

(continued)

BASES (continued)

REFERENCES 2

1. U FSAR, Section ~~[~~5.2.2.2.~~3]~~ 1 4
2. U FSAR, ~~Section~~ Chapter 5 ~~[~~15~~]~~. 4

JUSTIFICATION FOR DEVIATIONS FROM NUREG-1434, REVISION 1
ITS BASES: 3.4.12 - REACTOR STEAM DOME PRESSURE

1. Editorial change made for enhanced clarity or to be consistent with similar statements in other places in the Bases.

2. Changes have been made (additions, deletions, and/or changes to the NUREG) to reflect the plant specific nomenclature, number, reference, system description, or analysis description.

3. Changes have been made to more closely match the LCO requirements.

4. The brackets have been removed and the proper plant specific information/value has been provided.

5. Typographical/grammatical error corrected.

GENERIC NO SIGNIFICANT HAZARDS CONSIDERATION
ITS: SECTION 3.4 - REACTOR COOLANT SYSTEM

ADMINISTRATIVE CHANGES
("A.x" Labeled Comments/Discussions)

In accordance with the criteria set forth in 10 CFR 50.92, ComEd has evaluated this proposed Technical Specifications change and determined it does not represent a significant hazards consideration. The following is provided in support of this conclusion.

1. Does the change involve a significant increase in the probability or consequences of an accident previously evaluated?

   The proposed change involves reformatting, renumbering, and rewording the existing Technical Specifications. The reformatting, renumbering, and rewording process involves no technical changes to the existing Technical Specifications. As such, this change is administrative in nature and does not impact initiators of analyzed events or assumed mitigation of accident or transient events. Therefore, this change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. Does the change create the possibility of a new or different kind of accident from any accident previously evaluated?

   The proposed change does not involve a physical alteration of the plant (no new or different type of equipment will be installed) or changes in methods governing normal plant operation. The proposed change will not impose any new or eliminate any old requirements. Thus, this change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. Does this change involve a significant reduction in a margin of safety?

   The proposed change will not reduce a margin of safety because it has no impact on any safety analyses assumptions. This change is administrative in nature. Therefore, the change does not involve a significant reduction in a margin of safety.

RELOCATED SPECIFICATIONS
("R.x" Labeled Comments/Discussions)

In accordance with the criteria set forth in 10 CFR 50.92, ComEd has evaluated this proposed Technical Specifications change and determined it does not represent a significant hazards consideration. The following is provided in support of this conclusion.

1. Does the change involve a significant increase in the probability or consequences of an accident previously evaluated?

   The proposed change relocates requirements and surveillances for structures, systems, components or variables that do not meet the criteria for inclusion in Technical Specifications as identified in the Application of Selection Criteria to the LaSalle 1 and 2 Technical Specifications. The affected structures, systems, components or variables are not assumed to be initiators of analyzed events and are not assumed to mitigate accident or transient events. The requirements and surveillances for these affected structures, systems, components or variables will be relocated from the Technical Specifications to an appropriate administratively controlled document which will be maintained pursuant to 10 CFR 50.59. In addition, the affected structures, systems, components or variables are addressed in existing surveillance procedures which are also controlled by 10 CFR 50.59 and subject to the change control provisions imposed by plant administrative procedures, which endorse applicable regulations and standards. Therefore, this change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. Does the change create the possibility of a new or different kind of accident from any accident previously evaluated.

   The proposed change does not involve a physical alteration of the plant (no new or different type of equipment will be installed) or a change in the methods governing normal plant operation. The proposed change will not impose or eliminate any requirements and adequate control of existing requirements will be maintained. Thus, this change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. Does this change involve a significant reduction in a margin of safety?

   The proposed change will not reduce a margin of safety because it has no impact on any safety analysis assumptions. In addition, the relocated requirements and surveillances for the affected structure, system, component or variable remain the same as the existing Technical Specifications. Since any future changes to these requirements or the surveillance procedures will be evaluated per the requirements of 10 CFR 50.59, no reduction in a margin of safety will be permitted.

RELOCATED SPECIFICATIONS
("R.x" Labeled Comments/Discussions

3. (continued)

The existing requirement for NRC review and approval of revisions, in accordance with 10 CFR 50.92, to these details proposed for relocation does not have a specific margin of safety upon which to evaluate. However, since the proposed change is consistent with the BWR ISTS, NUREG-1434, Rev. 1, approved by the NRC Staff, revising the Technical Specifications to reflect the approved level of detail ensures no significant reduction in the margin of safety.

TECHNICAL CHANGES - MORE RESTRICTIVE
("M.x" Labeled Comments/Discussions)

In accordance with the criteria set forth in 10 CFR 50.92, ComEd has evaluated this proposed Technical Specifications change and determined it does not represent a significant hazards consideration. The following is provided in support of this conclusion.

1. Does the change involve a significant increase in the probability or consequences of an accident previously evaluated?

   The proposed change provides more stringent requirements for operation of the facility. These more stringent requirements do not result in operation that will increase the probability of initiating an analyzed event and do not alter assumptions relative to mitigation of an accident or transient event. The more restrictive requirements continue to ensure process variables, structures, systems, and components are maintained consistent with the safety analyses and licensing basis. Therefore, this change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. Does the change create the possibility of a new or different kind of accident from any accident previously evaluated?

   The proposed change does not involve a physical alteration of the plant (no new or different type of equipment will be installed) or changes in the methods governing normal plant operation. The proposed change does impose different requirements. However, these changes are consistent with the assumptions in the safety analyses and licensing basis. Thus, this change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. Does this change involve a significant reduction in a margin of safety?

   The imposition of more restrictive requirements either has no impact on or increases the margin of plant safety. As provided in the discussion of the change, each change in this category is by definition, providing additional restrictions to enhance plant safety. The change maintains requirements within the safety analyses and licensing basis. Therefore, this change does not involve a significant reduction in a margin of safety.

"GENERIC" LESS RESTRICTIVE CHANGES:
RELOCATING DETAILS TO TECHNICAL SPECIFICATION BASES, UFSAR, TRM, OR OTHER PLANT CONTROLLED DOCUMENTS
("LA.x" Labeled Comments/Discussions)

In accordance with the criteria set forth in 10 CFR 50.92, ComEd has evaluated this proposed Technical Specifications change and determined it does not represent a significant hazards consideration. The following is provided in support of this conclusion.

1. Does the change involve a significant increase in the probability or consequences of an accident previously evaluated?

   The proposed change relocates certain details from the Technical Specifications to the Bases, UFSAR, TRM, or other plant controlled documents. The Bases, UFSAR, TRM, and other plant controlled documents containing the relocated information will be maintained in accordance with 10 CFR 50.59. In addition to 10 CFR 50.59 provisions, the Technical Specification Bases are subject to the change control provisions in the Administrative Controls Chapter of the ITS. The UFSAR is subject to the change control provisions of 10 CFR 50.71(e), and the plant procedures and other plant controlled documents are subject to controls imposed by plant administrative procedures, which endorse applicable regulations and standards. Since any changes to the Bases, UFSAR, TRM, or other plant controlled documents will be evaluated per the requirements of the Bases Control Program in Chapter 5.0 of the ITS or 10 CFR 50.59, no increase (significant or insignificant) in the probability or consequences of an accident previously evaluated will be allowed. Therefore, this change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. Does the change create the possibility of a new or different kind of accident from any accident previously evaluated?

   The proposed change does not involve a physical alteration of the plant (no new or different type of equipment will be installed) or a change in the methods governing normal plant operation. The proposed change will not impose or eliminate any requirements, and adequate control of the information will be maintained. Thus, this change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. Does this change involve a significant reduction in a margin of safety?

   The proposed change will not reduce a margin of safety because it has no impact on any safety analysis assumptions. In addition, the details to be transposed from the Technical Specifications to the Bases, UFSAR, TRM, or other plant controlled

"GENERIC" LESS RESTRICTIVE CHANGES:
RELOCATING DETAILS TO TECHNICAL SPECIFICATION BASES, UFSAR, TRM, OR OTHER PLANT CONTROLLED DOCUMENTS
("LA.x" Labeled Comments/Discussions)

3. (continued)

documents are the same as the existing Technical Specifications. Since any future changes to these details in the Bases, UFSAR, TRM, or other plant controlled documents will be evaluated per the requirements of 10 CFR 50.59, no reduction (significant or insignificant) in a margin of safety will be allowed. Based on 10 CFR 50.92, the existing requirement for NRC review and approval of revisions, to these details proposed for relocation, does not have a specific margin of safety upon which to evaluate. However, since the proposed change is consistent with the BWR ISTS, NUREG-1434, Rev. 1, approved by the NRC Staff, revising the Technical Specifications to reflect the approved level of detail ensures no significant reduction in the margin of safety.

"GENERIC" LESS RESTRICTIVE CHANGES: RELOCATION OF INSTRUMENTATION ONLY REQUIREMENTS ("LC.x" Labeled Comments/Discussions)

In accordance with the criteria set forth in 10 CFR 50.92, ComEd has evaluated this proposed Technical Specifications change and determined it does not represent a significant hazards consideration. The following is provided in support of this conclusion.

1. Does the change involve a significant increase in the probability or consequences of an accident previously evaluated?

   The proposed change relocates instrumentation requirements, which provide no post-accident function, from the Technical Specifications to the Bases, UFSAR, TRM, or other plant controlled documents. These requirements are part of the routine operational monitoring and are not considered in the safety analysis. The Bases, UFSAR, TRM, and other plant controlled documents containing the relocated information will be maintained in accordance with 10 CFR 50.59. In addition to 10 CFR 50.59 provisions, the Technical Specification Bases are subject to the change control provisions in the Administrative Controls Chapter of the ITS. The UFSAR is subject to the change control provisions of 10 CFR 50.71(e), and plant procedures and other plant controlled documents are subject to controls imposed by plant administrative procedures, which endorse applicable regulations and standards. Since any changes to the Bases, UFSAR, TRM, or other plant controlled documents will be evaluated per the requirements of the Bases Control Program in Chapter 5.0 of the ITS or 10 CFR 50.59, no increase (significant or insignificant) in the probability or consequences of an accident previously evaluated will be allowed. Therefore, this change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. Does the change create the possibility of a new or different kind of accident from any accident previously evaluated?

   The proposed change does not involve a physical alteration of the plant (no new or different type of equipment will be installed) or a change in the methods governing normal plant operation. The proposed change will not impose or eliminate any requirements, and adequate control of the requirements will be maintained. Thus, this change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

"GENERIC" LESS RESTRICTIVE CHANGES:
RELOCATION OF INSTRUMENTATION ONLY REQUIREMENTS
("LC.x" Labeled Comments/Discussions) (continued)

3. Does this change involve a significant reduction in a margin of safety?

The proposed change will not reduce a margin of safety because it has no impact on any safety analysis assumption. In addition, the requirements to be transposed from the Technical Specifications to the Bases, UFSAR, TRM, or other plant controlled documents are the same as the existing Technical Specifications. Since any future changes to these requirements in the Bases, UFSAR, TRM, or other plant controlled documents will be evaluated per the requirements of 10 CFR 50.59, no reduction (significant or insignificant) in a margin of safety will be allowed.

Based on 10 CFR 50.92, the existing requirement for NRC review and approval of revisions to these requirements proposed for relocation, does not have a specific margin of safety upon which to evaluate. However, since the proposed change is consistent with the BWR Standard Technical Specifications, NUREG-1434, Rev. 1, approved by the NRC Staff, revising the Technical Specifications to reflect the approved level of instrumentation requirements ensures no significant reduction in the margin of safety.

"GENERIC" LESS RESTRICTIVE CHANGES:
EXTENDING SURVEILLANCE FREQUENCIES FROM 18 MONTHS TO 24 MONTHS FOR SURVEILLANCES OTHER THAN CHANNEL CALIBRATIONS
("LD.x" Labeled Comments/Discussions)

In accordance with the criteria set forth in 10 CFR 50.92, ComEd has evaluated this proposed Technical Specifications change and determined it does not represent a significant hazards consideration. The following is provided in support of this conclusion.

1. Does the change involve a significant increase in the probability or consequences of an accident previously evaluated?

   The proposed change involves a change in the surveillance testing intervals from 18 months to 24 months. The proposed change does not physically impact the plant nor does it impact any design or functional requirements of the associated systems. That is, the proposed change does not degrade the performance or increase the challenges of any safety systems assumed to function in the accident analysis. The proposed change does not impact the Surveillance Requirements themselves nor the way in which the Surveillances are performed. Additionally, the proposed change does not introduce any new accident initiators since no accidents previously evaluated have as their initiators anything related to the frequency of surveillance testing. The proposed change does not affect the availability of equipment or systems required to mitigate the consequences of an accident because of the availability of redundant systems or equipment and because other test performed more frequently will identify potential equipment problems. Furthermore, an historical review of surveillance test results indicated that all failures identified were unique, non-repetitive, and not related to any time-based failure modes, and indicated no evidence of any failures that would invalidate the above conclusions. Therefore, the proposed change does not increase the probability or consequences of an accident previously evaluated.

2. Does the change create the possibility of a new or different kind of accident from any accident previously evaluated?

   The proposed change involves a change in the surveillance testing intervals from 18 months to 24 months. The proposed change does not introduce any failure mechanisms of a different type than those previously evaluated since there are no physical changes being made to the facility. In addition, the Surveillance Requirements themselves and the way Surveillances are performed will remain unchanged. Furthermore, an historical review of surveillance test results indicated no evidence of any failures that would invalidate the above conclusions. Therefore, the proposed change does not create the possibility of a new or different kind of accident from any previously evaluated.

"GENERIC" LESS RESTRICTIVE CHANGES:
EXTENDING SURVEILLANCE FREQUENCIES FROM 18 MONTHS TO 24 MONTHS FOR SURVEILLANCES OTHER THAN CHANNEL CALIBRATIONS
("LD.x" Labeled Comments/Discussions) (continued)

3. Does this change involve a significant reduction in a margin of safety?

   Although the proposed change will result in an increase in the interval between surveillance tests, the impact on system availability is minimal based on other, more frequent testing or redundant systems or equipment, and there is no evidence of any failures that would impact the availability of the systems. Therefore, the assumptions in the licensing basis are not impacted, and the proposed change does not involve a significant reduction in a margin of safety.

"GENERIC" LESS RESTRICTIVE CHANGES:
EXTENDING SURVEILLANCE FREQUENCIES FROM 18 MONTHS TO 24 MONTHS FOR CHANNEL CALIBRATION SURVEILLANCES
("LE.x" Labeled Comments/Discussions)

In accordance with the criteria set forth in 10 CFR 50.92, ComEd has evaluated this proposed Technical Specifications change and determined it does not represent a significant hazards consideration. The following is provided in support of this conclusion.

1. Does the change involve a significant increase in the probability or consequences of an accident previously evaluated?

   The proposed change involves a change in the instrumentation channel calibration surveillance testing intervals from 18 months to 24 months. The proposed change does not physically impact the plant nor does it impact any design or functional requirements of the associated systems. That is, the proposed change does not degrade the performance or increase the challenges of any safety systems assumed to function in the accident analysis. The proposed change does not impact the Surveillance Requirements themselves nor the way in which the Surveillances are performed. Additionally, the proposed change does not introduce any new accident initiators since no accidents previously evaluated have as their initiators anything related to the frequency of surveillance testing. The proposed change does not affect the availability of equipment or systems required to mitigate the consequences of an accident because of the availability of redundant systems or equipment and because other tests performed more frequently will identify potential equipment problems. Furthermore, an historical review of surveillance test results indicated that all failures identified were unique, non-repetitive, and not related to any time-based failure modes, and indicated no evidence of any failures that would invalidate the above conclusions. Therefore, the proposed change does not increase the probability or consequences of an accident previously evaluated.

2. Does the change create the possibility of a new or different kind of accident from any accident previously evaluated?

   The proposed change involves a change in the instrumentation channel calibration surveillance testing intervals from 18 months to 24 months. The proposed change does not introduce any failure mechanisms of a different type than those previously evaluated since there are no physical changes being made to the facility. In addition, the Surveillance Requirements themselves and the way Surveillances are performed will remain unchanged. Furthermore, an historical review of surveillance test results indicated no evidence of any failures that would invalidate the above conclusions. Therefore, the proposed change does not create the possibility of a new or different kind of accident from any previously evaluated.

"GENERIC" LESS RESTRICTIVE CHANGES:
EXTENDING SURVEILLANCE FREQUENCIES FROM 18 MONTHS TO 24 MONTHS FOR CHANNEL CALIBRATION SURVEILLANCES
("LE.x" Labeled Comments/Discussions) (continued)

3. Does this change involve a significant reduction in a margin of safety?

   Although the proposed change will result in an increase in the interval between surveillance tests, the impact on system availability is minimal based on other, more frequent testing or redundant systems or equipment, and there is no evidence of any failures that would impact the availability of the systems. Therefore, the assumptions in the licensing basis are not impacted, and the proposed change does not involve a significant reduction in a margin of safety.

NO SIGNIFICANT HAZARDS CONSIDERATION
ITS: 3.4.1 - RECIRCULATION LOOPS OPERATING

L.1 CHANGE

In accordance with the criteria set forth in 10 CFR 50.92, ComEd has evaluated this proposed Technical Specifications change and determined it does not represent a significant hazards consideration. The following is provided in support of this conclusion.

1. Does the change involve a significant increase in the probability or consequences of an accident previously evaluated?

   The proposed change increases the time to adjust power distribution limits and Reactor Protection System (RPS) and Control Rod Block instrumentation Allowable Values for single loop operation to 12 hours. The time required to perform power distribution limit adjustments or RPS or Control Rod Block instrumentation adjustment is not considered as an initiator of any accidents previously evaluated. Therefore, this change does not significantly increase the probability of a previously analyzed accident. Additionally, the consequences of an event occurring while the unit is in single loop operation without the power distribution limit adjustments or RPS or Control Rod Block instrumentation adjustments are not altered by the proposed change. Therefore, the change does not involve a significant increase in the consequences of an accident previously evaluated.

2. Does the change create the possibility of a new or different kind of accident from any accident previously evaluated?

   The proposed change does not involve new equipment, design or operations, but provides for additional time to complete the previously approved TS Actions. Therefore, the proposed change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. Does this change involve a significant reduction in a margin of safety?

   The proposed change allows additional time to adjust power distribution limits and Reactor Protection System (RPS) and Control Rod Block instrumentation Allowable Values. However, this condition occurs infrequently and any minor decrease in the margin during this additional time is offset by the benefit of not hastily adjusting the instrumentation for single loop operation which could increase the probability of a plant transient. The additional time to adjust the power distribution limits and instrumentation for single recirculation loop operation is considered acceptable based on the low probability of an accident occurring during this period and frequent core monitoring by operations allowing abrupt changes in core flow conditions to be quickly detected. Therefore, the proposed change does not involve a significant reduction in the margin of safety.

L.2 CHANGE

In accordance with the criteria set forth in 10 CFR 50.92, ComEd has evaluated this proposed Technical Specifications change and determined it does not represent a significant hazards consideration. The following is provided in support of this conclusion.

1. Does the change involve a significant increase in the probability or consequences of an accident previously evaluated?

   This change extends the time to reach MODE 3 from 6 hours to 12 hours, which provides a reasonable amount of time to perform an orderly plant shutdown, thus further minimizing a potential unit upset from a too rapid decrease in plant power. This change does not result in any hardware or operating procedure changes. Reactor Recirculation System operation in natural circulation is not assumed to be an initiator of any analyzed event. Additionally, the consequences of an event occurring while the unit is decreasing power during the extra time are the same as the consequences of an event occurring for the current shutdown time. Therefore, the change does not involve a significant increase in the probability of an accident previously evaluated.

2. Does the change create the possibility of a new or different kind of accident from any accident previously evaluated?

   The change does not introduce a new mode of plant operation and does not involve physical modification to the plant. Therefore, it does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. Does this change involve a significant reduction in a margin of safety?

   The increased time allowed for reaching the applicable condition with no recirculation loops in operation is acceptable since in a natural circulation condition, the severity of a DBA is reduced, and there is minimal dependence on the recirculation loop coastdown characteristics. The time extension to shutdown provides sufficient time for the unit to reach the applicable condition in an orderly manner. As a result, the potential for human error is reduced. As such, any reduction in a margin of safety is insignificant and offset by the benefit gained from providing sufficient time to reach the applicable condition, thus avoiding potential plant transients from attempting to reach the applicable condition in the current period of time.

L.3 CHANGE

In accordance with the criteria set forth in 10 CFR 50.92, ComEd has evaluated this proposed Technical Specifications change and determined it does not represent a significant hazards consideration. The following is provided in support of this conclusion.

1. Does the change involve a significant increase in the probability or consequences of an accident previously evaluated?

   The proposed change to the Surveillance Frequency would allow time to perform the Surveillance when required. However, recirculation pump flow mismatch is not considered as an initiator of any previously evaluated accident. Therefore, the proposed change will not increase the probability of any accident previously evaluated. Additionally, the proposed Surveillance Frequency will continue to provide adequate confirmation of the appropriate operation of the recirculation pumps at the earliest opportunity when they are required. Therefore, the proposed change will not increase the consequences of any accident previously evaluated.

2. Does the change create the possibility of a new or different kind of accident from any accident previously evaluated?

   The proposed change does not introduce a new mode of plant operation and does not involve physical modification to the plant. Therefore, it does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. Does this change involve a significant reduction in a margin of safety?

   This change does not involve a significant reduction in a margin of safety since the proposed Surveillance Frequency will continue to provide the necessary assurance of appropriate operation of the recirculation pumps at the earliest opportunity, while providing time to perform the Surveillance.

NO SIGNIFICANT HAZARDS CONSIDERATION
ITS: 3.4.2 - FLOW CONTROL VALVES (FCVs)

There were no plant specific less restrictive changes identified for this Specification.

NO SIGNIFICANT HAZARDS CONSIDERATION
ITS: 3.4.3 - JET PUMPS

L.1 CHANGE

In accordance with the criteria set forth in 10 CFR 50.92, ComEd has evaluated this proposed Technical Specifications change and determined it does not represent a significant hazards consideration. The following is provided in support of this conclusion.

1. Does the change involve a significant increase in the probability or consequences of an accident previously evaluated?

   The proposed change would allow 4 hours to perform the Surveillance after placing a recirculation loop in operation. The jet pumps are not considered as initiators of any previously evaluated accident. Therefore, the proposed change will not increase the probability of any accident previously evaluated. Additionally, the proposed Surveillance will continue to provide adequate confirmation of the OPERABILITY of the jet pumps. Therefore, the proposed change will not increase the consequences of any accident previously evaluated.

2. Does the change create the possibility of a new or different kind of accident from any accident previously evaluated?

   The proposed change does not introduce a new mode of plant operation and does not involve physical modification to the plant. Therefore, it does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. Does this change involve a significant reduction in a margin of safety?

   This change does not involve a significant reduction in a margin of safety since the proposed Surveillance will continue to provide the necessary assurance of OPERABILITY of the jet pumps at the earliest opportunity.

NO SIGNIFICANT HAZARDS CONSIDERATION
ITS: 3.4.3 - JET PUMPS

L.2 CHANGE

In accordance with the criteria set forth in 10 CFR 50.92, ComEd has evaluated this proposed Technical Specifications change and determined it does not represent a significant hazards consideration. The following is provided in support of this conclusion.

1. Does the change involve a significant increase in the probability or consequences of an accident previously evaluated?

   The jet pumps are not considered as initiators of any previously evaluated accidents. Therefore the proposed change will not increase the probability of any accident previously evaluated. The proposed change merely limits testing to be performed in appropriate plant configurations to ensure meaningful data is obtained. The change does not affect the requirement for jet pump operability, or to perform the surveillance test. Therefore, the proposed change will not increase the consequences of any accident previously evaluated.

2. Does the change create the possibility of a new or different kind of accident from any accident previously evaluated?

   The proposed change does not introduce a new mode of plant operation and does not involve physical modification to the plant. The proposed change merely provides an opportunity to perform the required test at a time when it results in repeatable and meaningful data. The requirement for jet pump operability, and to perform the associated surveillance testing is not changed by the proposed change. Therefore, the proposed change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. Does this change involve a significant reduction in a margin of safety?

   The proposed change does not affect the margin of safety because it merely permits the surveillance testing to be performed at an appropriate time and plant operating condition. The proposed change will continue to provide the necessary assurance of jet pump operability at the earliest opportunity. The requirement for jet pump operability is not affected by the proposed change and therefore it does not involve a significant reduction in the margin of safety.

L.3 CHANGE

In accordance with the criteria set forth in 10 CFR 50.92, ComEd has evaluated this proposed Technical Specifications change and determined it does not represent a significant hazards consideration. The following is provided in support of this conclusion.

1. Does the change involve a significant increase in the probability or consequences of an accident previously evaluated?

   The jet pumps are not considered as initiators of any previously evaluated accidents. Therefore the proposed change will not increase the probability of any accident previously evaluated. The proposed change removes the requirement to perform jet pump surveillance testing on jet pumps that are not in operation since the system characteristics, and instrumentation available to perform the surveillance test, do not provide meaningful data in this condition. The change does not affect the requirement for jet pump operability, it merely removes the requirement to perform a surveillance test on pumps associated with a loop not in operation. Therefore, the proposed change will not increase the consequences of any accident previously evaluated.

2. Does the change create the possibility of a new or different kind of accident from any accident previously evaluated?

   The proposed change does not introduce a new mode of plant operation and does not involve physical modification to the plant. The proposed change merely permits performing the required testing on equipment that will provide repeatable and meaningful data. The requirement for jet pump operability is not changed by the proposed change. Therefore, the proposed change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. Does this change involve a significant reduction in a margin of safety?

   The proposed change does not affect the margin of safety because it merely limits the surveillance testing to be performed to an appropriate plant operating configuration. The requirement for jet pump operability is not affected by the proposed change and therefore it does not involve a significant reduction in the margin of safety.

L.4 CHANGE

In accordance with the criteria set forth in 10 CFR 50.92, ComEd has evaluated this proposed Technical Specifications change and determined it does not represent a significant hazards consideration. The following is provided to support this conclusion.

1. Does the change involve a significant increase in the probability or consequences of an accident previously evaluated?

   The proposed change adjusts the jet pump Surveillance acceptance criteria from 10% to 20% for individual jet pump diffuser-to-lower plenum differential pressure variations from the established pattern. This change corrects an error made in the Technical Specifications. The error resulted in LaSalle Units 1 and 2 acceptance criteria being more conservative than required. SIL-330 and NUREG/CR-3052 recommend certain requirements be met for the jet pumps to be Operable. One specified a 10% criteria from individual jet pump flow distribution. When measured by jet pump diffuser-to-lower plenum differential pressure the equivalent limit is 20% because of the relationship between flow and delta-P. Since LaSalle Units 1 and 2 utilize the diffuser-to-lower plenum differential pressure measurement, the variance allowed is being changed to 20% as was recommended in SIL-330 and NUREG/CR-3052. The proposed change does not effect the probability of an accident, because the jet pumps are not assumed to be an initiator of any analyzed event. This change increases the variance allowed in a Surveillance acceptance criteria consistent with the recommendations of the SIL and NUREG. Adopting the recommendations of the SIL and NUREG, which are the recommendations to ensure jet pump Operability, will not affect the consequences of an accident since the recommended acceptance criteria still provide adequate assurance the jet pumps are Operable. This change will not alter assumptions relative to the mitigation of an accident or transient event. Therefore, this change will not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. Does the change create the possibility of a new or different kind of accident from any accident previously evaluated?

   The proposed change adjusts the jet pump Surveillance acceptance criteria from 10% to 20% for individual jet pump diffuser-to-lower plenum differential pressure variations from the established pattern. This change corrects an error in the Technical Specifications. The error resulted in LaSalle Units 1 and 2 acceptance criteria being more conservative than required. The proposed changes to adopt the recommended acceptance criteria will not create the possibility of an accident. This change will not physically alter the plant (no new or different type of equipment will be installed). The changes in methods governing normal plant operation are consistent with the current safety analysis assumptions. Therefore, this change will not create the possibility of a new or different kind of accident from any accident previously evaluated.

L.4 CHANGE (continued)

3. Does this change involve a significant reduction in a margin of safety?

The proposed change adjusts the jet pump Surveillance acceptance criteria from 10% to 20% for individual jet pump diffuser-to-lower plenum differential pressure variations from the established pattern. This change corrects an error in the Technical Specifications. The error resulted in LaSalle Units 1 and 2 acceptance criteria being more conservative than required. The margin of safety is not significantly reduced because the proposed changes to the acceptance criteria will continue to verify jet pump Operability. The changes reflect the recommendations in SIL-330 and NUREG/CR-3052. The safety analysis assumptions will still be maintained, thus no question of safety exits. In addition, this change provides the benefit of avoiding a shutdown transient, when the jet pumps are still capable of performing their safety function. Therefore, this change does not involve a significant reduction in a margin of safety.

NO SIGNIFICANT HAZARDS CONSIDERATION
ITS: 3.4.4 - SAFETY/RELIEF VALVES (S/RVs)

L.1 CHANGE

In accordance with the criteria set forth in 10 CFR 50.92, ComEd has evaluated this proposed Technical Specifications change and determined it does not represent a significant hazards consideration. The following is provided in support of this conclusion.

1. Does the change involve a significant increase in the probability or consequences of an accident previously evaluated?

   The proposed change does not substantively change the requirements applicable to the Safety/Relief Valves (S/RVs) by removing the detail of valve position from CTS 3.4.2. Other specifications continue to require that the primary system pressure boundary remain intact (i.e. the valves remain closed). Since the valves are required to continue to function as a portion of the reactor pressure boundary, the proposed change does not result in any change to the operations of the facility. Therefore the proposed change will not increase the probability or consequences of any accident previously evaluated.

2. Does the change create the possibility of a new or different kind of accident from any accident previously evaluated?

   The proposed change does not introduce a new mode of plant operation and does not involve physical modification to the plant. The proposed change merely removes a requirement to maintain the valves in their closed position that is redundant with other specifications. The overall technical specification requirement for the valves to be maintained closed is not changed by the proposed change. Therefore, the proposed change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. Does this change involve a significant reduction in a margin of safety?

   The proposed change does not affect the margin of safety because it merely removes a redundant requirement from one specification. The requirement the S/RVs to remain closed is not affected by the proposed change and other Technical Specifications requirements are adequate to ensure the S/RVs remain closed. Therefore, the change does not involve a significant reduction in the margin of safety.

L.2 CHANGE

In accordance with the criteria set forth in 10 CFR 50.92, ComEd has evaluated this proposed Technical Specifications change and determined it does not represent a significant hazards consideration. The following is provided in support of this conclusion.

1. Does the change involve a significant increase in the probability or consequences of an accident previously evaluated?

   The S/RVs safety function is to provide overpressure protection for the reactor coolant system. Any failure associated with the low-low set instrumentation logic does not impact the ability of the S/RVs to mechanically open on overpressure. Therefore, removing the requirement for a channel calibration of the low-low set function does not effect the safety function of the valves. Since the change does not effect the ability of the valves to perform their safety function as described in the accident analyses, the proposed change will not increase the probability or consequences of any accident previously evaluated.

2. Does the change create the possibility of a new or different kind of accident from any accident previously evaluated?

   The proposed change does not introduce a new mode of plant operation and does not involve physical modification to the plant. The proposed change removes a requirement to perform a test of instrumentation logic that does not effect the ability of the S/RVs to perform their credited functions. The requirement for S/RV operability is not changed by the proposed change. The Technical Specification requirements will continue to provide adequate assurance that S/RV operability is maintained. Therefore, the proposed change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. Does this change involve a significant reduction in a margin of safety?

   The proposed change does not affect the margin of safety because it merely limits the surveillance testing to be performed to an appropriate plant operating configuration. The requirement for S/RV operability is not affected by the proposed change. The Technical Specifications requirements will continue to provide adequate assurance that S/RV operability is maintained. Therefore, the change does not involve a significant reduction in the margin of safety.

L.3 CHANGE

In accordance with the criteria set forth in 10 CFR 50.92, ComEd has evaluated this proposed Technical Specifications change and determined it does not represent a significant hazards consideration. The following is provided in support of this conclusion.

1. Does the change involve a significant increase in the probability or consequences of an accident previously evaluated?

   The proposed change does not affect the relief capacity required to be operable to provide overpressure protection for the reactor coolant system. The proposed change merely removes a limit on the duration in which it is acceptable to use S/RVs with lower setpoints in lieu of inoperable S/RVs. While the CTS limit the duration of this practice to the next refueling outage, the proposed ITS remove this limit on the duration. The only change is the amount of time in which the plant is allowed to operate in this configuration. The duration of operations in this configuration is unrelated to any accident probability. Therefore, the proposed change will not increase the probability of any accident previously evaluated.

   The proposed change removes an arbitrary limit on the duration of operations with two S/RVs with lower setpoints replacing inoperable S/RVs. The duration of operations in this configuration is unrelated to the consequences of any accident since the S/RVs remain capable of mitigating the consequences of an overpressure event as assumed in the safety analysis. Therefore, the proposed change will not increase the consequences of any accident previously evaluated.

2. Does the change create the possibility of a new or different kind of accident from any accident previously evaluated?

   The proposed change does not introduce a new mode of plant operation and does not involve physical modification to the plant. The proposed change merely allows operating in a currently approved configuration indefinitely. The amount of time the plant is operated in this configuration has no effect on the ability of the S/RVs to perform their safety function. The requirement for S/RV operability is not changed by the proposed change. The S/RVs will continue to provide overpressure protection as assumed in the safety analysis. Therefore, the proposed change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

L.3 CHANGE (continued)

3. Does this change involve a significant reduction in a margin of safety?

The proposed change does not impact the assumptions of any safety analysis. The proposed change does not affect the margin of safety. It merely permits the operation in currently accepted plant operating configuration indefinitely rather than for a limited time period. The requirement for S/RV operability is not affected by the proposed change. The S/RVs capability to provide overpressure protection as assumed in the safety analysis is maintained. Therefore, the change does not involve a significant reduction in the margin of safety.

NO SIGNIFICANT HAZARDS CONSIDERATION
ITS: 3.4.5 - RCS OPERATIONAL LEAKAGE

L.1 CHANGE

In accordance with the criteria set forth in 10 CFR 50.92, ComEd has evaluated this proposed Technical Specifications change and determined it does not represent a significant hazards consideration. The following is provided in support of this conclusion.

1. Does the change involve a significant increase in the probability or consequences of an accident previously evaluated?

   This change would decrease the Surveillance Frequency of CTS 4.4.3.2.1, the RCS Operational LEAKAGE verification, so that it is required to be performed every 12 hours instead of on average every 8 hours, not to exceed 12 hours. This change essentially allows the 25% extension specified in proposed SR 3.0.2 to be applied to the current 12 hour surveillance interval. The proposed change does not affect the actual leakage limit, and the normal Surveillance Frequency is consistent with NRC guidance provided in Generic Letter 88-01, Supplement 1. The probability of a pipe break occurring in the primary containment during the 25% extension period is small and the vast majority of the surveillances are completed with no indication of excessive RCS Operational LEAKAGE. Therefore, the proposed change does not involve a significant increase in the probability of an accident previously evaluated. Further, since the change impacts only the frequency of verification and does not result in any change in the actual leakage limit, the change does not increase the consequences of any previously analyzed accident.

2. Does the change create the possibility of a new or different kind of accident from any accident previously evaluated?

   This change does not result in any changes to the equipment design or capabilities or to the operation of the plant. Further, since the change impacts only the frequency of verification and does not change the leakage limit, the change does not create the possibility of a new or different kind of accident from any previously analyzed accident.

3. Does this change involve a significant reduction in a margin of safety?

   This change impacts only the frequency of verification of the leakage limit. Since the leakage is routinely monitored and alarms are provided for excessive leakage and industry experience has shown the leakage is, with few exceptions, always found to be within limits, the proposed 12 hour frequency (including the 25% extension allowed by proposed SR 3.0.2) will provide the same assurance as the current average 8 hour, not to exceed 12 hour, frequency. Therefore, this change does not involve a significant reduction in the margin of safety.

L.2 CHANGE

In accordance with the criteria set forth in 10 CFR 50.92, ComEd has evaluated this proposed Technical Specifications change and determined it does not represent a significant hazards consideration. The following is provided in support of this conclusion.

1. Does the change involve a significant increase in the probability or consequences of an accident previously evaluated?

   The proposed change would revise the Applicability of the unidentified leakage rate increase to include only MODE 1, instead of the current MODES 1, 2, and 3. The limit is intended to be applied to changes from normal steady state operation leakage rates. These are typically established at operating pressure and temperatures consistent with MODE 1. In this manner, a change that indicates a potential problem can be investigated prior to a catastrophic pipe rupture. However, a change during a heatup or startup that does not exceed an unidentified leakage of 5 gpm, in most cases, does not indicate a potential problem that could result in a catastrophic pipe rupture. The overall unidentified LEAKAGE limit of 5 gpm remains unchanged and will ensure changes that exceed this limit will not go unrecognized in MODES 2 and 3. Therefore, the probability and consequences of a previously analyzed accident are not significantly increased.

2. Does the change create the possibility of a new or different kind of accident from any accident previously evaluated?

   The proposed change does not introduce a new mode of plant operation and does not involve physical modification to the plant. Therefore, it does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. Does this change involve a significant reduction in a margin of safety?

   This change does not involve a significant reduction in a margin of safety since the proposed change does not modify the total unidentified LEAKAGE limit, and this limit is well below the leakage rate expected just prior to the onset of rapid crack propagation.

NO SIGNIFICANT HAZARDS CONSIDERATION
ITS: 3.4.6 - RCS PRESSURE ISOLATION VALVE (PIV) LEAKAGE

L.1 CHANGE

In accordance with the criteria set forth in 10 CFR 50.92, ComEd has evaluated this proposed Technical Specifications change and determined it does not represent a significant hazards consideration. The following is provided in support of this conclusion.

1. Does the change involve a significant increase in the probability or consequences of an accident previously evaluated?

   The proposed change would allow continued operation of the RHR Shutdown Cooling System with PIV leakage above the limits. However, this exception is provided only when the system is in operation i.e., only when the reactor coolant pressure has been reduced to below the RHR cut-in permissive pressure. Therefore, this change does not increase the probability of any accident previously evaluated. Additionally, the PIVs do not provide any accident mitigation functions once the Reactor Coolant System is depressurized. Therefore, the proposed change will not increase the consequences of any accident previously evaluated.

2. Does the change create the possibility of a new or different kind of accident from any accident previously evaluated?

   The proposed change does not introduce a new mode of plant operation and does not involve physical modification to the plant. Therefore, it does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. Does this change involve a significant reduction in a margin of safety?

   This change does not involve a significant reduction in a margin of safety since the proposed exception provides for the intended use of the primary system for decay heat removal.

NO SIGNIFICANT HAZARDS CONSIDERATION
ITS: 3.4.6 - RCS PRESSURE ISOLATION VALVE (PIV) LEAKAGE

L.2 CHANGE

In accordance with the criteria set forth in 10 CFR 50.92, ComEd has evaluated this proposed Technical Specifications change and determined it does not represent a significant hazards consideration. The following is provided in support of this conclusion.

1. Does the change involve a significant increase in the probability or consequences of an accident previously evaluated?

   The change provides an increased time for isolation of the second valve for double isolation of a high-low pressure interface flow path that does not meet PIV leakage limits. This change does not involve a significant increase in the probability or consequences of an accident previously evaluated. The isolation of the first valve within the existing time frame provides assurance that the leakage can be contained, based on the low probability of a second valve in the flow path failing to meet the leakage criteria during this time frame and the low probability of a pressure boundary rupture of the low pressure ECCS piping when overpressurized to reactor pressure (as cited in NEDC-31339, Nov. 1986).

2. Does the change create the possibility of a new or different kind of accident from any accident previously evaluated?

   The proposed change does not introduce a new mode of operation or physical modification of the plant. Operation with the high low pressure interface compromised by leaking PIVs is still restricted and must be responded to within 4 hours. The double isolation time requirement for the flow path has been extended, but does not preclude that the flow path is isolated by two valves. No new or different type of accident from any previously evaluated is created.

3. Does this change involve a significant reduction in a margin of safety?

   The change in isolation time for the second isolation of the leaking high-low pressure interface valves does not involve a significant reduction in the margin of safety. Isolation is still accomplished by one valve in the previously required time frame. Based on the low probability of a second isolation valve failure and the low probability of the rupture of an ECCS system due to overpressurization by reactor pressure, a significant reduction in the margin of safety does not result.

L.3 CHANGE

In accordance with the criteria set forth in 10 CFR 50.92, ComEd has evaluated this proposed Technical Specifications change and determined it does not represent a significant hazards consideration. The following is provided in support of this conclusion.

1. Does the change involve a significant increase in the probability or consequences of an accident previously evaluated?

   The requirement to perform a leak test to verify the restoration of a PIV is not assumed in the initiation of any analyzed event. This requirement was specified in the Technical Specifications to ensure the leakage of a restored PIV was positively verified to be within limits following repair, maintenance, or replacement work that could affect the valve leakage rate. The proposed deletion of this explicit requirement is considered acceptable since proposed SR 3.0.1 requires the appropriate SRs to be performed to demonstrate OPERABILITY after restoration of a component that caused the SR to be failed. In this case, proposed SR 3.0.1 would require proposed SR 3.4.6.1 to be performed, which requires a leak test of the PIV be performed. As a result, the accident consequences are unaffected by this change. Therefore, this change will not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. Does the change create the possibility of a new or different kind of accident from any accident previously evaluated?

   The possibility of a new or different kind of accident from any accident previously evaluated is not created because the proposed change does not introduce a new mode of plant operation and does not involve physical modification to the plant.

3. Does this change involve a significant reduction in a margin of safety?

   The proposed deletion of the explicit requirement to perform a leak test on the affected PIV following repair, maintenance, or replacement work that could affect the leak rate of the valve is considered acceptable since proposed SR 3.0.1 requires the appropriate SRs to be performed to demonstrate OPERABILITY after restoration of a component that caused the SR to be failed. In this case, proposed SR 3.0.1 would require proposed SR 3.4.6.1 to be performed, which requires a leak test of the PIV be performed. As a result, the existing requirement to perform a leak test on the affected PIV following repair, maintenance, or replacement work that could affect the leak rate of the valve is maintained. Therefore, this change does not involve a significant reduction in a margin of safety.

L.4 CHANGE

In accordance with the criteria set forth in 10 CFR 50.92, ComEd has evaluated this proposed Technical Specifications change and determined it does not represent a significant hazards consideration. The following is provided in support of this conclusion.

1. Does the change involve a significant increase in the probability or consequences of an accident previously evaluated?

   This change does not result in any hardware or operating procedure changes. The PIV leakage limits are not assumed to be an initiator of any analyzed event. The leakage limits provide assurance of valve integrity thereby reducing the probability of gross PIV failure and thereby eliminating potential consequences. The change to the limits acknowledges that smaller valves should not be allowed to leak as much as larger valves. The change provides assurance the PIVs will not be subject to gross failure due to leakage. In addition, an evaluation has been performed, NEDC-31339, "BWR Owners' Group Assessment of Emergency Core Cooling System Pressurization in Boiling Water Reactors," November 1986, that showed the probability of a pressure boundary rupture of the low pressure ECCS piping when over pressurized to the reactor pressure is very low. Therefore, this change will not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. Does the change create the possibility of a new or different kind of accident from any accident previously evaluated?

   The proposed change does not introduce a new mode of plant operation and does not involve a physical modification to the plant. Therefore, it does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. Does this change involve a significant reduction in a margin of safety?

   The change to PIV leakage limits is acceptable since it recognizes the difference in allowable leakage based on valve size while assuring that leakage limits are maintained such that the probability for gross PIV failure is reduced. Additionally, the Technical Specification limitation on total allowable RCS leakage continues to be maintained. Therefore, any reduction in a margin of safety will be insignificant, and offset by the benefit of avoiding an unnecessary maintenance when PIV leakage would not be indicative of gross PIV failure.

NO SIGNIFICANT HAZARDS CONSIDERATION
ITS: 3.4.7 - RCS LEAKAGE DETECTION INSTRUMENTATION

L.1 CHANGE

In accordance with the criteria set forth in 10 CFR 50.92, ComEd has evaluated this proposed Technical Specifications change and determined it does not represent a significant hazards consideration. The following is provided in support of this conclusion.

1. Does the change involve a significant increase in the probability or consequences of an accident previously evaluated?

   The proposed change would allow continued operation with inoperable leakage detection systems. The leakage detection systems are not considered as initiators of any previously evaluated accident. However, they may provide information to the operator of potential conditions that may be precursors to an accident. In the proposed conditions, sufficient indication will remain OPERABLE to provide the operator with the information necessary to evaluate the potential precursor conditions. Therefore, the proposed change will not significantly increase the probability of any accident previously evaluated. Additionally, the leakage detection systems do not provide any accident mitigation functions. Therefore, the proposed change will not significantly increase the consequences of any accident previously evaluated.

2. Does the change create the possibility of a new or different kind of accident from any accident previously evaluated?

   The proposed change does not introduce a new mode of plant operation and does not involve physical modification to the plant. Therefore, it does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. Does this change involve a significant reduction in a margin of safety?

   This change does not involve a significant reduction in a margin of safety since the proposed LCO will maintain adequate indications to the operator, and in addition will continue to provide appropriate compensatory measures.

L.2 CHANGE

In accordance with the criteria set forth in 10 CFR 50.92, ComEd has evaluated this proposed Technical Specifications change and determined it does not represent a significant hazards consideration. The following is provided in support of this conclusion.

1. Does the change involve a significant increase in the probability or consequences of an accident previously evaluated?

   Mode changes are proposed to be allowed with portions of the leakage detection instrumentation inoperable. The RCS leakage detection instrumentation is not considered to be an initiator for any previously evaluated accident. Therefore, the probability of an accident previously evaluated is not significantly increased. However, they do provide information to the operator of potential conditions that may be precursors to an accident. In the proposed conditions, sufficient indication will remain OPERABLE to provide the operator with the information necessary to evaluate the potential precursor conditions. Therefore, the proposed change will not increase the probability of any accident previously evaluated. Additionally, the leakage detection systems do not provide any accident mitigation functions. Therefore, the proposed change will not increase the consequences of any accident previously evaluated.

2. Does the change create the possibility of a new or different kind of accident from any accident previously evaluated?

   The proposed change does not introduce a new mode of plant operation and does not involve physical modification to the plant. Therefore, it does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. Does this change involve a significant reduction in a margin of safety?

   This change does not involve a significant reduction in a margin of safety since the proposed LCO will maintain adequate indications to the operator, and in addition will continue to provide appropriate compensatory measures.

L.3 CHANGE

In accordance with the criteria set forth in 10 CFR 50.92, ComEd has evaluated this proposed Technical Specifications change and determined it does not represent a significant hazards consideration. The following is provided in support of this conclusion.

1. Does the change involve a significant increase in the probability or consequences of an accident previously evaluated?

   The change modifies the Surveillance to indicate when a channel is placed in an inoperable status solely for performance of required Surveillances, entry into associated Conditions and Required Actions may be delayed for up to 6 hours, provided the other required Leakage Detection System channel is OPERABLE. The Leakage Detection System Instrumentation is not considered as an initiator for any accidents previously analyzed. Therefore, this change does not significantly increase the probability of a previously analyzed accident. Also, this change does not further degrade the capability of the monitors to perform their required function under these circumstances since one channel is still OPERABLE. Therefore, this change does not significantly increase the consequences of a previously analyzed accident.

2. Does the change create the possibility of a new or different kind of accident from any accident previously evaluated?

   The proposed change does not introduce a new mode of plant operation and does not involve physical modification to the plant. Therefore, it does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. Does this change involve a significant reduction in a margin of safety?

   This change does not involve a significant reduction in a margin of safety since the monitors are not required to provide automatic response to any design basis accident. The additional time does not significantly affect the contribution of the monitors to risk reduction since the function is still being monitored by the other OPERABLE channel.

L.1 CHANGE

In accordance with the criteria set forth in 10 CFR 50.92, ComEd has evaluated this proposed Technical Specifications change and determined it does not represent a significant hazards consideration. The following is provided in support of this conclusion.

1. Does the change involve a significant increase in the probability or consequences of an accident previously evaluated?

   The proposed change deletes CTS 3.4.5.b, which requires that the reactor coolant gross specific activity remain less than or equal to 100/E-bar μCi/gm and the Surveillance Requirements to determine gross beta/gamma activity at least once per 72 hours and to determine E-bar at least once per 6 months, CTS Table 4.4.5-1, Items 1 and 3. The proposed change also deletes CTS 3.4.5 Actions a.2 and b associated with LCO 3.4.5.b that requires the plant to be in HOT SHUTDOWN with the main steam isolation valves closed within 12 hours and to perform sampling requirements of CTS Table 4.4.5-1, Item 4.a after the reactor coolant gross specific activity exceeds 100/E-bar μCi/gm.

   BWR operating experience has shown that as fuel leakage increases, DOSE EQUIVALENT I-131 (DEI) approaches the TS limit much more rapidly than does the gross specific activity. The BWR design utilizes main condenser air ejectors to remove non-condensible gases from the reactor coolant. The non-condensible gases are then sampled, monitored, and processed by the Offgas Treatment System prior to release to the environment. The offgas pretreatment sample provides a more representative sample of the noble gases that would be released in the event of a main steam line failure outside containment than does the reactor coolant sample currently being taken from the Reactor Recirculation System. The offgas pretreatment monitor includes a setpoint which responds to release rates above a specified level which is established to ensure that untreated releases would not result in a whole body dose that exceeds a small fraction of the 10 CFR 100 limits.

   The intent of the requirement to limit specific activity in the reactor coolant is to ensure that the whole body and thyroid doses at the site boundary will not exceed a small fraction of the 10 CFR 100 limits (i.e., 10 percent of 25 rem and 300 rem, respectively) in the limiting event of a main steam line failure outside containment. To ensure that offsite thyroid doses do not exceed 30 rem, reactor coolant DEI is limited to less than or equal to 0.2 μCi/gm. Likewise, reactor coolant gross specific activity is limited by current Technical Specifications to less than or equal to 100/E-bar μCi/gm to ensure that offsite whole body doses do not exceed 2.5 rem. Reactor coolant gross

L.1 CHANGE

1. (continued)

specific activity is not an initiator of any accident evaluated in the UFSAR and therefore, deletion of CTS 3.4.5.b which limits reactor coolant gross specific activity to a value less than or equal to 100/E-bar μCi/gm will not result in an increase in the probability of an accident previously evaluated in the UFSAR.

CTS 3.11.2.2 and ITS 3.7.6, associated with radioactive effluents, requires that the gross gamma radioactivity rate of the noble gases measured at the Offgas System pretreatment monitor station be limited to less than or equal to 340,000 μCi/second. The current Bases for CTS 3.11.2.2 state that 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 10 CFR 100 limits in the event this effluent is inadvertently discharged without treatment directly to the environment.

The Offgas System, as required by CTS 3.11.2.2 and ITS 3.7.6, provides reasonable assurance the reactor coolant gross specific activity is maintained at a sufficiently low level to preclude offsite doses from exceeding a small fraction of the 10 CFR 100 limits in the event of a main steam line failure. Additional assurance that the offsite doses will not exceed a small fraction of the 10 CFR 100 limits is provided by increasing the frequency of sampling and analysis of the reactor coolant for DEI from at least once per 31 days to at least once per 7 days (proposed SR 3.4.8.1). Since the proposed change will ensure that the offsite doses resulting from a main steam line failure or an instrument line break will continue to be limited to a small fraction of the 10 CFR 100 limits, the proposed change will not involve a significant increase in the consequences of an accident previously evaluated.

2. Does the change create the possibility of a new or different kind of accident from any accident previously evaluated?

The proposed change does not involve a physical modification to the plant or to plant operation. The reactor coolant gross specific activity is a parameter that is monitored to prevent offsite doses from exceeding a small fraction (10%) of the 10 CFR 100 limits and support calculation of offsite doses in the event of a main steam line failure outside containment. As such, the reactor coolant specific activity is utilized to mitigate the radiological consequences of a main steam line failure and is not considered to be an initiator for any accident.

L.1 CHANGE

2. (continued)

Additionally, the Offgas System will provide an equal or better means for monitoring the reactor coolant gross specific activity than would the Reactor Recirculation System currently being used for this purpose. In the event of a main steam line break upstream of the condenser that would prevent use of the Offgas System to monitor reactor coolant gross specific activity, the existing sample points on the Reactor Recirculation System and RWCU System would continue to be available. Accordingly, deletion of the requirement to limit reactor coolant gross specific activity will not create the possibility of a new or different kind of accident from any previously evaluated.

3. Does this change involve a significant reduction in a margin of safety?

The intent of the requirement to limit specific activity in the reactor coolant is to ensure that the whole body and thyroid doses at the site boundary will not exceed a small fraction of the 10 CFR 100 limits (i.e., 10 percent of 25 rem and 300 rem, respectively) in the limiting event of a main steam line failure outside containment.

As stated above, CTS 3.11.2.2 associated with radioactive effluents requires that the gross gamma radioactivity of the noble gases measured at the Offgas system pretreatment monitor station be limited to less than or equal to 340,000 μCi/second. The current Bases for CTS 3.11.2.2 state that 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 10 CFR 100 limits in the event this effluent is inadvertently discharged without treatment directly to the environment.

The Offgas System, as required by CTS 3.11.2.2 and ITS 3.7.6, provides reasonable assurance the reactor coolant gross specific activity is maintained at a level sufficiently to preclude offsite doses from exceeding a small fraction of the 10 CFR 100 limits in the event of a main steam line failure. Therefore, CTS 3.4.5.b is redundant and places an unnecessary burden on the licensee without a commensurate increase in the margin of safety. Elimination of CTS 3.4.5.b will allow plant personnel to focus attention on efficient, safe operation of the plant without the distraction of an unnecessary Surveillance Requirement. Accordingly, the proposed change enhances operation of the plant without reducing the margin of safety associated with a main steam line failure outside of containment (i.e., offsite doses remain a small fraction of the 10 CFR 100 limits).

L.1 CHANGE

3. (continued)

Additional assurance that the offsite doses will not exceed a small fraction of the 10 CFR 100 limits is provided by increasing the frequency of sampling and analysis of the reactor coolant for DEI from at least once per 31 days to at least once per 7 days (proposed SR 3.4.8.1). Therefore, the proposed change does not result in a significant reduction in a margin of safety.

L.2 CHANGE

In accordance with the criteria set forth in 10 CFR 50.92, ComEd has evaluated this proposed Technical Specifications change and determined it does not represent a significant hazards consideration. The following is provided in support of this conclusion.

1. Does the change involve a significant increase in the probability or consequences of an accident previously evaluated?

   The proposed change would limit the CTS 3.4.5 Applicability for specific activity to those conditions that have potential impact on the consequences of an accident. The specific activity is not considered as an initiator of any previously evaluated accident. Therefore, the proposed change will not increase the probability of any accident previously evaluated. Specific activity is an assumption that must be met to limit the consequences of an accident. However, in MODE 4 there is no potential for leakage since the reactor is depressurized, and with the main steam lines isolated in MODES 2 and 3, there is no significant leakage path. Therefore, the proposed change will not increase the consequences of any accident previously evaluated.

2. Does the change create the possibility of a new or different kind of accident from any accident previously evaluated?

   The proposed change does not introduce a new mode of plant operation and does not involve physical modification to the plant. Therefore, it does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. Does this change involve a significant reduction in a margin of safety?

   This change does not involve a significant reduction in a margin of safety since the proposed conditions maintain Applicability of the appropriate limits for all conditions that represent potential to impact the consequences of any accident previously evaluated.

L.3 CHANGE

In accordance with the criteria set forth in 10 CFR 50.92, ComEd has evaluated this proposed Technical Specifications change and determined it does not represent a significant hazards consideration. The following is provided in support of this conclusion.

1. Does the change involve a significant increase in the probability or consequences of an accident previously evaluated?

   The proposed change would allow entry into the applicable conditions while depending on compliance with the ACTION. The specific activity is not considered as an initiator of any previously evaluated accident. Therefore, the proposed change will not increase the probability of any accident previously evaluated. Specific activity is an assumption that must be met to limit the consequences of an accident. However, operation has been determined to be acceptable for a short period of time with the limits not met. The consequences of an accident while operating during the proposed period of time are the same as those while operating under the constraints of the ACTION which has previously been determined acceptable. Therefore, the proposed change will not increase the consequences of any accident previously evaluated.

2. Does the change create the possibility of a new or different kind of accident from any accident previously evaluated?

   The proposed change does not introduce a new mode of plant operation and does not involve physical modification to the plant. Therefore it does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. Does this change involve a significant reduction in a margin of safety?

   This change does not involve a significant reduction in a margin of safety since the proposed period of time for operating beyond the limits has not changed.

NO SIGNIFICANT HAZARDS CONSIDERATION
ITS: 3.4.9 - RHR SHUTDOWN COOLING SYSTEM — HOT SHUTDOWN

L.1 CHANGE

In accordance with the criteria set forth in 10 CFR 50.92, ComEd has evaluated this proposed Technical Specifications change and determined it does not represent a significant hazards consideration. The following is provided in support of this conclusion.

1. Does the change involve a significant increase in the probability or consequences of an accident previously evaluated?

   The proposed change allows a recirculation pump to be in operation during MODE 3 as an acceptable method for assuring the necessary flow conditions, in lieu of operating a RHR shutdown cooling pump. This provides additional flexibility and diversity in assuring forced circulation is available for decay heat removal from the reactor core. Since additional diversity is afforded, a significant increase in the probability of a previously evaluated accident is not involved. The addition of diversity in providing forced circulation for shutdown cooling does not affect the consequences of a previously evaluated accident. The proposed change continues to ensure the decay heat removal function is satisfied.

2. Does the change create the possibility of a new or different kind of accident from any accident previously evaluated?

   The proposed change introduces no new mode of plant operation and does not involve physical modification to the plant. The proposed change does not provide for a new or different kind of accident in that a loss of shutdown cooling has previously been evaluated. This change provides diversity for providing forced circulation through the reactor core. Therefore, it does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. Does this change involve a significant reduction in a margin of safety?

   This change provides diversity in providing a forced circulation method for decay heat removal from the reactor core. A recirculation pump is capable of providing the necessary forced circulation through the core during shutdown for removal of decay heat. Therefore, no reduction in a margin of safety is involved.

NO SIGNIFICANT HAZARDS CONSIDERATION
ITS: 3.4.9 - RHR SHUTDOWN COOLING SYSTEM — HOT SHUTDOWN

L.2 CHANGE

In accordance with the criteria set forth in 10 CFR 50.92, ComEd has evaluated this proposed Technical Specifications change and determined it does not represent a significant hazards consideration. The following is provided in support of this conclusion.

1. Does the change involve a significant increase in the probability or consequences of an accident previously evaluated?

   The proposed change allows time to place the system in service after reaching the applicable conditions. Since the system can not physically be placed in service until the cut-in permissive pressure setpoint is reached, this change only allows the activity to take place without resorting to intentional noncompliance with the requirements. Since no actual change to the operation of the plant is involved, the proposed change will not increase the probability or consequences of any accident previously evaluated.

2. Does the change create the possibility of a new or different kind of accident from any accident previously evaluated?

   The proposed change introduces no new mode of plant operation and it does not involve physical modification to the plant. Therefore, it does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. Does this change involve a significant reduction in a margin of safety?

   This change does not involve a significant reduction in a margin of safety since the proposed change only allows time to conduct the necessary manipulations to place the required system in service.

L.3 CHANGE

In accordance with the criteria set forth in 10 CFR 50.92, ComEd has evaluated this proposed Technical Specifications change and determined it does not represent a significant hazards consideration. The following is provided in support of this conclusion.

1. Does the change involve a significant increase in the probability or consequences of an accident previously evaluated?

   The proposed change will allow the required RHR shutdown cooling loops to be inoperable for 2 hours for surveillance testing and both RHR shutdown cooling loops and recirculation pumps to be not in operation for 2 hours per 8 hour period. Currently, the 2 hour allowances are only applicable if one RHR shutdown cooling loop is Operable. While the UFSAR evaluates the loss of all RHR shutdown cooling, the event is not an assumed accident. In addition, the change still requires alternate methods for decay heat removal for each inoperable RHR shutdown cooling loop to be in place when both loops are inoperable when the 2 hour allowance of Note 1 is used, and still requires reactor coolant circulation when one RHR shutdown cooling loop is inoperable in accordance with Note 2. The alternate methods must each be fully capable of removing the decay heat load, thus the method is essentially equivalent to the RHR shutdown cooling loops in this respect. In addition, the current Technical Specifications allow use of the alternate methods in lieu of the RHR shutdown cooling loops for an unlimited amount of time. Therefore, this change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. Does the change create the possibility of a new or different kind of accident from any accident previously evaluated?

   The proposed change introduces no new mode of plant operation and it does not involve physical modification to the plant. Therefore, it does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. Does this change involve a significant reduction in a margin of safety?

   The proposed change will allow the required RHR shutdown cooling loops to be inoperable for 2 hours for surveillance testing and both RHR shutdown cooling loops and recirculation pumps to be not in operation for 2 hours per 8 hour period. Currently, the 2 hour allowances are only applicable if one RHR shutdown cooling loop is Operable. The change does not affect the requirement to have an alternate method capable of decay heat removal for each inoperable RHR shutdown cooling loop when both loops are inoperable and Note 1 is being used and the reactor coolant to be recirculating and one loop to be Operable when Note 2 is being used. Each alternate method must be fully capable of removing the decay heat load and circulating reactor

L.3 CHANGE

3. (continued)

coolant, thus the alternate methods are essentially equivalent to the RHR shutdown cooling loops in this respect. In addition, the current Technical Specifications allow use of the alternate methods in lieu of the RHR shutdown cooling loops for an unlimited amount of time. Therefore, this change does not involve a significant reduction in a margin of safety.

L.1 CHANGE

In accordance with the criteria set forth in 10 CFR 50.92, ComEd has evaluated this proposed Technical Specifications change and determined it does not represent a significant hazards consideration. The following is provided in support of this conclusion.

1. Does the change involve a significant increase in the probability or consequences of an accident previously evaluated?

   The proposed change allows a recirculation pump to be in operation during MODE 4 as an acceptable method for assuring the necessary flow conditions, in lieu of operating a RHR shutdown cooling pump. This provides additional flexibility and diversity in assuring forced circulation is available. Since additional diversity is afforded, a significant increase in the probability of a previously evaluated accident is not involved. The addition of diversity in providing forced circulation for shutdown cooling does not affect the consequences of a previously evaluated accident. The proposed change continues to ensure the forced reactor coolant circulation function is satisfied.

2. Does the change create the possibility of a new or different kind of accident from any accident previously evaluated?

   The proposed change introduces no new mode of plant operation and it does not involve physical modification to the plant. The proposed change does not provide for a new or different kind of accident in that a loss of shutdown cooling has previously been evaluated. This change provides diversity for providing forced circulation through the reactor core. Therefore, it does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. Does this change involve a significant reduction in a margin of safety?

   This change provides diversity in providing a forced circulation method for decay heat removal from the reactor core. A recirculation pump is capable of providing the necessary forced circulation through the core during shutdown. Therefore, no reduction in a margin of safety is involved.

L.2 CHANGE

In accordance with the criteria set forth in 10 CFR 50.92, ComEd has evaluated this proposed Technical Specifications change and determined it does not represent a significant hazards consideration. The following is provided in support of this conclusion.

1. Does the change involve a significant increase in the probability or consequences of an accident previously evaluated?

   The proposed change will allow the required RHR shutdown cooling loops to be inoperable for 2 hours for surveillance testing and both RHR shutdown cooling loops and recirculation pumps to be not in operation for 2 hours per 8 hour period. Currently, the 2 hour allowances are only applicable if one RHR shutdown cooling loop is Operable. While the UFSAR evaluates the loss of all RHR shutdown cooling, the event is not an assumed accident. In addition, the change still requires alternate methods for decay heat removal for each inoperable RHR shutdown cooling loop to be in place and reactor coolant to be recirculating when the 2 hour allowance is used. The alternate methods must each be fully capable of removing the decay heat load, thus the method is essentially equivalent to the RHR shutdown cooling loops in this respect. In addition, the current Technical Specifications allow use of the alternate methods in lieu of the RHR shutdown cooling loops for an unlimited amount of time. Therefore, this change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. Does the change create the possibility of a new or different kind of accident from any accident previously evaluated?

   The proposed change introduces no new mode of plant operation and it does not involve physical modification to the plant. Therefore, it does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. Does this change involve a significant reduction in a margin of safety?

   The proposed change will allow the required RHR shutdown cooling loops to be inoperable for 2 hours for surveillance testing and both RHR shutdown cooling loops and recirculation pumps to be not in operation for 2 hours per 8 hour period. Currently, the 2 hour allowances are only applicable if one RHR shutdown cooling loop is Operable. The change does not affect the requirement to have an alternate method capable of decay heat removal for each inoperable RHR shutdown cooling loop and the reactor coolant to be recirculating. Each alternate method must be fully capable of removing the decay heat load and circulating reactor coolant, thus the alternate methods are essentially equivalent to the RHR shutdown cooling loops in this

L.2 CHANGE

3. (continued)

respect. In addition, the current Technical Specifications allow use of the alternate methods in lieu of the RHR shutdown cooling loops for an unlimited amount of time. Therefore, this change does not involve a significant reduction in a margin of safety.

L.3 CHANGE

In accordance with the criteria set forth in 10 CFR 50.92, ComEd has evaluated this proposed Technical Specifications change and determined it does not represent a significant hazards consideration. The following is provided in support of this conclusion.

1. Does the change involve a significant increase in the probability or consequences of an accident previously evaluated?

   The proposed change will allow both RHR shutdown cooling subsystems to be inoperable during hydrostatic testing. Currently, the allowance is that the RHR shutdown cooling mode loop may be removed from operation during hydrostatic testing. The RHR Shutdown Cooling System is in fact inoperable during hydrostatic testing since the system is not capable of circulating reactor coolant. The RHR Shutdown Cooling System is automatically isolated above the RHR cut-in permissive pressure. The isolation is necessary since the RHR Shutdown Cooling System is not designed to operate at the Reactor Coolant System pressure achieved during hydrostatic testing. The proposed allowance is acceptable since hydrostatic testing is performed after each refueling outage (prior to reactor criticality) when decay heat levels are low, adequate reactor coolant circulation will be achieved by operation of a reactor recirculation pump, and other systems are available to control reactor coolant temperature. Therefore, this change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. Does the change create the possibility of a new or different kind of accident from any accident previously evaluated?

   The proposed change introduces no new mode of plant operation and it does not involve physical modification to the plant. Therefore, it does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. Does this change involve a significant reduction in a margin of safety?

   The proposed change will allow both RHR shutdown cooling subsystems to be inoperable during hydrostatic testing. Currently, the allowance is that the RHR shutdown cooling mode loop may be removed from operation during hydrostatic testing. The RHR Shutdown Cooling System is in fact inoperable during hydrostatic testing since the system is not capable of circulating reactor coolant. The RHR Shutdown Cooling System is automatically isolated above the RHR cut-in permissive pressure. This isolation is necessary since the RHR Shutdown Cooling System is not designed to operate at the Reactor Coolant System pressure achieved during hydrostatic testing. Hydrostatic testing is performed after each refueling outage (prior to reactor

L.3 CHANGE

3. (continued)

criticality) when decay heat levels are low. Adequate reactor coolant circulation will be achieved by operation of a reactor recirculation pump and other systems are available to control reactor coolant temperature. Therefore, the allowance that the RHR Shutdown Cooling System may be inoperable during hydrostatic testing is considered acceptable, and this change does not involve a significant reduction in a margin of safety.

NO SIGNIFICANT HAZARDS CONSIDERATION
ITS: 3.4.11 - RCS PRESSURE AND TEMPERATURE (P/T) LIMITS

There were no plant specific less restrictive changes identified for this Specification.

NO SIGNIFICANT HAZARDS CONSIDERATION
ITS: 3.4.12 - REACTOR STEAM DOME PRESSURE

L.1 CHANGE

In accordance with the criteria set forth in 10 CFR 50.92, ComEd has evaluated this proposed Technical Specifications change and determined it does not represent a significant hazards consideration. The following is provided in support of this conclusion.

1. Does the change involve a significant increase in the probability or consequences of an accident previously evaluated?

   This change would allow the reactor steam dome pressure limit to be raised an infinitesimally small amount to be equal to 1020 psig and still be within the limit. The reactor steam dome pressure is not considered an initiator of any previously analyzed accident. Therefore, this change does not significantly increase the probability of such accidents. The proposed change would allow continued operation at exactly 1020 psig. However, the consequences of an event that may occur at 1020 psig would not be any different than an event that occurs at slightly less than 1020 psig since the safety analyses assume the reactor steam dome pressure at the start of the accident is equal to 1020 psig, and the analyses show that the ASME limits are not exceeded during the overpressure transient. Therefore, this change does not significantly increase the consequences of any previously analyzed accident.

2. Does the change create the possibility of a new or different kind of accident from any accident previously evaluated?

   The proposed change does not involve any design changes, plant modifications, or changes in plant operation. Therefore, the proposed change does not create the possibility of a new or different kind of accident from any previously evaluated.

3. Does this change involve a significant reduction in a margin of safety?

   This change would allow the reactor steam dome pressure limit to be raised an infinitesimally small amount to be equal to 1020 psig and still be within the limit. The safety analyses assume the reactor steam dome pressure at the start of the accident is equal to 1020 psig, and the analyses show that the ASME limits are not exceeded during the overpressure transient. Therefore, the change does not involve a significant reduction in the margin of safety.

NO SIGNIFICANT HAZARDS CONSIDERATION
CTS: 3/4.4.8 - STRUCTURAL INTEGRITY

There were no plant specific less restrictive changes identified for this Specification.

ENVIRONMENTAL ASSESSMENT
ITS: SECTION 3.4 - REACTOR COOLANT SYSTEM

In accordance with the criteria set forth in 10 CFR 50.21, ComEd has evaluated this proposed Technical Specification change for identification of licensing and regulatory actions requiring environmental assessment, determined it meets the criteria for a categorical exclusion set forth in 10 CFR 51.22(c)(9) and as such, has determined that no irreversible consequences exist in accordance with 10 CFR 50.92(b). This determination is based on the fact that this change is being proposed as an amendment to a license issued pursuant to 10 CFR which changes a requirement with respect to installation or use of a facility component located within the restricted area, as defined in 10 CFR 20, or which changes an inspection or a surveillance requirement, and the amendment meets the following specific criteria:

4. The amendment involves no significant hazards consideration.

   As demonstrated in the No Significant Hazards Consideration, this proposed amendment does not involve any significant hazards consideration.

5. There is no significant change in the type or significant increase in the amounts of any effluents that may be released offsite.

   The proposed change will not result in changes in the operation or configuration of the facility. There will be no change in the level of controls or methodology used for processing of radioactive effluents or handling of solid radioactive waste, nor will the proposal result in any change in the normal radiation levels within the plant. Therefore, there will be no change in the types or significant increase in the amounts of any effluents released offsite resulting from this change.

6. There is no significant increase in individual or cumulative occupational radiation exposure.

   The proposed change will not result in changes in the operation or configuration of the facility which impact radiation exposure. There will be no change in the level of controls or methodology used for processing of radioactive effluents or handling of solid radioactive waste, nor will the proposal result in any change in the normal radiation levels within the plant. Therefore, there will be no increase in individual or cumulative occupational radiation exposure resulting from this change.

Therefore, based upon the above evaluation, ComEd has concluded that no irreversible consequences exist with the proposed change.

3.5 EMERGENCY CORE COOLING SYSTEMS (ECCS) AND REACTOR CORE ISOLATION COOLING (RCIC) SYSTEM

3.5.1 ECCS – Operating

LCO 3.5.1 Each ECCS injection/spray subsystem and the Automatic Depressurization System (ADS) function of six safety/relief valves shall be OPERABLE.

--------------------------NOTE--------------------------
Low pressure coolant injection (LPCI) subsystems may be considered OPERABLE during alignment and operation for decay heat removal with reactor vessel pressure less than the residual heat removal cut-in permissive pressure in MODE 3, if capable of being manually realigned and not otherwise inoperable.

APPLICABILITY: MODE 1,
MODES 2 and 3, except ADS valves are not required to be OPERABLE with reactor steam dome pressure ≤ 150 psig.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One low pressure ECCS injection/spray subsystem inoperable.	A.1 Restore low pressure ECCS injection/spray subsystem to OPERABLE status.	7 days

(continued)

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
B. High Pressure Core Spray (HPCS) System inoperable.	B.1	Verify by administrative means RCIC System is OPERABLE when RCIC is required to be OPERABLE.	Immediately
	<u>AND</u>		
	B.2	Restore HPCS System to OPERABLE status.	14 days
C. Two low pressure ECCS injection/spray subsystems inoperable.	C.1	Restore one low pressure ECCS injection/spray subsystem to OPERABLE status.	72 hours
D. ADS accumulator backup compressed gas system bottle pressure < 500 psig.	D.1	Restore ADS accumulator backup compressed gas system bottle pressure $\geq$ 500 psig.	72 hours
	<u>OR</u>		
	D.2	Declare associated ADS valves inoperable.	72 hours
E. Required Action and associated Completion Time of Condition A, B, or C not met.	E.1	Be in MODE 3.	12 hours
	<u>AND</u>		
	E.2	Be in MODE 4.	36 hours

(continued)

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
F. One required ADS valve inoperable.	F.1 Restore required ADS valve to OPERABLE status.	14 days
G. Required Action and associated Completion Time of Condition F not met. OR Two or more required ADS valves inoperable.	G.1 Be in MODE 3. AND G.2 Reduce reactor steam dome pressure to $\leq$ 150 psig.	12 hours 36 hours
H. HPCS and one or more low pressure ECCS injection/spray subsystems inoperable. OR Three or more ECCS injection/spray subsystems inoperable. OR One or more ECCS injection/spray subsystems and one or more required ADS valves inoperable.	H.1 Enter LCO 3.0.3.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.5.1.1	Verify, for each ECCS injection/spray subsystem, the piping is filled with water from the pump discharge valve to the injection valve.	31 days
SR 3.5.1.2	Verify each ECCS injection/spray subsystem manual, power operated, and automatic valve in the flow path, that is not locked, sealed, or otherwise secured in position, is in the correct position.	31 days
SR 3.5.1.3	Verify ADS accumulator supply header pressure is ≥ 150 psig.	31 days
SR 3.5.1.4	Verify ADS accumulator backup compressed gas system bottle pressure is ≥ 500 psig.	31 days
SR 3.5.1.5	Verify each ECCS pump develops the specified flow rate against the specified test line pressure. SYSTEM / FLOW RATE / TEST LINE PRESSURE LPCS / ≥ 6350 gpm / ≥ 290 psig LPCI / ≥ 7200 gpm / ≥ 130 psig HPCS (Unit 1) / ≥ 6250 gpm / ≥ 370 psig HPCS (Unit 2) / ≥ 6200 gpm / ≥ 330 psig	In accordance with the Inservice Testing Program

(continued)

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.5.1.6	------------------NOTE------------------ Vessel injection/spray may be excluded. -- Verify each ECCS injection/spray subsystem actuates on an actual or simulated automatic initiation signal.	24 months
SR 3.5.1.7	------------------NOTE------------------ Valve actuation may be excluded. -- Verify the ADS actuates on an actual or simulated automatic initiation signal.	24 months
SR 3.5.1.8	Verify each required ADS valve opens when manually actuated.	24 months on a STAGGERED TEST BASIS for each valve solenoid
SR 3.5.1.9	------------------NOTE------------------ Instrumentation response time may be assumed to be the design instrumentation response time. -- Verify the ECCS RESPONSE TIME for each ECCS injection/spray subsystem is within limits.	24 months

3.5 EMERGENCY CORE COOLING SYSTEMS (ECCS) AND REACTOR CORE ISOLATION COOLING (RCIC) SYSTEM

3.5.2 ECCS — Shutdown

LCO 3.5.2 Two ECCS injection/spray subsystems shall be OPERABLE.

-------------------------NOTE-------------------------
One low pressure coolant injection (LPCI) subsystem may be considered OPERABLE during alignment and operation for decay heat removal, if capable of being manually realigned and not otherwise inoperable.

APPLICABILITY: MODE 4,
MODE 5 except with the spent fuel storage pool gates removed and water level $\geq$ 22 ft over the top of the reactor pressure vessel flange.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One required ECCS injection/spray subsystem inoperable.	A.1 Restore required ECCS injection/spray subsystem to OPERABLE status.	4 hours
B. Required Action and associated Completion Time of Condition A not met.	B.1 Initiate action to suspend operations with a potential for draining the reactor vessel (OPDRVs).	Immediately

(continued)

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
C. Two required ECCS injection/spray subsystems inoperable.	C.1	Initiate action to suspend OPDRVs.	Immediately
	AND		
	C.2	Restore one required ECCS injection/spray subsystem to OPERABLE status.	4 hours
D. Required Action C.2 and associated Completion Time not met.	D.1	Initiate action to restore secondary containment to OPERABLE status.	Immediately
	AND		
	D.2	Initiate action to restore one standby gas treatment subsystem to OPERABLE status.	Immediately
	AND		
	D.3	Initiate action to restore isolation capability in each required secondary containment penetration flow path not isolated.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.5.2.1	Verify, for each required low pressure ECCS injection/spray subsystem, the suppression pool water level is $\geq$ -12 ft 7 in.	12 hours
SR 3.5.2.2	Verify, for the required High Pressure Core Spray (HPCS) System, the suppression pool water level is $\geq$ -12 ft 7 in.	12 hours
SR 3.5.2.3	Verify, for each required ECCS injection/ spray subsystem, the piping is filled with water from the pump discharge valve to the injection valve.	31 days
SR 3.5.2.4	Verify each required ECCS injection/spray subsystem manual, power operated, and automatic valve in the flow path, that is not locked, sealed, or otherwise secured in position, is in the correct position.	31 days
SR 3.5.2.5	Verify each required ECCS pump develops the specified flow rate against the specified test line pressure. SYSTEM / FLOW RATE / TEST LINE PRESSURE LPCS / $\geq$ 6350 gpm / $\geq$ 290 psig LPCI / $\geq$ 7200 gpm / $\geq$ 130 psig HPCS (Unit 1) / $\geq$ 6250 gpm / $\geq$ 370 psig HPCS (Unit 2) / $\geq$ 6200 gpm / $\geq$ 330 psig	In accordance with the Inservice Testing Program

(continued)

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.5.2.6	-------------------NOTE------------------- Vessel injection/spray may be excluded. -- Verify each required ECCS injection/spray subsystem actuates on an actual or simulated automatic initiation signal.	24 months
SR 3.5.2.7	-------------------NOTE------------------- Instrumentation response time may be assumed to be the design instrumentation response time. -- Verify the ECCS RESPONSE TIME for each required ECCS injection/spray subsystem is within limits.	24 months

3.5 EMERGENCY CORE COOLING SYSTEMS (ECCS) AND REACTOR CORE ISOLATION COOLING (RCIC) SYSTEM

3.5.3 RCIC System

LCO 3.5.3 The RCIC System shall be OPERABLE.

APPLICABILITY: MODE 1,
MODES 2 and 3 with reactor steam dome pressure > 150 psig.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. RCIC System inoperable.	A.1 Verify by administrative means High Pressure Core Spray System is OPERABLE. <u>AND</u> A.2 Restore RCIC System to OPERABLE status.	Immediately 14 days
B. Required Action and associated Completion Time not met.	B.1 Be in MODE 3. <u>AND</u> B.2 Reduce reactor steam dome pressure to $\leq$ 150 psig.	12 hours 36 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.5.3.1	Verify the RCIC System piping is filled with water from the pump discharge valve to the injection valve.	31 days
SR 3.5.3.2	Verify each RCIC System manual, power operated, and automatic valve in the flow path, that is not locked, sealed, or otherwise secured in position, is in the correct position.	31 days
SR 3.5.3.3	-------------------NOTE------------------- Not required to be performed until 12 hours after reactor steam pressure and flow are adequate to perform the test. -- Verify, with reactor pressure ≤ 1020 psig and ≥ 920 psig, the RCIC pump can develop a flow rate ≥ 600 gpm against a system head corresponding to reactor pressure.	92 days
SR 3.5.3.4	-------------------NOTE------------------- Not required to be performed until 12 hours after reactor steam pressure and flow are adequate to perform the test. -- Verify, with reactor pressure ≤ 165 psig, the RCIC pump can develop a flow rate ≥ 600 gpm against a system head corresponding to reactor pressure.	24 months

(continued)

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.5.3.5	-------------------NOTE------------------- Vessel injection may be excluded. -- Verify the RCIC System actuates on an actual or simulated automatic initiation signal.	24 months

B 3.5 EMERGENCY CORE COOLING SYSTEMS (ECCS) AND REACTOR CORE ISOLATION COOLING (RCIC) SYSTEM

B 3.5.1 ECCS — Operating

BASES

BACKGROUND

The ECCS is designed, in conjunction with the primary and secondary containment, to limit the release of radioactive materials to the environment following a loss of coolant accident (LOCA). The ECCS uses two independent methods (flooding and spraying) to cool the core during a LOCA. The ECCS network is composed of the High Pressure Core Spray (HPCS) System, the Low Pressure Core Spray (LPCS) System, and the low pressure coolant injection (LPCI) mode of the Residual Heat Removal (RHR) System. The ECCS also consists of the Automatic Depressurization System (ADS). The suppression pool provides the required source of water for the ECCS.

On receipt of an initiation signal, ECCS pumps automatically start; the system aligns, and the pumps inject water, taken from the suppression pool, into the Reactor Coolant System (RCS) as RCS pressure is overcome by the discharge pressure of the ECCS pumps. Although the system is initiated, ADS action is delayed, allowing the operator to interrupt the timed sequence if the system is not needed. The HPCS pump discharge pressure almost immediately exceeds that of the RCS, and the pump injects coolant into the spray sparger above the core. If the break is small, HPCS will maintain coolant inventory, as well as vessel level, while the RCS is still pressurized. If HPCS fails, it is backed up by ADS in combination with LPCI and LPCS. In this event, the ADS timed sequence would be allowed to time out and open the selected safety/relief valves (S/RVs), depressurizing the RCS and allowing the LPCI and LPCS to overcome RCS pressure and inject coolant into the vessel. If the break is large, RCS pressure initially drops rapidly, and the LPCI and LPCS systems cool the core.

Water from the break returns to the suppression pool where it is used again and again. Water in the suppression pool is circulated through a heat exchanger cooled by the Residual Heat Removal Service Water (RHRSW) System. Depending on the location and size of the break, portions of

(continued)

BASES

BACKGROUND (continued)

the ECCS may be ineffective; however, the overall design is effective in cooling the core regardless of the size or location of the piping break.

All ECCS subsystems are designed to ensure that no single active component failure will prevent automatic initiation and successful operation of the minimum required ECCS subsystems.

The LPCS System (Ref. 1) consists of a motor driven pump, a spray sparger above the core, piping, and valves to transfer water from the suppression pool to the sparger. The LPCS System is designed to provide cooling to the reactor core when the reactor pressure is low. Upon receipt of an initiation signal, the LPCS pump is automatically started when AC power is available. When the RPV pressure drops sufficiently, LPCS flow to the RPV begins. A full flow test line is provided to route water to the suppression pool to allow testing of the LPCS System without spraying water into the RPV.

LPCI is an independent operating mode of the RHR System. There are three LPCI subsystems. Each LPCI subsystem (Ref. 2) consists of a motor driven pump, piping, and valves to transfer water from the suppression pool to the core. Each LPCI subsystem has its own suction and discharge piping and separate vessel nozzle that connects with the core shroud through internal piping. The LPCI subsystems are designed to provide core cooling at low RPV pressure. Upon receipt of an initiation signal, each LPCI pump is automatically started. (If AC power is supplied by the diesel generators, C pump starts immediately when AC power is available and A and B pumps approximately 5 seconds after AC power is available). When the RPV pressure drops sufficiently, LPCI flow to the RPV begins. RHR System valves in the LPCI flow path are automatically positioned to ensure the proper flow path for water from the suppression pool to inject into the core. A full flow test line is provided to route water to the suppression pool to allow testing of each LPCI pump without injecting water into the RPV.

The HPCS System (Ref. 3) consists of a single motor driven pump, a spray sparger above the core, and piping and valves

(continued)

BASES

BACKGROUND (continued)

to transfer water from the suppression pool to the sparger. The HPCS System is designed to provide core cooling over a wide range of RPV pressures (0 psid to 1200 psid, vessel to suction source). Upon receipt of an initiation signal, the HPCS pump automatically starts (when AC power is available) and valves in the flow path begin to open. Since the HPCS System is designed to operate over the full range of expected RPV pressures, HPCS flow begins as soon as the necessary valves are open. A full flow test line is provided to route water to the suppression pool to allow testing of the HPCS System during normal operation without spraying water into the RPV.

The ECCS pumps are provided with minimum flow bypass lines, which discharge to the suppression pool. The valves in these lines automatically open to prevent pump damage due to overheating when other discharge line valves are closed or RPV pressure is greater than the LPCS or LPCI pump discharge pressures following system initiation. To ensure rapid delivery of water to the RPV and to minimize water hammer effects, the ECCS discharge line "keep fill" systems are designed to maintain all pump discharge lines filled with water.

The ADS (Ref. 4) consists of 7 of the 18 S/RVs for Unit 1 and 7 of the 13 S/RVs for Unit 2. It is designed to provide depressurization of the primary system during a small break LOCA if HPCS fails or is unable to maintain required water level in the RPV. ADS operation reduces the RPV pressure to within the operating pressure range of the low pressure ECCS subsystems (LPCS and LPCI), so that these subsystems can provide core cooling.

The Drywell Pneumatic System discharges from the air receiver (or nitrogen receiver when the primary containment is inerted) and after filtration is divided into two supply headers, one of which supplies all the ADS accumulators with approximately 175 psig air (or nitrogen). There is a check valve between each ADS accumulator and the supply. Drywell Pneumatic System low header pressure and high ADS pressure are alarmed in the control room.

The accumulators for the ADS valves are normally maintained by the Drywell Pneumatic System compressors. There are two full-capacity compressors which cycle as needed to maintain

(continued)

BASES

BACKGROUND (continued)

pressure in the drywell pneumatic receiver tank. Nitrogen bottle banks provide a backup source to maintain the ADS accumulators charged following isolation of the normal pneumatic supply. Each ADS accumulator is provided with a pressure switch to detect low pressure (< 150 psig). These pressure switches are provided with alarms in the control room. A control room alarm is also annunciated for low pressure in the ADS nitrogen bottle banks supply headers.

APPLICABLE SAFETY ANALYSES

The ECCS performance is evaluated for the entire spectrum of break sizes for a postulated LOCA. The accidents for which ECCS operation is required are presented in References 5, 6, and 7. The required analyses and assumptions are defined in 10 CFR 50 (Ref. 8), and the results of these analyses are described in Reference 9.

This LCO helps to ensure that the following acceptance criteria for the ECCS, established by 10 CFR 50.46 (Ref. 10), will be met following a LOCA assuming the worst case single active component failure in the ECCS:

a. Maximum fuel element cladding temperature is ≤ 2200°F;

b. Maximum cladding oxidation is ≤ 0.17 times the total cladding thickness before oxidation;

c. Maximum hydrogen generation from zirconium water reaction is ≤ 0.01 times the hypothetical amount that would be generated if all of the metal in the cladding surrounding the fuel, excluding the cladding surrounding the plenum volume, were to react;

d. The core is maintained in a coolable geometry; and

e. Adequate long term cooling capability is maintained.

The limiting single failures are discussed in Reference 11. For the LOCA evaluation model which covers the entire spectrum of break sizes (large breaks to small breaks), failure of the HPCS ECCS subsystem in Division 3 due to failure of its associated diesel generator is, in general, the most severe failure. The remaining OPERABLE ECCS subsystems, which include one spray subsystem, provide the

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

capability to adequately cool the core, under near-term and long-term conditions, and prevent excessive fuel damage. For all LOCA analyses, only six ADS valves are assumed to function. An additional analysis has been performed which assumes five ADS valves function, however in this analysis all low pressure and high pressure ECCS subsystems are also assumed to be available.

The ECCS satisfy Criterion 3 of 10 CFR 50.36(c)(2)(ii).

LCO

Each ECCS injection/spray subsystem and six ADS valves are required to be OPERABLE. The ECCS injection/spray subsystems are defined as the three LPCI subsystems, the LPCS System, and the HPCS System. The low pressure ECCS injection/spray subsystems are defined as the LPCS System and the three LPCI subsystems.

With less than the required number of ECCS subsystems OPERABLE during a limiting design basis LOCA concurrent with the worst case single failure, the limits specified in 10 CFR 50.46 (Ref. 10) could potentially be exceeded. All ECCS subsystems must therefore be OPERABLE to satisfy the single failure criterion required by 10 CFR 50.46 (Ref. 10).

As noted, LPCI subsystems 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. Alignment and operation for decay heat removal includes when the required RHR pump is not operating or when the system is realigned from or to the RHR shutdown cooling mode. This allowance is necessary since the RHR System may be required to operate in the shutdown cooling mode to remove decay heat and sensible heat from the reactor. At these low pressures and decay heat levels, a reduced complement of ECCS subsystems should provide the required core cooling, thereby allowing operation of RHR shutdown cooling when necessary.

APPLICABILITY

All ECCS subsystems are required to be OPERABLE during MODES 1, 2, and 3 when there is considerable energy in the reactor core and core cooling would be required to prevent fuel damage in the event of a break in the primary system piping. In MODES 2 and 3, the ADS function is not required

(continued)

BASES

APPLICABILITY (continued)

when pressure is ≤ 150 psig because the low pressure ECCS subsystems (LPCS and LPCI) are capable of providing flow into the RPV below this pressure. ECCS requirements for MODES 4 and 5 are specified in LCO 3.5.2, "ECCS — Shutdown.

ACTIONS

A.1

If any one low pressure ECCS injection/spray subsystem is inoperable, the inoperable subsystem must be restored to OPERABLE status within 7 days. In this Condition, the remaining OPERABLE subsystems provide adequate core cooling during a LOCA. However, overall ECCS reliability is reduced because a single failure in one of the remaining OPERABLE subsystems concurrent with a LOCA may result in the ECCS not being able to perform its intended safety function. The 7 day Completion Time is based on a reliability study (Ref. 12) that evaluated the impact on ECCS availability by assuming that various components and subsystems were taken out of service. The results were used to calculate the average availability of ECCS equipment needed to mitigate the consequences of a LOCA as a function of allowed outage times (i.e., Completion Times).

B.1 and B.2

If the HPCS System is inoperable, and the RCIC System is immediately verified to be OPERABLE (when RCIC is required to be OPERABLE), the HPCS System must be restored to OPERABLE status within 14 days. In this Condition, adequate core cooling is ensured by the OPERABILITY of the redundant and diverse low pressure ECCS injection/spray subsystems in conjunction with the ADS. Also, the RCIC System will automatically provide makeup water at most reactor operating pressures. Immediate verification of RCIC OPERABILITY is therefore required when HPCS is inoperable and RCIC is required to be OPERABLE. This may be performed by an administrative check, by examining logs or other information, to determine if RCIC is out of service for maintenance or other reasons. It is not necessary to perform the Surveillances needed to demonstrate the OPERABILITY of the RCIC System. However, if the OPERABILITY

(continued)

BASES

ACTIONS

B.1 and B.2 (continued)

of the RCIC System cannot be immediately verified and RCIC is required to be OPERABLE, Condition D must be entered. If a single active component fails concurrent with a design basis LOCA, there is a potential, depending on the specific failure, that the minimum required ECCS equipment will not be available. A 14 day Completion Time is based on the results of a reliability study (Ref. 12) and has been found to be acceptable through operating experience.

C.1

With two ECCS injection subsystems inoperable or one ECCS injection and the low pressure ECCS spray subsystem (LPCS) inoperable, at least one ECCS injection/spray subsystem must be restored to OPERABLE status within 72 hours. In this Condition, the remaining OPERABLE subsystems provide adequate core cooling during a LOCA. However, overall ECCS reliability is reduced in this Condition because a single failure in one of the remaining OPERABLE subsystems concurrent with a design basis LOCA may result in the ECCS not being able to perform its intended safety function. Since the ECCS availability is reduced relative to Condition A, a more restrictive Completion Time is imposed. The 72 hour Completion Time is based on a reliability study, as provided in Reference 12.

D.1 and D.2

With the ADS accumulator backup compressed gas system bottle pressure less than the specified limit, bottle pressure must be restored within 72 hours, or the associated ADS valves must be declared inoperable. In this condition, the remaining Drywell Pneumatic System and ADS accumulators are sufficient to ensure ADS valve operation. However, overall ECCS reliability is reduced in this condition because with insufficient bottle bank pressure, the capability of ADS valves to operate for long periods of time following an accident (without the Drywell Pneumatic System) is reduced. Each ADS valve is equipped with an individual accumulator of sufficient capacity to operate the valves in the event of a loss of air supply. The 72 hour Completion Time is based on a reliability study, as provided in Reference 12.

(continued)

BASES

ACTIONS (continued)

E.1 and E.2

If any Required Action and associated Completion Time of Condition A, B, or C are not met, the plant must be brought to a MODE in which the LCO does not apply. To achieve this status, the plant must be brought to at least MODE 3 within 12 hours and to MODE 4 within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

F.1

The LCO requires six ADS valves to be OPERABLE to provide the ADS function. Reference 11 contains the results of an evaluation of the effect of one required ADS valve being out of service. Per this evaluation, operation of only five ADS valves will provide the required depressurization. However, overall reliability of the ADS is reduced because a single failure in the OPERABLE ADS valves could result in a reduction in depressurization capability. Therefore, operation is only allowed for a limited time. The 14 day Completion Time is based on a reliability study (Ref. 12) and has been found to be acceptable through operating experience.

G.1 and G.2

If any Required Action and associated Completion Time of Condition F is not met or if two or more required ADS valves are inoperable, the plant must be brought to a condition in which the LCO does not apply. To achieve this status, the plant must be brought to at least MODE 3 within 12 hours and reactor steam dome pressure reduced to $\leq$ 150 psig within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

(continued)

BASES

ACTIONS (continued)

H.1

When multiple ECCS subsystems are inoperable, as stated in Condition H, the plant is in a condition outside of the design basis. Therefore, LCO 3.0.3 must be entered immediately.

SURVEILLANCE REQUIREMENTS

SR 3.5.1.1

The flow path piping has the potential to develop voids and pockets of entrained air. Maintaining the pump discharge lines of the HPCS System, LPCS System, and LPCI subsystems full of water ensures that the systems will perform properly, injecting their full capacity into the RCS upon demand. This will also prevent a water hammer following an ECCS initiation signal. One acceptable method of ensuring the lines are full is to vent at the high points. The 31 day Frequency is based on operating experience, on the procedural controls governing system operation, and on the gradual nature of void buildup in the ECCS piping.

SR 3.5.1.2

Verifying the correct alignment for manual, power operated, and automatic valves in the ECCS flow paths provides assurance that the proper flow paths will exist for ECCS operation. This SR does not apply to valves that are locked, sealed, or otherwise secured in position since these valves were verified to be in the correct position prior to locking, sealing, or securing. A valve that receives an initiation signal is allowed to be in a nonaccident position provided the valve will automatically reposition in the proper stroke time. This SR does not require any testing or valve manipulation; rather, it involves verification that those valves potentially capable of being mispositioned are in the correct position. This SR does not apply to valves that cannot be inadvertently misaligned, such as check valves.

The 31 day Frequency of this SR was derived from the Inservice Testing Program requirements for performing valve testing at least once every 92 days. The Frequency of

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.5.1.2 (continued)

31 days is further justified because the valves are operated under procedural control and because improper valve alignment would only affect a single subsystem. This Frequency has been shown to be acceptable through operating experience.

SR 3.5.1.3

Verification every 31 days that ADS accumulator supply header pressure is ≥ 150 psig assures adequate pneumatic pressure for reliable ADS operation. The accumulator on each ADS valve provides pneumatic pressure for valve actuation. The ADS valve accumulators are sized to provide two cycles of the ADS valves upon loss of the nitrogen supply (Ref. 13). The ECCS safety analysis assumes only one actuation to achieve the depressurization required for operation of the low pressure ECCS. The accumulator supply header pressure verification may be accomplished by monitoring control room alarms. The 31 day Frequency takes into consideration alarms for low pneumatic pressure.

SR 3.5.1.4

Verification every 31 days that ADS accumulator backup compressed gas system bottle pressure is ≥ 500 psig assures availability of an adequate backup pneumatic supply to the ADS accumulators following a loss of the drywell pneumatic supply. The 31 day frequency is adequate because each ADS bottle bank is monitored by a low pressure alarm. Also, unless the normal drywell pneumatic supply is lost, the only expected losses from the bottles are due to leakage, which is minimal.

SR 3.5.1.5

The performance requirements of the ECCS pumps are determined through application of the 10 CFR 50, Appendix K, criteria (Ref. 8). This periodic Surveillance is performed (in accordance with the ASME Code, Section XI, requirements

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BASES

SURVEILLANCE REQUIREMENTS

SR 3.5.1.5 (continued)

for the ECCS pumps) to verify that the ECCS pumps will develop the flow rates required by the respective analyses. The ECCS pump flow rates ensure that adequate core cooling is provided to satisfy the acceptance criteria of 10 CFR 50.46 (Ref. 10).

The pump flow rates are verified against a test line pressure that was determined during preoperational testing to be equivalent to the RPV pressure expected during a LOCA. Under these conditions, the total system pump outlet pressure is adequate to overcome the elevation head pressure between the pump suction and the vessel discharge, the piping friction losses, and RPV pressure present during LOCAs. The Frequency for this Surveillance is in accordance with the Inservice Testing Program requirements.

SR 3.5.1.6

The ECCS subsystems are required to actuate automatically to perform their design functions. This Surveillance verifies that, with a required system initiation signal (actual or simulated), the automatic initiation logic of HPCS, LPCS, and LPCI will cause the systems or subsystems to operate as designed, including actuation of the system throughout its emergency operating sequence, automatic pump startup, and actuation of all automatic valves to their required position. This Surveillance also ensures that the HPCS System injection valve will automatically reopen on an RPV low water level (Level 2) signal received subsequent to an RPV high water level (Level 8) injection valve closure signal. The LOGIC SYSTEM FUNCTIONAL TEST performed in LCO 3.3.5.1 overlaps this Surveillance to provide complete testing of the assumed safety function.

The 24 month Frequency is based on the need to perform this Surveillance under the conditions that apply during a plant outage and the potential for an unplanned transient if the Surveillance were performed with the reactor at power. Operating experience has shown that these components usually pass the SR when performed at the 24 month Frequency, which is based on the refueling cycle. Therefore, the Frequency was concluded to be acceptable from a reliability standpoint.

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BASES

SURVEILLANCE REQUIREMENTS

SR 3.5.1.6 (continued)

This SR is modified by a Note that excludes vessel injection/spray during the Surveillance. Since all active components are testable and full flow can be demonstrated by recirculation through the test line, coolant injection into the RPV is not required during the Surveillance.

SR 3.5.1.7

The ADS designated S/RVs are required to actuate automatically upon receipt of specific initiation signals. A system functional test is performed to demonstrate that the mechanical portions of the ADS function (i.e., solenoids) operate as designed when initiated either by an actual or simulated initiation signal, causing proper actuation of all the required components. SR 3.5.1.8 and the LOGIC SYSTEM FUNCTIONAL TEST performed in LCO 3.3.5.1 overlap this Surveillance to provide complete testing of the assumed safety function.

The 24 month Frequency is based on the need to perform this Surveillance under the conditions that apply during a plant outage and the potential for an unplanned transient if the Surveillance were performed with the reactor at power. Operating experience has shown that these components usually pass the SR when performed at the 24 month Frequency, which is based on the refueling cycle. Therefore, the Frequency was concluded to be acceptable from a reliability standpoint.

This SR is modified by a Note that excludes valve actuation since the valves are individually tested in accordance with SR 3.5.1.8. This also prevents an RPV pressure blowdown.

SR 3.5.1.8

A manual actuation of each required ADS valve, and observing the expected change in the indicated valve position, is performed to verify that the valve and solenoids are functioning properly. SR 3.5.1.7 and the LOGIC SYSTEM FUNCTIONAL TEST performed in LCO 3.3.5.1 overlap this Surveillance to provide complete testing of the assumed safety function.

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BASES

SURVEILLANCE REQUIREMENTS

SR 3.5.1.8 (continued)

The Frequency of 24 months on a STAGGERED TEST BASIS ensures that both solenoids for each required ADS valve are alternately tested. The Frequency is based on the need to perform this Surveillance under the conditions that apply just prior to or during a startup from a plant outage. Operating experience has shown that these components usually pass the SR when performed at the 24 month Frequency, which is based on the refueling cycle. Therefore, the Frequency was concluded to be acceptable from a reliability standpoint.

SR 3.5.1.9

This SR ensures that the ECCS RESPONSE TIME for each ECCS injection/spray subsystem is less than or equal to the maximum value assumed in the accident analysis. Response time testing acceptance criteria are included in Reference 14. This SR is modified by a Note that allows the instrumentation portion of the response time to be assumed to be the design instrumentation response time and therefore, is excluded from the ECCS RESPONSE TIME testing. This is allowed since the instrumentation response time is a small part of the ECCS RESPONSE TIME (e.g., sufficient margin exists in the diesel generator start time when compared to the instrumentation response time) (Ref. 15). ECCS RESPONSE TIME tests are conducted every 24 months. The 24 month Frequency is consistent with the typical industry refueling cycle and is based upon plant operating experience.

REFERENCES

1. UFSAR, Section 6.3.2.2.3.
2. UFSAR, Section 6.3.2.2.4.
3. UFSAR, Section 6.3.2.2.1.
4. UFSAR, Section 6.3.2.2.2.
5. UFSAR, Section 15.2.8.
6. UFSAR, Section 15.6.4.

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BASES

REFERENCES (continued)

7. UFSAR, Section 15.6.5.

8. 10 CFR 50, Appendix K.

9. UFSAR, Section 6.3.3.

10. 10 CFR 50.46.

11. UFSAR, Section 6.3.3.3.

12. Memorandum from R.L. Baer (NRC) to V. Stello, Jr. (NRC), "Recommended Interim Revisions to LCO's for ECCS Components," December 1, 1975.

13. UFSAR, Section 7.3.1.2.

14. UFSAR, Table 6.3-2.

15. NEDO-32291-A, "System Analysis for the Elimination of Selected Response Time Testing Requirements," October 1995.

B 3.5 EMERGENCY CORE COOLING SYSTEMS (ECCS) AND REACTOR CORE ISOLATION COOLING (RCIC) SYSTEM

B 3.5.2 ECCS — Shutdown

BASES

BACKGROUND

A description of the High Pressure Core Spray (HPCS) System, Low Pressure Core Spray (LPCS) System, and low pressure coolant injection (LPCI) mode of the Residual Heat Removal (RHR) System is provided in the Bases for LCO 3.5.1, "ECCS — Operating."

APPLICABLE SAFETY ANALYSES

The ECCS performance is evaluated for the entire spectrum of break sizes for a postulated loss of coolant accident (LOCA). The long term cooling analysis following a design basis LOCA (Ref. 1) demonstrates that only one ECCS injection/spray subsystem is required, post LOCA, to maintain adequate reactor vessel water level in the event of an inadvertent vessel draindown. It is reasonable to assume, based on engineering judgment, that while in MODES 4 and 5, one ECCS injection/spray subsystem can maintain adequate reactor vessel water level. To provide redundancy, a minimum of two ECCS injection/spray subsystems are required to be OPERABLE in MODES 4 and 5.

The ECCS satisfy Criterion 3 of 10 CFR 50.36(c)(2)(ii).

LCO

Two ECCS injection/spray subsystems are required to be OPERABLE. The ECCS injection/spray subsystems are defined as the three LPCI subsystems, the LPCS System, and the HPCS System. The LPCS System and each LPCI subsystem consist of one motor driven pump, piping, and valves to transfer water from the suppression pool to the RPV. The HPCS System consists of one motor driven pump, piping, and valves to transfer water from the suppression pool to the RPV. The necessary portions of the Diesel Generator Cooling Water System are also required to provide appropriate cooling to each required ECCS injection/spray subsystem

As noted, one LPCI subsystem (A or B) may be considered OPERABLE during alignment and operation for decay heat removal, if capable of being manually realigned (remote or

(continued)

BASES

LCO (continued)

local) to the LPCI mode and is not otherwise inoperable. Alignment and operation for decay heat removal includes when the required RHR is not operating or when the system is realigned from or to the RHR shutdown cooling mode. This allowance is necessary since the RHR System may be required to operate in the shutdown cooling mode to remove decay heat and sensible heat from the reactor. Because of low pressure and low temperature conditions in MODES 4 and 5, sufficient time will be available to manually align and initiate LPCI subsystem operation to provide core cooling prior to postulated fuel uncovery.

APPLICABILITY

OPERABILITY of the ECCS injection/spray subsystems is required in MODES 4 and 5 to ensure adequate coolant inventory and sufficient heat removal capability for the irradiated fuel in the core in case of an inadvertent draindown of the vessel. Requirements for ECCS OPERABILITY during MODES 1, 2, and 3 are discussed in the Applicability section of the Bases for LCO 3.5.1. ECCS subsystems are not required to be OPERABLE during MODE 5 with the spent fuel storage pool gates removed and the water level maintained at $\geq$ 22 ft above the RPV flange. This provides sufficient coolant inventory to allow operator action to terminate the inventory loss prior to fuel uncovery in case of an inadvertent draindown.

The Automatic Depressurization System is not required to be OPERABLE during MODES 4 and 5 because the RPV pressure is < 150 psig, and the LPCS, HPCS, and LPCI subsystems can provide core cooling without any depressurization of the primary system.

ACTIONS

A.1 and B.1

If any one required ECCS injection/spray subsystem is inoperable, the required inoperable ECCS injection/spray subsystem must be restored to OPERABLE status within 4 hours. In this Condition, the remaining OPERABLE subsystem can provide sufficient RPV flooding capability to recover from an inadvertent vessel draindown. However, overall system reliability is reduced because a single failure in the remaining OPERABLE subsystem concurrent with

(continued)

BASES

ACTIONS

A.1 and B.1 (continued)

a vessel draindown could result in the ECCS not being able to perform its intended function. The 4 hour Completion Time for restoring the required ECCS injection/spray subsystem to OPERABLE status is based on engineering judgment that considered the availability of one subsystem and the low probability of a vessel draindown event.

With the inoperable subsystem not restored to OPERABLE status within the required Completion Time, action must be initiated immediately to suspend operations with a potential for draining the reactor vessel (OPDRVs) to minimize the probability of a vessel draindown and the subsequent potential for fission product release. Actions must continue until OPDRVs are suspended.

C.1, C.2, D.1, D.2, and D.3

If both of the required ECCS injection/spray subsystems are inoperable, all coolant inventory makeup capability may be unavailable. Therefore, actions must be initiated immediately to suspend OPDRVs in order to minimize the probability of a vessel draindown and the subsequent potential for fission product release. Actions must continue until OPDRVs are suspended. One ECCS injection/spray subsystem must also be restored to OPERABLE status within 4 hours. The 4 hour Completion Time to restore at least one ECCS injection/spray subsystem to OPERABLE status ensures that prompt action will be taken to provide the required cooling capacity or to initiate actions to place the plant in a condition that minimizes any potential fission product release to the environment.

If at least one ECCS injection/spray subsystem is not restored to OPERABLE status within the 4 hour Completion Time, additional actions are required to minimize any potential fission product release to the environment. This includes ensuring secondary containment is OPERABLE; one standby gas treatment subsystem is OPERABLE; and secondary containment isolation capability is available in each secondary containment penetration flow path not isolated that is assumed to be isolated to mitigate radioactivity

(continued)

BASES

ACTIONS

C.1, C.2, D.1, D.2, and D.3 (continued)

releases (i.e., one secondary containment isolation valve and associated instrumentation are OPERABLE or other acceptable administrative controls to assure isolation capability. The administrative controls consist of stationing a dedicated operator, who is in continuous communication with the control room at the controls of the isolation device. In this way, the penetration can be rapidly isolated when a need for secondary containment isolation is indicated.) This may be performed by an administrative check, by examining logs or other information, to determine if the components are out of service for maintenance or other reasons. It is not necessary to perform the Surveillances needed to demonstrate OPERABILITY of the components. If, however, any required component is inoperable, then it must be restored to OPERABLE status. In this case, the Surveillances may need to be performed to restore the component to OPERABLE status. Actions must continue until all required components are OPERABLE.

SURVEILLANCE REQUIREMENTS

SR 3.5.2.1 and SR 3.5.2.2

The minimum water level of -12 ft 7 in (referenced to a plant elevation of 699 ft 11 in) required for the suppression pool, equivalent to a contained water volume of 70,000 ft^3, is periodically verified to ensure that the suppression pool will provide adequate net positive suction head (NPSH) for the ECCS pumps, recirculation volume, and vortex prevention. With the suppression pool water level less than the required limit, all ECCS injection/spray subsystems are inoperable.

The 12 hour Frequency of these SRs was developed considering operating experience related to suppression pool water level variations and instrument drift during the applicable MODES. Furthermore, the 12 hour Frequency is considered adequate in view of other indications in the control room to alert the operator to an abnormal suppression pool water level condition.

(continued)

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.5.2.3, SR 3.5.2.5, and SR 3.5.2.6 and SR 3.5.2.7

The Bases provided for SR 3.5.1.1, SR 3.5.1.4, SR 3.5.1.5, and SR 3.5.1.8 are applicable to SR 3.5.2.3, SR 3.5.2.5, SR 3.5.2.6, and SR 3.5.2.7, respectively.

SR 3.5.2.4

Verifying the correct alignment for manual, power operated, and automatic valves in the ECCS flow paths provides assurance that the proper flow paths will exist for ECCS operation. This SR does not apply to valves that are locked, sealed, or otherwise secured in position since these valves were verified to be in the correct position prior to locking, sealing, or securing. A valve that receives an initiation signal is allowed to be in a nonaccident position provided the valve will automatically reposition in the proper stroke time. This SR does not require any testing or valve manipulation; rather, it involves verification that those valves capable of potentially being mispositioned are in the correct position. This SR does not apply to valves that cannot be inadvertently misaligned, such as check valves. The 31 day Frequency is appropriate because the valves are operated under procedural control and the probability of their being mispositioned during this time period is low.

REFERENCES

1. UFSAR, Section 6.3.3.2.

B 3.5 EMERGENCY CORE COOLING SYSTEMS (ECCS) AND REACTOR CORE ISOLATION COOLING (RCIC) SYSTEM

B 3.5.3 RCIC System

BASES

BACKGROUND

The RCIC System is not part of the ECCS; however, the RCIC System is included with the ECCS section because of their similar functions.

The RCIC System is designed to operate either automatically or manually following reactor pressure vessel (RPV) isolation accompanied by a loss of coolant flow from the feedwater system to provide adequate core cooling and control of RPV water level. Under these conditions, the High Pressure Core Spray (HPCS) and RCIC systems perform similar functions. The RCIC System design requirements ensure that the criteria of Reference 1 are satisfied.

The RCIC System (Ref. 2) consists of a steam driven turbine pump unit, piping and valves to provide steam to the turbine, as well as piping and valves to transfer water from the suction source to the core via the head spray nozzle. Suction piping is provided from the condensate storage tank (CST) and the suppression pool. Pump suction is normally aligned to the CST to minimize injection of suppression pool water into the RPV. However, if the CST water supply is low an automatic transfer to the suppression pool water source ensures a water supply for continuous operation of the RCIC System. The steam supply to the turbine is piped from main steam line B, upstream of the inboard main steam line isolation valve.

The RCIC System is designed to provide core cooling for a wide range of reactor pressures, 135 psig to 1185 psig. Upon receipt of an initiation signal, the RCIC turbine accelerates to a specified speed. As the RCIC flow increases, the turbine control valve is automatically adjusted to maintain design flow. Exhaust steam from the RCIC turbine is discharged to the suppression pool. A full flow test line is provided to route water to the CST or the suppression pool to allow testing of the RCIC System during normal operation without injecting water into the RPV.

(continued)

BASES

BACKGROUND (continued)	The RCIC pump is provided with a minimum flow bypass line, which discharges to the suppression pool. The valve in this line automatically opens to prevent pump damage due to overheating when other discharge line valves are closed. To ensure rapid delivery of water to the RPV and to minimize water hammer effects, the RCIC System discharge line "keep fill" system is designed to maintain the pump discharge line filled with water.
APPLICABLE SAFETY ANALYSES	The function of the RCIC System is to respond to transient events by providing makeup coolant to the reactor. The RCIC System is not an Engineered Safety Feature System and no credit is taken in the safety analyses for RCIC System operation. Based on its contribution to the reduction of overall plant risk, the system satisfies Criterion 4 of 10 CFR 50.36(c)(2)(ii).
LCO	The OPERABILITY of the RCIC System provides adequate core cooling such that actuation of any of the ECCS subsystems is not required in the event of RPV isolation accompanied by a loss of feedwater flow. The RCIC System has sufficient capacity to maintain RPV inventory during an isolation event.
APPLICABILITY	The RCIC System is required to be OPERABLE in MODE 1, and MODES 2 and 3 with reactor steam dome pressure > 150 psig since RCIC is the primary non-ECCS water source for core cooling when the reactor is isolated and pressurized. In MODES 2 and 3 with reactor steam dome pressure ≤ 150 psig, and in MODES 4 and 5, RCIC is not required to be OPERABLE since the ECCS injection/spray subsystems can provide sufficient flow to the vessel.
ACTIONS	<u>A.1 and A.2</u> If the RCIC System is inoperable during MODE 1, or MODES 2 or 3 with reactor steam dome pressure > 150 psig, and the HPCS System is immediately verified to be OPERABLE, the RCIC System must be restored to OPERABLE status within 14 days. In this Condition, loss of the RCIC System will not affect

(continued)

BASES

ACTIONS

A.1 and A.2 (continued)

the overall plant capability to provide makeup inventory at high RPV pressure since the HPCS System is the only high pressure system assumed to function during a loss of coolant accident (LOCA). OPERABILITY of the HPCS is therefore immediately verified when the RCIC System is inoperable. This may be performed as an administrative check, by examining logs or other information, to determine if the HPCS is out of service for maintenance or other reasons. Verification does not require performing the Surveillances needed to demonstrate the OPERABILITY of the HPCS System. If the OPERABILITY of the HPCS System cannot be immediately verified, however, Condition B must be entered. For transients and certain abnormal events with no LOCA, RCIC (as opposed to HPCS) is the preferred source of makeup coolant because of its relatively small capacity, which allows easier control of RPV water level. Therefore, a limited time is allowed to restore the inoperable RCIC to OPERABLE status.

The 14 day Completion Time is based on a reliability study (Ref. 3) that evaluated the impact on ECCS availability, assuming that various components and subsystems were taken out of service. The results were used to calculate the average availability of ECCS equipment needed to mitigate the consequences of a LOCA as a function of allowed outage times (AOTs). Because of the similar functions of the HPCS and RCIC, the AOTs (i.e., Completion Times) determined for the HPCS are also applied to RCIC.

B.1 and B.2

If the RCIC System cannot be restored to OPERABLE status within the associated Completion Time, or if the HPCS System is simultaneously inoperable, the plant must be brought to a condition in which the LCO does not apply. To achieve this status, the plant must be brought to at least MODE 3 within 12 hours and reactor steam dome pressure reduced to ≤ 150 psig within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

(continued)

BASES (continued)

SURVEILLANCE REQUIREMENTS

SR 3.5.3.1

The flow path piping has the potential to develop voids and pockets of entrained air. Maintaining the pump discharge line of the RCIC System full of water ensures that the system will perform properly, injecting its full capacity into the Reactor Coolant System upon demand. This will also prevent a water hammer following an initiation signal. One acceptable method of ensuring the line is full is to vent at the high points. The 31 day Frequency is based on the gradual nature of void buildup in the RCIC piping, the procedural controls governing system operation, and operating experience.

SR 3.5.3.2

Verifying the correct alignment for manual, power operated, and automatic valves (including the RCIC pump flow controller) in the RCIC flow path provides assurance that the proper flow path will exist for RCIC operation. This SR does not apply to valves that are locked, sealed, or otherwise secured in position since these were verified to be in the correct position prior to locking, sealing, or securing. A valve that receives an initiation signal is allowed to be in a nonaccident position provided the valve will automatically reposition in the proper stroke time. This SR does not require any testing or valve manipulation; rather, it involves verification that those valves capable of potentially being mispositioned are in the correct position. This SR does not apply to valves that cannot be inadvertently misaligned, such as check valves. For the RCIC System, this SR also includes the steam flow path for the turbine and the flow controller position.

The 31 day Frequency of this SR was derived from the Inservice Testing Program requirements for performing valve testing at least every 92 days. The Frequency of 31 days is further justified because the valves are operated under procedural control and because improper valve position would affect only the RCIC System. This Frequency has been shown to be acceptable through operating experience.

(continued)

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.5.3.3 and SR 3.5.3.4

The RCIC pump flow rates ensure that the system can maintain reactor coolant inventory during pressurized conditions with the RPV isolated. The flow tests for the RCIC System are performed at two different pressure ranges such that system capability to provide rated flow against a test line pressure corresponding to reactor pressure is tested both at the higher and lower operating ranges of the system. The required system head should overcome the RPV pressure and associated discharge line losses. Adequate reactor steam pressure must be available to perform these tests. Additionally, adequate steam flow must be passing through the main turbine or turbine bypass valves to continue to control reactor pressure when the RCIC System diverts steam flow. Therefore, sufficient time is allowed after adequate pressure and flow are achieved to perform these SRs. Reactor steam pressure must be ≥ 920 psig to perform SR 3.5.3.3 and ≥ 135 psig to perform SR 3.5.3.4. Adequate steam flow is represented by at least one turbine bypass valve opened 50%. Reactor startup is allowed prior to performing the low pressure Surveillance because the reactor pressure is low and the time to satisfactorily perform the Surveillance is short. The reactor pressure is allowed to be increased to normal operating pressure since it is assumed that the low pressure test has been satisfactorily completed and there is no indication or reason to believe that RCIC is inoperable. Therefore, these SRs are modified by Notes that state the Surveillances are not required to be performed until 12 hours after the reactor steam pressure and flow are adequate to perform the test. The 12 hours allowed for the flow tests after the required pressure and flow are reached are sufficient to achieve stable conditions for testing and provides a reasonable time to complete the SRs.

A 92 day Frequency for SR 3.5.3.3 is consistent with the Inservice Testing Program requirements. The 24 month Frequency for SR 3.5.3.4 is based on the need to perform this Surveillance under the conditions that apply during startup from a plant outage. Operating experience has shown that these components usually pass the SR when performed at the 24 month Frequency, which is based on the refueling cycle. Therefore, the Frequency was concluded to be acceptable from a reliability standpoint.

(continued)

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.5.3.5

The RCIC System is required to actuate automatically to perform its design function. This Surveillance verifies that with a required system initiation signal (actual or simulated) the automatic initiation logic of RCIC will cause the system to operate as designed, i.e., actuation of the system throughout its emergency operating sequence, which includes automatic pump startup and actuation of all automatic valves to their required positions. This Surveillance also ensures that the RCIC System will automatically restart on an actual or simulated RPV low water level (Level 2) signal received subsequent to an actual or simulated RPV high water level (Level 8) shutdown signal, and that the suction is automatically transferred from the CST to the suppression pool. The LOGIC SYSTEM FUNCTIONAL TEST performed in LCO 3.3.5.2 overlaps this Surveillance to provide complete testing of the assumed design function.

While this Surveillance can be performed with the reactor at power, operating experience has shown that these components usually pass the SR when performed at the 24 month Frequency, which is based on the refueling cycle. Therefore, the Frequency was concluded to be acceptable from a reliability standpoint.

This SR is modified by a Note that excludes vessel injection during the Surveillance. Since all active components are testable and full flow can be demonstrated by recirculation through the test line, coolant injection into the RPV is not required during the Surveillance.

REFERENCES

1. 10 CFR 50, Appendix A, GDC 33.

2. UFSAR, Section 5.4.6.2.

3. Memorandum from R.L. Baer (NRC) to V. Stello, Jr. (NRC), "Recommended Interim Revisions to LCO's for ECCS Components," December 1, 1975.

3/4.5 EMERGENCY CORE COOLING SYSTEMS

A.1

3/4.5.1 ECCS - OPERATING

LIMITING CONDITION FOR OPERATION

LCO 3.5.1 3.5.1 ECCS divisions 1, 2 and 3 shall be OPERABLE with:

a. ECCS division 1 consisting of:

1. The OPERABLE low pressure core spray (LPCS) system ~~with a flow path capable of taking suction from the suppression chamber and transferring the water through the spray sparger to the reactor vessel.~~ LA.1

2. The OPERABLE low pressure coolant injection (LPCI) subsystem "A" of the RHR system ~~with a flow path capable of taking suction from the suppression chamber and transferring the water to the reactor vessel.~~ LA.1

3. At least 6 OPERABLE** ADS valves. A.6

b. ECCS division 2 consisting of:

1. The OPERABLE low pressure coolant injection (LPCI) subsystems "B" and "C" of the RHR system, ~~each with a flow path capable of taking suction from the suppression chamber and transferring the water to the reactor vessel.~~ LA.1

2. At least 6 OPERABLE** ADS valves. A.6

c. ECCS division 3 consisting of the OPERABLE high pressure core spray (HPCS) system ~~with a flow path capable of taking suction from the suppression chamber and transferring the water through the spray sparger to the reactor vessel.~~ LA.1

APPLICABILITY: OPERATIONAL CONDITION 1, 2*# and 3*. A.2 Add LCO Note L.3

150 L.1

APPL *The ADS is not required to be OPERABLE when reactor steam dome pressure is less than or equal to ~~122~~ psig.

~~**See Specification 3/3.3 for trip system operability.~~ A.6

~~#See Special Test Exception 3.10.6.~~ A.2

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LIMITING CONDITION FOR OPERATION (Continued)

ACTION:

a. ~~For ECCS division 1, provided that ECCS divisions 2 and 3 are OPERABLE:~~ A.3

ACTION A
1. With the LPCS system inoperable, restore the inoperable LPCS system to OPERABLE status within 7 days.
2. With LPCI subsystem "A" inoperable, restore the inoperable LPCI subsystem "A" to OPERABLE status within 7 days.

ACTION C
3. With the LPCS system inoperable and LPCI subsystem "A" inoperable, restore at least the inoperable LPCI subsystem "A" or the inoperable LPCS system to OPERABLE status within 72 hours.

ACTION E
4. Otherwise, be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

b. ~~For ECCS division 2, provided that ECCS divisions 1 and 3 are OPERABLE:~~ A.3

ACTION A
1. With either LPCI subsystem "B" or "C" inoperable, restore the inoperable LPCI subsystem "B" or "C" to OPERABLE status within 7 days.

ACTION C
2. With both LPCI subsystems "B" and "C" inoperable, restore at least the inoperable LPCI subsystem "B" or "C" to OPERABLE status within 72 hours.

ACTION E
3. Otherwise, be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours~~*~~. A.4

c. For ECCS division 3, provided ~~that ECCS divisions 1 and 2 and~~ the RCIC system are OPERABLE: A.3 L.6

ACTION B
1. With ECCS division 3 inoperable, restore the inoperable division to OPERABLE status within 14 days.

ACTION E
2. Otherwise, be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

~~*Whenever two or more RHR 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.~~ A.4

LIMITING CONDITION FOR OPERATION (Continued)

ACTION: (Continued)

d. ~~For ECCS divisions 1 and 2, provided that ECCS division 3 is OPERABLE:~~ A.3

ACTION C — 1. With LPCI subsystem "A" and either LPCI subsystem "B" or "C" inoperable, restore at least the inoperable LPCI subsystem "A" or inoperable LPCI subsystem "B" or "C" to OPERABLE status within 72 hours.

ACTION C — 2. With the LPCS system inoperable and either LPCI subsystems "B" or "C" inoperable, restore at least the inoperable LPCS system or inoperable LPCI subsystem "B" or "C" to OPERABLE status within 72 hours.

ACTION E — 3. Otherwise, be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours*. A.4

e. ~~For ECCS divisions 1 and 2, provided that ECCS division 3 is OPERABLE and divisions 1 and 2 are otherwise OPERABLE:~~ A.3

ACTION F — 1. With one of the above required ADS valves inoperable, restore the inoperable ADS valve to OPERABLE status within 14 days or be

ACTION G — in at least HOT SHUTDOWN within the next 12 hours and reduce reactor steam dome pressure to ≤ ~~122~~ 150 psig within the next 24 hours. L.1

ACTION G — 2. 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 ≤ ~~122~~ 150 psig within the next 24 hours. L.1

f. ~~With an ECCS discharge line "keep filled" pressure alarm instrumentation channel inoperable, perform Surveillance Requirement 4.5.1.a.1 at least once per 24 hours.~~ L.5

g. ~~With an ECCS header delta P instrumentation channel inoperable, restore the inoperable channel to OPERABLE status within 72 hours or determine ECCS header delta P locally at least once per 12 hours; otherwise, declare the associated ECCS inoperable.~~ L.5

h. ~~With Surveillance Requirement 4.5.1.d.2 not performed at the required interval due to low reactor steam pressure, the provisions of Specification 4.0.4 are not applicable provided the surveillance is performed within 12 hours after reactor steam pressure is adequate to perform the test.~~ M.2

ADD proposed ACTION H — A.3

A.4

~~*Whenever two or more RHR 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.~~

LIMITING CONDITION FOR OPERATION (Continued)

ACTION: (Continued)

L.2

i. ~~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.C 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.~~

j. ~~With one or more ECCS corner room watertight doors inoperable, restore all the inoperable ECCS corner room watertight doors to OPERABLE status within 14 days, otherwise, be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.~~ LA.3

k. With ADS accumulator backup compressed gas system bottle pressure less than 500 psig, restore ADS accumulator backup compressed gas system bottle pressure to greater than 500 psig within 72 hours or declare the associated ADS valves inoperable, and follow Action e of this specification.

ACTION D

ACTION E

4.5.1 ECCS divisions 1, 2, and 3 shall be demonstrated OPERABLE by:

a. At least once per 31 days for the LPCS, LPCI, and HPCS systems: LA.2

SR 3.5.1.1 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. Performance of a CHANNEL FUNCTIONAL TEST of the:~~

~~a) Discharge line "keep filled" pressure alarm instrumentation, and~~

~~b) Header delta P instrumentation.~~ L.5

SR 3.5.1.2 3. 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.

~~4. Verifying that each ECCS corner room watertight door is closed except during entry to and exit from the room.~~ LA.3

b. Verifying that, when tested pursuant to Specification 4.0.5, each:

SR 3.5.1.5

1. LPCS pump develops a flow of at least 6350 gpm against a test line pressure greater than or equal to 290 psig.

2. LPCI pump develops a flow of at least 7200 gpm against a test line pressure greater than or equal to 130 psig.

3. HPCS pump develops a flow of at least 6250 gpm against a test line pressure greater than or equal to 370 psig.

c. For the LPCS, LPCI and HPCS systems, at least once per ~~18~~ 24 LD.1 months: L.4

SR 3.5.1.6 1. Performing a system functional test which includes an actual or 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.~~ LA.2 Actual injection of coolant into the reactor vessel may be excluded from this test.

~~2. Performing a CHANNEL CALIBRATION of the:~~

~~a) Discharge line "keep filled" pressure alarm instrumentation and verifying the:~~

~~1) High pressure setpoint allowable value and the low pressure setpoint allowable value of the:~~ L.5

(a) LPCS system to be ≤500 psig and ≥45.5 psig, respectively.

(b) LPCI subsystem "A" to be ≤400 psig and ≥41.0 psig, respectively.

(c) LPCI subsystem "B" to be ≤400 psig and ≥38.5 psig, respectively.

(d) LPCI subsystem "C" to be ≤400 psig and ≥45.0 psig, respectively.

2) Low pressure setpoint allowable value of the HPCS system to be ≥42.5 psig.

b) Header delta P instrumentation and verifying the setpoint allowable value of the:

1) LPCS system and LPCI subsystems to be ±1 psid.

2) HPCS system to be 5 ±2.0 psid greater than the normal indicated ΔP.

L.5

3. Deleted.

4. Visually inspecting the ECCS corner room watertight door seals and room penetration seals and verifying no abnormal degradation, damage, or obstructions.

LA.3

d. For the ADS by:

1. At least once per 31 days:

SR 3.5.1.3 a) Verify ADS accumulator supply header pressure is ≥ 150 psig.

SR 3.5.1.4 b) Verify ADS accumulator backup compressed gas system bottle pressure is ≥ 500 psig.

2. At least once per 18 months 24 LD.1

L.4 an actuator

SR 3.5.1.7 a) Performing a system functional test which includes simulated automatic actuation of the system throughout its emergency operating sequence, but excluding actual valve actuation.

SR 3.5.1.8 b) Manually opening each ADS valve ~~and observing the expected change in the indicated valve position.~~

LA.2

on a STAGGERED TEST BASIS for each valve solenoid (SR 3.5.1.8 only)

M.1

A.1

ITS 3.5.1

3/4.3.3 EMERGENCY CORE COOLING SYSTEM ACTUATION INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.3 The emergency core cooling system (ECCS) actuation instrumentation channels shown in Table 3.3.3-1 shall be OPERABLE with their trip setpoints set consistent with the values shown in the Trip Setpoint column of Table 3.3.3-2 ~~and with EMERGENCY CORE COOLING SYSTEM RESPONSE TIME as shown in Table 3.3.3-3~~. LA.4

APPLICABILITY: As shown in Table 3.3.3-1.

ACTION:

a. With an ECCS actuation instrumentation channel trip setpoint less conservative than the value shown in the Allowable Values column of Table 3.3.3-2, declare the channel inoperable until the channel is restored to OPERABLE status with its trip setpoint adjusted consistent with the Trip Setpoint value.

b. With one or more ECCS actuation instrumentation channels inoperable, take the ACTION required by Table 3.3.3-1.

c. With either ADS trip system "A" or "B" inoperable, restore the inoperable trip system to OPERABLE status within:

1. 7 days, provided that the HPCS and RCIC systems are OPERABLE.

2. 72 hours.

Otherwise, be in at least HOT SHUTDOWN within the next 12 hours and reduce reactor steam dome pressure to less than or equal to 122 psig within the following 24 hours.

SURVEILLANCE REQUIREMENTS

4.3.3.1 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 CONDITIONS and at the frequencies shown in Table 4.3.3.1-1.

4.3.3.2 LOGIC SYSTEM FUNCTIONAL TESTS and simulated automatic operation of all channels shall be performed at least once per 18 months.

(See ITS 3.3.5.1)

SR 3.5.1.9 4.3.3.3 The ECCS RESPONSE TIME of each ECCS trip function ~~shown in Table 3.3.3-3~~ shall be demonstrated to be within the limit at least once per ~~18~~ 24 months. ~~Each test shall include at least one channel per trip system such that all channels are tested at least once every N times 18 months where N is the total number of redundant channels in a specific ECCS trip system.~~ LA4

24 LD.1

ADD Proposed SR 3.5.1.9 Note A.5

TABLE 3.3.3-3

EMERGENCY CORE COOLING SYSTEM RESPONSE TIMES

ECCS	RESPONSE TIME (Seconds)
1. LOW PRESSURE CORE SPRAY SYSTEM	≤ 60*, #
2. LOW PRESSURE COOLANT INJECTION MODE OF RHR SYSTEM (Pumps A, B, and C)	≤ 60*, #
3. AUTOMATIC DEPRESSURIZATION SYSTEM	NA
4. HIGH PRESSURE CORE SPRAY SYSTEM	≤ 41#
5. LOSS OF POWER	NA

LA.4

*Injection valves shall be fully OPEN within 40 seconds after receipt of the reactor vessel pressure and ECCS Injection Line Pressure Interlock signal concurrently with power source availability and receipt of an accident initiation signal.

#ECCS actuation instrumentation is eliminated from response time testing.

SR 3.5.1.9 NOTE

A.1

3/4.5 EMERGENCY CORE COOLING SYSTEMS

3/4.5.1 ECCS - OPERATING

LIMITING CONDITION FOR OPERATION

LCO 3.5.1 3.5.1 ECCS divisions 1, 2 and 3 shall be OPERABLE with:

a. ECCS division 1 consisting of:

1. The OPERABLE low pressure core spray (LPCS) system ~~with a flow path capable of taking suction from the suppression chamber and transferring the water through the spray sparger to the reactor vessel.~~ LA.1

2. The OPERABLE low pressure coolant injection (LPCI) subsystem "A" of the RHR system ~~with a flow path capable of taking suction from the suppression chamber and transferring the water to the reactor vessel.~~ LA.1

3. At least 6 OPERABLE ~~**~~ ADS valves. A.6

b. ECCS division 2 consisting of:

1. The OPERABLE low pressure coolant injection (LPCI) subsystems "B" and "C" of the RHR system, ~~each with a flow path capable of taking suction from the suppression chamber and transferring the water to the reactor vessel.~~ LA.1

2. At least 6 OPERABLE ~~**~~ ADS valves. A.6

c. ECCS division 3 consisting of the OPERABLE high pressure core spray (HPCS) system ~~with a flow path capable of taking suction from the suppression chamber and transferring the water through the spray sparger to the reactor vessel.~~ LA.1

APPLICABILITY: OPERATIONAL CONDITION 1, 2* ~~#~~ and 3*. A.2 Add LCO Note L.3

APPL *The ADS is not required to be OPERABLE when reactor steam dome pressure is less than or equal to ~~122~~ 150 psig. L.1

~~**See Specification 3/3.3 for trip system operability.~~ A.6

~~#See Special Test Exception 3.10.6.~~ A.2

A.1

EMERGENCY CORE COOLING SYSTEMS

LIMITING CONDITION FOR OPERATION (Continued)

ACTION:

a. ~~For ECCS division 1, provided that ECCS divisions 2 and 3 are OPERABLE:~~ A.3

ACTION A
1. With the LPCS system inoperable, restore the inoperable LPCS system to OPERABLE status within 7 days.
2. With LPCI subsystem "A" inoperable, restore the inoperable LPCI subsystem "A" to OPERABLE status within 7 days.

ACTION C
3. With the LPCS system inoperable and LPCI subsystem "A" inoperable, restore at least the inoperable LPCI subsystem "A" or the inoperable LPCS system to OPERABLE status within 72 hours.

ACTION E
4. Otherwise, be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

b. ~~For ECCS division 2, provided that ECCS divisions 1 and 3 are OPERABLE:~~ A.3

ACTION A
1. With either LPCI subsystem "B" or "C" inoperable, restore the inoperable LPCI subsystem "B" or "C" to OPERABLE status within 7 days.

ACTION C
2. With both LPCI subsystems "B" and "C" inoperable, restore at least the inoperable LPCI subsystem "B" or "C" to OPERABLE status within 72 hours.

ACTION E
3. Otherwise, be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.* A.4

ACTION B
c. For ECCS division 3, provided ~~that ECCS divisions 1 and 2 and~~ the RCIC system are OPERABLE: A.3 L.6
1. With ECCS division 3 inoperable, restore the inoperable division to OPERABLE status within 14 days.

ACTION E
2. Otherwise, be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

d. ~~For ECCS divisions 1 and 2, provided that ECCS division 3 is OPERABLE:~~ A.3

ACTION C
1. With LPCI subsystem "A" and either LPCI subsystem "B" or "C" inoperable, restore at least the inoperable LPCI subsystem "A" or inoperable LPCI subsystem "B" or "C" to OPERABLE status within 72 hours.

~~*Whenever two or more RHR 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.~~ A.4

ACTION: (Continued)

ACTION C — 2. With the LPCS system inoperable and either LPCI subsystems "B" or "C" inoperable, restore at least the inoperable LPCS system or inoperable LPCI subsystem "B" or "C" to OPERABLE status within 72 hours.

ACTION E — 3. Otherwise, be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours~~*~~. A.4

~~e. For ECCS divisions 1 and 2, provided that ECCS division 3 is OPERABLE and divisions 1 and 2 are otherwise OPERABLE:~~ A.3

ACTION F — 1. With one of the above required ADS valves inoperable, restore the inoperable ADS valve to OPERABLE status within 14 days or be

ACTION G — in at least HOT SHUTDOWN within the next 12 hours and reduce reactor steam dome pressure to ≤ ~~122~~ 150 psig within the next 24 hours. L.1

ACTION G — 2. 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 ≤ ~~122~~ 150 psig within the next 24 hours. L.1

~~f. With an ECCS discharge line "keep filled" pressure alarm instrumentation channel inoperable, perform Surveillance Requirement 4.5.1.a.1 at least once per 24 hours.~~ L.5

~~g. With an ECCS header delta P instrumentation channel inoperable, restore the inoperable channel to OPERABLE status within 72 hours or determine ECCS header delta P locally at least once per 12 hours; otherwise, declare the associated ECCS inoperable.~~ L.5

~~h. With Surveillance Requirement 4.5.1.d.2 not performed at the required interval due to low reactor steam pressure, the provisions of Specification 4.0.4 are not applicable provided the surveillance is performed within 12 hours after reactor steam pressure is adequate to perform the test.~~ M.1

ADD proposed ACTION H — A.3

~~*Whenever two or more RHR 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.~~ A.4

A.1 ITS 3.5.1

LIMITING CONDITION FOR OPERATION (Continued)

ACTION: (Continued)

i. ~~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.C 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.~~ L.2

j. ~~With one or more ECCS corner room watertight doors inoperable, restore all the inoperable ECCS corner room watertight doors to OPERABLE status within 14 days, otherwise, be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.~~ LA.3

k. With ADS accumulator backup compressed gas system bottle pressure less than 500 psig, restore ADS accumulator backup compressed gas system bottle pressure to greater than 500 psig within 72 hours or declare the associated ADS valves inoperable, and follow Action e of this specification.

ACTION D

ACTION E

A.1

EMERGENCY CORE COOLING SYSTEMS

SURVEILLANCE REQUIREMENTS

4.5.1 ECCS divisions 1, 2, and 3 shall be demonstrated OPERABLE by:

a. At least once per 31 days for the LPCS, LPCI, and HPCS systems: LA.2

SR 3.5.1.1 — 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. Performance of a CHANNEL FUNCTIONAL TEST of the:~~

~~a) Discharge line "keep filled" pressure alarm instrumentation, and~~ — L.5

~~b) Header delta P instrumentation.~~

SR 3.5.1.2 — 3. 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.

~~4. Verifying that each ECCS corner room watertight door is closed, except during entry to and exit from the room.~~ — LA.3

b. Verifying that, when tested pursuant to Specification 4.0.5, each:

SR 3.5.1.5 —

1. LPCS pump develops a flow of at least 6350 gpm against a test line pressure greater than or equal to 290 psig.

2. LPCI pump develops a flow of at least 7200 gpm against a test line pressure greater than or equal to 130 psig.

3. HPCS pump develops a flow of at least 6200 gpm against a test line pressure greater than or equal to 330 psig.

c. For the LPCS, LPCI and HPCS systems, at least once per ~~18~~ 24 months: LD.1

SR 3.5.1.6 — 1. Performing a system functional test which includes an actual or 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. L.4 LA.2

2. Performing a CHANNEL CALIBRATION of the:

   a) Discharge line "keep filled" pressure alarm instrumentation and verifying the:

      1) High pressure setpoint allowable value and the low pressure setpoint allowable value of the:

         (a) LPCS system to be ≤500 psig and ≥45.5 psig, respectively.

         (b) LPCI subsystem "A" to be ≤400 psig and ≥41.0 psig, respectively.

         (c) LPCI subsystem "B" to be ≤400 psig and ≥38.5 psig, respectively.

         (d) LPCI subsystem "C" to be ≤400 psig and ≥45.0 psig, respectively.

      2) Low pressure setpoint allowable value of the HPCS system to be ≥42.5 psig.

   b) Header delta P instrumentation and verifying the setpoint allowable value of the:

      1) LPCS system and LPCI subsystems to be ± 1 psid.

      2) HPCS system to be 5 ± 2.0 psid greater than the normal indicated ΔP.

L.5

3. Deleted

4. Visually inspecting the ECCS corner room watertight door seals and room penetration seals and verifying no abnormal degradation, damage, or obstructions.

LA.3

d. For the ADS by:

   1. At least once per 31 days:

SR 3.5.1.3

      a) Verify ADS accumulator supply header pressure is ≥ 150 psig.

SR 3.5.1.4

      b) Verify ADS accumulator backup compressed gas system bottle pressure is ≥ 500 psig.

   2. At least once per ~~18~~ 24 months. LD.1

SR 3.5.1.7

      a) Performing a system functional test which includes an actual or simulated automatic actuation of the system throughout its emergency operating sequence, but excluding actual valve actuation. L.4

SR 3.5.1.8

      b) Manually opening each ADS valve and observing the expected change in the indicated valve position. LA.2

on a STAGGERED TEST BASIS for each valve solenoid (SR 3.5.1.8 only) M.1

A.1

INSTRUMENTATION

3/4.3.3 EMERGENCY CORE COOLING SYSTEM ACTUATION INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.3 The emergency core cooling system (ECCS) actuation instrumentation channels shown in Table 3.3.3-1 shall be OPERABLE with their trip setpoints set consistent with the values shown in the Trip Setpoint column of Table 3.3.3-2 ~~and with EMERGENCY CORE COOLING SYSTEM RESPONSE TIME as shown in Table 3.3.3-3~~. LA.4

APPLICABILITY: As shown in Table 3.3.3-1.

ACTION:

a. With an ECCS actuation instrumentation channel trip setpoint less conservative than the value shown in the Allowable Values column of Table 3.3.3-2, declare the channel inoperable until the channel is restored to OPERABLE status with its trip setpoint adjusted consistent with the Trip Setpoint value.

b. With one or more ECCS actuation instrumentation channels inoperable, take the ACTION required by Table 3.3.3-1.

c. With either ADS trip system "A" or "B" inoperable, restore the inoperable trip system to OPERABLE status within:

1. 7 days, provided that the HPCS and RCIC systems are OPERABLE.
2. 72 hours.

Otherwise, be in at least HOT SHUTDOWN within the next 12 hours and reduce reactor steam dome pressure to less than or equal to 122 psig within the following 24 hours.

SURVEILLANCE REQUIREMENTS

4.3.3.1 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 CONDITIONS and at the frequencies shown in Table 4.3.3.1-1.

4.3.3.2 LOGIC SYSTEM FUNCTIONAL TESTS and simulated automatic operation of all channels shall be performed at least once per 18 months.

LA.4

SR 3.5.1.9 4.3.3.3 The ECCS RESPONSE TIME of each ECCS trip function ~~shown in Table 3.3.3-3~~ shall be demonstrated to be within the limit at least once per ~~18~~ 24 months. ~~Each test shall include at least one channel per trip system such that all channels are tested at least once every N times 18 months where N is the total number of redundant channels in a specific ECCS trip system.~~ LD.1

ADD proposed SR 3.5.1.9 Note A.5

(See ITS 3.3.5.1)

A.1

TABLE 3.3.3-3

EMERGENCY CORE COOLING SYSTEM RESPONSE TIMES

ECCS	RESPONSE TIME (Seconds)
1. LOW PRESSURE CORE SPRAY SYSTEM	≤ 60*,#
2. LOW PRESSURE COOLANT INJECTION MODE OF RHR SYSTEM (Pumps A, B, and C)	≤ 60*,#
3. AUTOMATIC DEPRESSURIZATION SYSTEM	NA
4. HIGH PRESSURE CORE SPRAY SYSTEM	≤ 41#
5. LOSS OF POWER	NA

LA.4

*Injection valves shall be fully OPEN within 40 seconds after receipt of the reactor vessel pressure and ECCS Injection Line Pressure Interlock signal concurrently with power source availability and receipt of an accident initiation signal.

#ECCS actuation instrumentation is eliminated from response time testing.

SR 3.5.1.9 Note

ADMINISTRATIVE

A.1 In the conversion of the LaSalle 1 and 2 current Technical Specifications (CTS) to the proposed plant specific Improved Technical Specifications (ITS), certain wording preferences or conventions are adopted that do not result in technical changes (either actual or interpretational). Editorial changes, reformatting, and revised numbering are adopted to make the ITS consistent with the BWR Standard Technical Specifications, NUREG-1434, Rev. 1 (i.e., the Improved Standard Technical Specifications (ISTS)).

A.2 CTS 3.5.1 Applicability footnote #, which provides a cross reference to CTS 3.10.6, has been deleted. The format of the proposed Technical Specifications does not include providing "cross references." Proposed LCO 3.0.7 adequately prescribes the use of the Special Operations LCOs without such references. Therefore the existing reference in CTS 3.5.1 footnote #, to "See Special Test Exception 3.10.6" serves no functional purpose, and its removal is administrative.

A.3 CTS 3.5.1 Actions a, b, c, d, and e provide Actions for each specific ECCS division or a combination of specific ECCS divisions. Each Action is only applicable provided the ECCS divisions not discussed by the individual Action are OPERABLE. ITS 3.5.1 ACTION G provides various combinations of ECCS subsystem inoperabilities which require entry into LCO 3.0.3 consistent with the CTS Actions for the same combinations. Therefore, the statements in CTS 3.5.1 Actions a, b, c, d, and e that require the opposite division equipment ("provided that..") are unnecessary and have been deleted.

A.4 CTS 3.5.1 Actions b.3 and d.3, footnote * provide an allowance that when two or more RHR subsystems are inoperable, if unable to attain COLD SHUTDOWN (MODE 4) as required by the Actions, then the unit is allowed to maintain reactor coolant temperature as low as practical, in lieu of attaining MODE 4. This footnote is removed since it provides unnecessary duplication of the ACTIONS required by CTS 3.4.9.1 and the proposed ACTIONS of ITS 3.4.9. Also, it contains no additional restrictions on the operation of the plant, and in fact, could be interpreted as a relaxation of the requirements to achieve MODE 4. The Action to be in MODE 4, which is modified by the footnote, adequately prescribes the requirement to make efforts to "maintain reactor coolant temperature as low as practical" (i.e., the duplicative requirement of the footnote). If conditions are such that MODE 4 cannot be attained, the Action remains in effect, essentially requiring efforts to reach MODE 4 to continue. Elimination of the footnote reflects an administrative presentation preference.

ADMINISTRATIVE (continued)

A.5 A Note has been added to CTS 4.3.3.3 (proposed SR 3.5.1.8) that exempts the ECCS instrumentation associated with each ECCS injection/spray subsystem from response time testing and allows the design instrumentation response time to be used in the determination of the ECCS RESPONSE TIME. In addition, the ECCS RESPONSE TIME testing requirement is moved to ITS 3.5.1, "Emergency Core Cooling Systems - Operating." Deletion of the response time test for this instrumentation was previously approved by the NRC in Amendment Numbers 114 and 99 for LaSalle Units 1 and 2, respectively.

As shown on CTS Table 3.3.3-3 (footnote #), for each ECCS injection/spray subsystem, only the instrumentation is eliminated from the response time testing. The overall ECCS system response time requirement for each ECCS injection/spray subsystem, which includes diesel generator, injection valves, pumps and other components, still applies. The diesel generator and injection valve TS response time requirements are not eliminated from the testing requirement, only the requirement to perform actual testing of instrumentation is eliminated. Therefore, the addition of the Note to CTS 4.3.3.3 and movement of the ECCS RESPONSE TIME testing requirements to ITS 3.5.1 are considered to be administrative in nature.

The above change is similar to that approved by the NRC in License Amendment No. 184 for Brunswick Units 1 and 2.

A.6 CTS 3.5.1 LCO footnote **, which provides a cross reference to CTS 3.3.3, has been deleted. The format of the proposed Technical Specifications does not include providing "cross references." Proposed LCO 3.3.5.1 adequately prescribes the conditions for trip system operability without such references. Therefore, the existing reference in CTS 3.5.1 to "See Specification 3.3.3 for trip system operability" serves no functional purpose and its removal is administrative.

TECHNICAL CHANGES - MORE RESTRICTIVE

M.1 CTS 4.5.1.d.2.b) requires each ADS valve to be manually opened every 18 months. The ADS valve has two solenoids, each of which can open the ADS valve. Thus, the same solenoid valve can be used to perform this SR every 18 months. Proposed SR 3.5.1.8 will now require both solenoids to be verified in the course of 48 months, as represented by the Staggered Test Basis requirement of the 24 month Frequency. This will ensure each ADS valve solenoid can open the ADS valve. This is an additional restriction on plant operation.

M.2 CTS 4.5.1.d.2.b requires each ADS valve to be manually opened every 18 months, but does not require it to be opened with reactor steam pressure. CTS 3.5.1 Action h provides a CTS 4.0.4 exception for this surveillance if it can not be performed due to low reactor steam pressure. This exception is proposed to be omitted for ITS. Experience at LaSalle 1 and 2 indicates that cycling S/RVs under steam flow conditions actually causes damage to the S/RV valve seats and results in valve leakage. By testing the S/RVs when the plant is shutdown and at low pressure (as is currently allowed by CTS 4.5.1.d.2.b) it is possible to slowly close the S/RVs and prevent this damage. As a result, LaSalle 1 and 2 have discontinued the practice of cycling the S/RVs at elevated pressures. Therefore, ITS SR 3.5.1.8 is not proposed to allow this test to be delayed based on reactor pressure and flow. This change represents an additional restriction on plant operation.

TECHNICAL CHANGES - LESS RESTRICTIVE

"Generic"

LA.1 The details of CTS 3.5.1 relating to ECCS OPERABILITY (in this case that the ECCS subsystems shall have flow paths capable of taking suction from the suppression chamber and transferring water to the reactor vessel) are proposed to be relocated to the Bases. The details for system OPERABILITY are not necessary in the LCO. The definition of OPERABILITY suffices. As such, the relocated details are not required to be in the ITS to provide adequate protection of the public health and safety. Changes to the Bases will be controlled by the provisions of the proposed Bases Control Program described in Chapter 5 of the ITS.

TECHNICAL CHANGES - LESS RESTRICTIVE (continued)

LA.2 The details of CTS 4.5.1.a.1, 4.5.1.c.1, and 4.5.1.d.2.b) relating to methods for performing Surveillances (i.e., venting at the high point vent, verifying actuation of the system throughout its emergency operating sequence, including each automatic valve actuating to the correct position, and verifying proper operation of the ADS valves) are proposed to be relocated to the Bases. These details are not necessary to ensure the OPERABILITY of the ECCS subsystems. The requirements of ITS 3.5.1, ECCS — Operating, and the associated Surveillance Requirements are adequate to ensure the ECCS subsystems are maintained OPERABLE. As such, the relocated details are not required to be in the ITS to provide adequate protection of the public health and safety. Changes to the Bases will be controlled by the provisions of the proposed Bases Control Program described in Chapter 5 of the ITS.

LA.3 The CTS requirements for corner room watertight doors to be Operable (i.e., closed and capable of repelling water), as denoted in CTS 3.5.1 Action j, 4.5.1.a.4, and 4.5.1.c.4, are being relocated to the Technical Requirements Manual (TRM). Like fire doors, these barriers protect essential plant equipment but do not provide any direct assurance for safe plant operations. As a result, these requirements are not required to be in the ITS to provide adequate protection of the public health and safety. Additionally, none of the four NRC Policy Statement criteria are applicable to this item. Therefore, moving these requirement to the TRM is appropriate and consistent with the NRC Policy Statement and 10 CFR 50.36. The TRM will be incorporated by reference into the UFSAR at ITS implementation. Changes to the TRM will be controlled in accordance with the provisions of 10 CFR 50.59.

LA.4 CTS Table 3.3.3-3 denoting ECCS Response Times is proposed to be relocated to the UFSAR. The specific notation of these response time limits are not necessary to ensure Operability of the ECCS. Proposed SR 3.5.1.9 requires the periodic verification of the ECCS Response Time for each ECCS injection (spray subsystem). As such, these relocated requirements are not required in the ITS to provide adequate protection of the public health and safety. Changes to the UFSAR will be controlled by the provisions of 10 CFR 50.59.

LD.1 The Frequencies for performing CTS 4.5.1.c.1, 4.5.1.d.2.a), 4.5.1.d.2.b), and 4.3.3.3 (proposed SRs 3.5.1.6, 3.5.1.7, 3.5.1.8, and 3.5.1.9) have been extended from 18 months to 24 months. The ECCS system functional tests, CTS 4.5.1.c.1 (proposed SR 3.5.1.6) ensure that a system initiation signal (actual or simulated) to the automatic initiation logic of HPCS, LPCS, and LPCI will cause the subsystems to operate as designed, including actuation of the system throughout its emergency operation sequence, automatic pump startup and actuation of all automatic valves to their required positions. The ECCS response

TECHNICAL CHANGES - LESS RESTRICTIVE

LD.1 (cont'd) time test, CTS 4.3.3.3 (proposed SR 3.5.1.9), ensures that each ECCS injection/spray subsystem responds in a manner consistent with the values assumed in the accident analysis. The ADS system functional test, CTS 4.5.d.2.a) (proposed SR 3.5.1.7), ensures the mechanical portions of the ADS function (i.e., solenoids) to operate as designed when initiated either by an actual or simulated initiation signal. The ADS manual actuation test, CTS 4.5.1.d.2.b) (proposed SR 3.5.1.8), ensures the valves and solenoids operate properly. The proposed change will allow these Surveillances to extend their Surveillance Frequency from the current 18 month Surveillance Frequency (i.e., a maximum of 22.5 months accounting for the allowable grace period specified in current Specification 4.0.2 and proposed SR 3.0.2) to a 24 month Surveillance Frequency (i.e., a maximum of 30 months accounting for the allowable grace period specified in current Specification 4.0.2 and proposed SR 3.0.2). This proposed change was evaluated in accordance with the guidance provided in NRC Generic Letter No. 91-04, "Changes in Technical Specification Surveillance Intervals to Accommodate a 24-Month Fuel Cycle," dated April 2, 1991. Reviews of historical maintenance and surveillance data have shown that these tests normally pass their Surveillances at the current Frequency. An evaluation has been performed using this data, and it has been determined that the effect on safety due to the extended Surveillance Frequency will be minimal. The ECCS network has built-in redundancy so that no single failure will prevent the starting of the ECCS system. Each of the ECCS injection/spray systems are tested every three months according to the ASME Section XI inservice testing program (proposed SR 3.5.1.5) to ensure that each subsystem can provide the proper flow against a specified test pressure. This test will detect significant failures in the ECCS subsystems to perform their safety function. In addition, SRs 3.5.1.1, 3.5.1.2, and 3.5.1.3 are also performed every 31 days to ensure the ECCS subsystems are available to perform their required function. Extending the surveillance requirement on the ADS functional test will not have a significant impact on reliability because ADS is equipped with two redundant trip systems. Additionally, the S/RVs associated with ADS are equipped with remote manual switches so that the entire system can be operated manually as well as automatically. The primary function of ADS is to serve as backup to the HPCS System. If HPCS were to fail, ADS must activate to lower reactor pressure so that the low pressure ECCS spray/injection systems may operate. Based on the inherent system and component reliability and the testing performed during the operating cycle, the impact, if any, from this change on system availability is minimal. The review of historical surveillance data also demonstrated that there are no failures that would invalidate this conclusion. In addition, the proposed 24 month Surveillance Frequencies, if performed at the maximum interval allowed by proposed SR 3.0.2 (30 months) do not invalidate any assumptions in the plant licensing basis.

TECHNICAL CHANGES - LESS RESTRICTIVE (continued)

"Specific"

L.1 The pressure at which ADS is required to be OPERABLE, as specified in the CTS 3.5.1 APPLICABILITY and ACTIONS e.1 and e.2, is increased from 122 psig in ITS 3.5.1 to 150 psig to provide consistency of the OPERABILITY requirements for all ECCS and RCIC equipment. Small break loss of coolant accidents at low pressures (i.e., between 122 psig and 150 psig) are bounded by analyses performed at higher pressures. The ADS is required to operate to lower the pressure sufficiently so that the low pressure coolant injection (LPCI) and low pressure core spray (LPCS) systems can provide makeup to mitigate such accidents up to approximately 200 psig. Therefore, there is no safety significance in the ADS not being OPERABLE between 122 psig and 150 psig.

L.2 The CTS 3.5.1 Action i requirement to submit a Special Report for ECCS actuation and injection is adequately addressed by 10 CFR 50.73(a)(2)(iv). This CFR section requires an LER to be submitted for any event or condition that resulted in manual or automatic ECCS "actuation." Therefore, this LER will cover any "actuation and injection" as stipulated by the Special Report. This LER is required to be submitted within 30 days which also meets the Special Report requirement of 90 days. The necessary actuation cycle information for LaSalle 1 and 2 will be controlled by plant procedures. Regulations provide sufficient control of these provisions for their removal from Technical Specifications.

L.3 A Note clarifying the alignment requirements of the LPCI subsystems is included in ITS LCO 3.5.1 (CTS LCO 3.5.1). The Note allows operation of one or more of the RHR subsystems in the shutdown cooling mode during MODE 3, if necessary, and clarifies that the subsystems may still be considered OPERABLE for the LPCI mode. Because manual valve positioning, required for this mode of operation, removes the capability of the subsystems to respond automatically, the subsystems would be considered inoperable without this Note. Although no specific analysis of this condition has been performed, the allowance provided by the Note is acceptable because the return to OPERABILITY entails only the repositioning of valves, either remote or locally, and the energy requiring dissipation in MODE 3, below the RHR cut-in permissive pressure is considerably less than that at 100% power with normal operating temperature and pressure. Further, because of the low probability of an event requiring an ECCS and the certain need for shutdown cooling, it is considered appropriate to have the RHR subsystems aligned for decay heat removal.

TECHNICAL CHANGES - LESS RESTRICTIVE (continued)

L.4 The phrase "actual or," in reference to the automatic initiation signal, has been added to CTS 4.5.1.c (proposed SR 3.5.1.5) and 4.5.1.d.2.a (proposed SR 3.5.1.7), the Surveillance Requirements that verify each ECCS subsystem and ADS actuates on a "simulated" automatic initiation signal. This allows satisfactory "actual" automatic system initiations to be used to fulfill the Surveillance Requirements. OPERABILITY is adequately demonstrated in either case since the ECCS subsystem and ADS themselves cannot discriminate between "actual" or "simulated" signals.

L.5 The CTS requirements for performing a Channel Functional Test (CTS 4.5.1.a.2), a Channel Calibration (CTS 4.5.1.c.2), and associated Actions for inoperable instrumentation (CTS 3.5.1, Actions f and g), on the ECCS discharge line keep fill and differential pressure instrumentation, are being deleted. These requirements do not necessarily relate directly to ECCS Operability. The BWR Standard Technical Specifications, NUREG-1434, Rev. 1, does not specify alarm-only equipment to be Operable to support Operability of a system or component. Control of the availability of, and necessary compensatory activities if not available for alarms, are addressed by plant operational procedures and policies. The requirements of proposed LCO 3.5.1 and associated Surveillance Requirements will ensure the ECCS pumps are maintained Operable and their discharge lines filled. In addition, 10 CFR 50, Appendix B, Part XII requires that measuring devices used in activities affecting quality are properly controlled, calibrated, and adjusted to maintain accuracy within necessary limits. LaSalle Units 1 and 2 are required to comply with 10 CFR 50, thus if instrumentation is used to comply with proposed SR 3.5.1.1, it would be required to meet the 10 CFR 50, Appendix B, Part XII requirements. Therefore, this instrumentation, along with the supporting Surveillances, are proposed to be deleted.

L.6 In the event that ECCS Division 3 (HPCS) is inoperable, CTS Action 3.5.1.c requires that the RCIC System be OPERABLE. CTS Action 3.5.1.c is applicable in MODES 1, 2, and 3. Under proposed ITS 3.5.1 ACTION B.1, if the HPCS System is INOPERABLE, the RCIC System must be verified to be OPERABLE "when it (RCIC) is required to be OPERABLE." The APPLICABILITY for proposed ITS 3.5.3, which provides OPERABILITY requirements for the RCIC System, is MODE 1, and MODES 2 and 3 with reactor steam dome pressure greater than 150 psig. As a result, while CTS requires the RCIC System to be OPERABLE at any reactor steam dome pressure in MODE 2 or 3, under proposed ITS 3.5.1, RCIC is only required to be OPERABLE when reactor steam dome pressure is greater than 150 psig in MODE 2 or 3. The RCIC System is designed to provide core cooling over a

TECHNICAL CHANGES - LESS RESTRICTIVE

L.6 (cont'd) wide range of reactor pressures, and has a lower limit of 135 psig that is only slightly below 150 psig. Additionally, multiple low pressure ECCS are available when reactor steam dome pressure is less than 150 psig. Therefore, this change is considered acceptable since it will have minimal impact on core cooling capability.

RELOCATED SPECIFICATIONS

None

EMERGENCY CORE COOLING SYSTEMS

A.1

3/4.5.2 ECCS - SHUTDOWN

LIMITING CONDITION FOR OPERATION

ECCS injection/spray subsystems

LA.1

3.5.2 At least two ~~of the following~~ shall be OPERABLE:

a. ~~The low pressure core spray (LPCS) system with a flow path capable of taking suction from the suppression chamber and transferring the water through the spray sparger to the reactor vessel.~~

b. ~~Low pressure coolant injection (LPCI) subsystem "A" of the RHR system with a flow path capable of taking suction from the suppression chamber upon being manually realigned and transferring the water to the reactor vessel.~~

c. ~~Low pressure coolant injection (LPCI) subsystem "B" of the RHR system with a flow path capable of taking suction from the suppression chamber upon being manually realigned and transferring the water to the reactor vessel.~~

d. ~~Low pressure coolant injection (LPCI) subsystem "C" of the RHR system with a flow path capable of taking suction from the suppression chamber upon being manually realigned and transferring the water to the reactor vessel.~~

e. ~~The high pressure core spray (HPCS) system with a flow path capable of taking suction from the suppression pool and transferring the water through the spray sparger to the reactor vessel.~~

APPLICABILITY: OPERATIONAL CONDITION 4 or 5*.

Add LCO Note — L.4

ACTION:

ACTION A / ACTION B

a. With one of the above required subsystems/systems inoperable, restore at least two subsystems/systems to OPERABLE status within 4 hours or suspend all operations that have a potential for draining the reactor vessel.

L.1

ACTION C / ACTION D

b. With both of the above required subsystems/systems inoperable, suspend CORE ALTERATIONS and all operations that have a potential for draining the reactor vessel. Restore at least one subsystem/system to OPERABLE status within 4 hours or establish SECONDARY CONTAINMENT INTEGRITY within the next 8 hours.

A.2

A.3

APPL

*The ECCS is not required to be OPERABLE provided ~~that the reactor vessel head is removed, the cavity is flooded,~~ the spent fuel pool gates are removed, and water level is maintained within the limits of Specifications 3.9.8 and 3.9.9.

A.4

A.1

SURVEILLANCE REQUIREMENTS

SR 3.5.2.3
SR 3.5.2.4
SR 3.5.2.5
SR 3.5.2.6

4.5.2.1 At least the above required ECCS shall be demonstrated OPERABLE per Surveillance Requirement 4.5.1~~, except that the header delta P instrumentation is not required to be OPERABLE~~.

A.5

EMERGENCY CORE COOLING SYSTEMS A.1

3/4.5.3 SUPPRESSION ~~CHAMBERS~~ A.8

LIMITING CONDITION FOR OPERATION

Moved to ITS 3.6.2.2

3.5.3 The suppression chamber shall be OPERABLE: A.6

a. In OPERATIONAL CONDITION 1, 2, or 3 with a contained water volume of at least 128,800 ft^3, equivalent to a level of -4 1/2 inches.**

b. In OPERATIONAL CONDITION 4 or 5* with ~~a contained water volume of at least 70,000 ft^3, equivalent to~~ a level of -12 feet 7 inches.** LA.3 LA.2

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2, 3, 4, and 5*.

ACTION:

Moved to ITS 3.6.2.2 A.6

a. In OPERATIONAL CONDITION 1, 2, or 3 with the suppression chamber water level less than the above limit, restore the water level to within the limit within 1 hour or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

ACTION C — b. In OPERATIONAL CONDITION 4 or 5* with the suppression chamber water level less than the above limit, suspend ~~CORE ALTERATIONS and~~ all operations that have a potential for draining the reactor vessel and L.1

ACTION D — ~~lock the reactor mode switch in the Shutdown position.~~ Establish SECONDARY CONTAINMENT INTEGRITY within 8 hours.

L.2 A.2 A.3

ADD Proposed Required Action C.2 L.3

~~#See Specification 3.6.2.1 for pressure suppression requirements.~~ A.8

APPL — *The suppression chamber is not required to be OPERABLE ~~provided that the reactor vessel head is removed, the cavity is flooded or being flooded from the suppression pool,~~ the spent fuel pool gates are removed ~~when the cavity is flooded,~~ and the water level is maintained within the limits of Specifications 3.9.8 and 3.9.9. A.4 M.1

~~**Level is referenced to a plant elevation of 699 feet 11 inches (See Figure B 3/4.6.2-1).~~ LA.3

A.1

SURVEILLANCE REQUIREMENTS

4.5.3.1 The suppression chamber shall be determined OPERABLE by verifying:

SR 3.5.2.1
SR 3.5.2.2

a. The water level to be greater than or equal to, as applicable:

1. -4 1/2 inches** at least once per 24 hours. A.6 Moved to ITS 3.6.2.2

2. -12 feet 7 inches** at least once per 12 hours. LA.3

4.5.3.2 With the suppression chamber level less than the above limit in OPERATIONAL CONDITION 5*, at least once per 12 hours verify footnote conditions* to be satisfied. A.7

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*The suppression chamber is not required to be OPERABLE provided that the reactor vessel head is removed, the cavity is flooded or being flooded from the suppression pool, the spent fuel pool gates are removed when the cavity is flooded, and the water level is maintained within the limits of Specifications 3.9.8 and 3.9.9. A.7

**Level is referenced to a plant elevation of 699 feet 11 inches (See Figure B 3/4.6.2-1). LA.3

3/4.3.3 EMERGENCY CORE COOLING SYSTEM ACTUATION INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.3 The emergency core cooling system (ECCS) actuation instrumentation channels shown in Table 3.3.3-1 shall be OPERABLE with their trip setpoints set consistent with the values shown in the Trip Setpoint column of Table 3.3.3-2 ~~and with EMERGENCY CORE COOLING SYSTEM RESPONSE TIME as shown in Table 3.3.3-3~~. LA.4

APPLICABILITY: As shown in Table 3.3.3-1.

ACTION:

a. With an ECCS actuation instrumentation channel trip setpoint less conservative than the value shown in the Allowable Values column of Table 3.3.3-2, declare the channel inoperable until the channel is restored to OPERABLE status with its trip setpoint adjusted consistent with the Trip Setpoint value.

b. With one or more ECCS actuation instrumentation channels inoperable, take the ACTION required by Table 3.3.3-1.

c. With either ADS trip system "A" or "B" inoperable, restore the inoperable trip system to OPERABLE status within:

1. 7 days, provided that the HPCS and RCIC systems are OPERABLE.

2. 72 hours.

Otherwise, be in at least HOT SHUTDOWN within the next 12 hours and reduce reactor steam dome pressure to less than or equal to 122 psig within the following 24 hours.

SURVEILLANCE REQUIREMENTS

4.3.3.1 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 CONDITIONS and at the frequencies shown in Table 4.3.3.1-1.

4.3.3.2 LOGIC SYSTEM FUNCTIONAL TESTS and simulated automatic operation of all channels shall be performed at least once per 18 months.

<See ITS 3.3.5.1>

LD.1

SR 3.5.2.7 4.3.3.3 The ECCS RESPONSE TIME of each ECCS trip function ~~shown in Table 3.3.3-3~~ shall be demonstrated to be within the limit at least once per ~~18~~ 24 months. ~~Each test shall include at least one channel per trip system such that all channels are tested at least once every N times 18 months where N is the total number of redundant channels in a specific ECCS trip system.~~ LA.4

ADD proposed NOTE TO SR 3.5.2.7 A.9

TABLE 3.3.3-3

EMERGENCY CORE COOLING SYSTEM RESPONSE TIMES

ECCS	RESPONSE TIME (Seconds)
1. LOW PRESSURE CORE SPRAY SYSTEM	≤ 60*, #
2. LOW PRESSURE COOLANT INJECTION MODE OF RHR SYSTEM (Pumps A, B, and C)	≤ 60*, #
3. AUTOMATIC DEPRESSURIZATION SYSTEM	NA
4. HIGH PRESSURE CORE SPRAY SYSTEM	≤ 41#
5. LOSS OF POWER	NA

LA.4

*Injection valves shall be fully OPEN within 40 seconds after receipt of the reactor vessel pressure and ECCS Injection Line Pressure Interlock signal concurrently with power source availability and receipt of an accident initiation signal.

#ECCS actuation instrumentation is eliminated from response time testing.

SR 3.5.2.7 NOTE

A.1

EMERGENCY CORE COOLING SYSTEMS

3/4.5.2 ECCS - SHUTDOWN

LIMITING CONDITION FOR OPERATION

ECCS injection/spray subsystems — LA.1

3.5.2 At least two ~~of the following~~ shall be OPERABLE:

a. ~~The low pressure core spray (LPCS) system with a flow path capable of taking suction from the suppression chamber and transferring the water through the spray sparger to the reactor vessel.~~

b. ~~Low pressure coolant injection (LPCI) subsystem "A" of the RHR system with a flow path capable of taking suction from the suppression chamber upon being manually realigned and transferring the water to the reactor vessel.~~

c. ~~Low pressure coolant injection (LPCI) subsystem "B" of the RHR system with a flow path capable of taking suction from the suppression chamber upon being manually realigned and transferring the water to the reactor vessel.~~

d. ~~Low pressure coolant injection (LPCI) subsystem "C" of the RHR system with a flow path capable of taking suction from the suppression chamber upon being manually realigned and transferring the water to the reactor vessel.~~

e. ~~The high pressure core spray (HPCS) system with a flow path capable of taking suction from the suppression pool and transferring the water through the spray sparger to the reactor vessel.~~

Add LCO Note — L.4

APPLICABILITY: OPERATIONAL CONDITION 4 or 5*.

ACTION:

ACTION A — a. With one of the above required subsystems/systems inoperable, restore at least two subsystems/systems to OPERABLE status within 4 hours or

ACTION B — suspend all operations that have a potential for draining the reactor vessel.

L.1

b. With both of the above required subsystems/systems inoperable,

ACTION C — suspend ~~CORE ALTERATIONS and~~ all operations that have a potential for draining the reactor vessel. Restore at least one subsystem/

ACTION D — system to OPERABLE status within 4 hours or establish ~~SECONDARY CONTAINMENT INTEGRITY~~ within the next 8 hours. — A.2 — A.3

APPL — *The ECCS is not required to be OPERABLE provided ~~that the reactor vessel head is removed, the cavity is flooded,~~ the spent fuel pool gates are removed, and water level is maintained within the limits of Specifications 3.9.8 and 3.9.9. — A.4

A.1

EMERGENCY CORE COOLING SYSTEMS

SURVEILLANCE REQUIREMENTS

SR 3.5.2.3
SR 3.5.2.4
SR 3.5.2.5
SR 3.5.2.6

4.5.2.1 At least the above required ECCS shall be demonstrated OPERABLE per Surveillance Requirement 4.5.1. ~~except that the header delta P instrumentation is not required to be OPERABLE.~~

A.5

EMERGENCY CORE COOLING SYSTEMS A.1

3/4.5.3 SUPPRESSION CHAMBER# A.8

LIMITING CONDITION FOR OPERATION

Moved to ITS 3.6.2.2 A.6

3.5.3 The suppression chamber shall be OPERABLE:

a. In OPERATIONAL CONDITION 1, 2, or 3 with a contained water volume of at least 128,800 ft^3, equivalent to a level of -4 1/2 inches.**

b. In OPERATIONAL CONDITION 4 or 5* with ~~a contained water volume of at least 70,000 ft^3, equivalent to~~ a level of -12 feet 7 inches.** LA.2 LA.3

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2, 3, 4, and 5*.

Moved to ITS 3.6.2.2 A.6

ACTION:

a. In OPERATIONAL CONDITION 1, 2, or 3 with the suppression chamber water level less than the above limit, restore the water level to within the limit within 1 hour or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

ACTION C — b. In OPERATIONAL CONDITION 4 or 5* with the suppression chamber water level less than the above limit, suspend ~~CORE ALTERATIONS and~~ all operations that have a potential for draining the reactor vessel and ~~lock the reactor mode switch in the Shutdown position.~~ L.1

ACTION D — Establish SECONDARY CONTAINMENT INTEGRITY within 8 hours. A.3 A.2 L.2

ADD PROPOSED Required Action C.2 L.3

~~#See Specification 3.6.2.1 for pressure suppression requirements.~~ A.8

APPL — *The suppression chamber is not required to be OPERABLE ~~provided that the reactor vessel head is removed, the cavity is flooded or being flooded from the suppression pool,~~ the spent fuel pool gates are removed ~~when the cavity is flooded,~~ and the water level is maintained within the limits of Specifications 3.9.8 and 3.9.9. A.4 M.1

~~**Level is referenced to a plant elevation of 699 feet 11 inches (see Figure B 3/4.6.2-1).~~ LA.3

SURVEILLANCE REQUIREMENTS

4.5.3.1 The suppression chamber shall be determined OPERABLE by verifying:

SR 3.5.2.1 SR 3.5.2.2

a. The water level to be greater than or equal to, as applicable:

1. -4 1/2 inches** at least once per 24 hours. Moved to ITS 3.6.2.2 A.6

2. -12 feet 7 inches** at least once per 12 hours. LA.3

~~4.5.3.2 With the suppression chamber level less than the above limit in OPERATIONAL CONDITION 5*, at least once per 12 hours verify footnote conditions* to be satisfied.~~ A.7

~~*The suppression chamber is not required to be OPERABLE provided that the reactor vessel head is removed, the cavity is flooded or being flooded from the suppression pool, the spent fuel pool gates are removed when the cavity is flooded, and the water level is maintained within the limits of Specifications 3.9.8 and 3.9.9.~~ A.7

~~**Level is referenced to a plant elevation of 699 feet 11 inches (See Figure B 3/4.6.2-1).~~ LA.3

INSTRUMENTATION

3/4.3.3 EMERGENCY CORE COOLING SYSTEM ACTUATION INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.3 The emergency core cooling system (ECCS) actuation instrumentation channels shown in Table 3.3.3-1 shall be OPERABLE with their trip setpoints set consistent with the values shown in the Trip Setpoint column of Table 3.3.3-2 ~~and with EMERGENCY CORE COOLING SYSTEM RESPONSE TIME as shown in Table 3.3.3-3.~~ LA.4

APPLICABILITY: As shown in Table 3.3.3-1.

ACTION:

a. With an ECCS actuation instrumentation channel trip setpoint less conservative than the value shown in the Allowable Values column of Table 3.3.3-2, declare the channel inoperable until the channel is restored to OPERABLE status with its trip setpoint adjusted consistent with the Trip Setpoint value.

b. With one or more ECCS actuation instrumentation channels inoperable, take the ACTION required by Table 3.3.3-1.

c. With either ADS trip system "A" or "B" inoperable, restore the inoperable trip system to OPERABLE status within:

1. 7 days, provided that the HPCS and RCIC systems are OPERABLE.

2. 72 hours.

Otherwise, be in at least HOT SHUTDOWN within the next 12 hours and reduce reactor steam dome pressure to less than or equal to 122 psig within the following 24 hours.

SURVEILLANCE REQUIREMENTS

4.3.3.1 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 CONDITIONS and at the frequencies shown in Table 4.3.3.1-1.

4.3.3.2 LOGIC SYSTEM FUNCTIONAL TESTS and simulated automatic operation of all channels shall be performed at least once per 18 months.

LA.4

SR 3.5.2.7 4.3.3.3 The ECCS RESPONSE TIME of each ECCS trip function ~~shown in Table 3.3.3-3~~ shall be demonstrated to be within the limit at least once per ~~18~~ 24 months. ~~Each test shall include at least one channel per trip system such that all channels are tested at least once every N times 18 months where N is the total number of redundant channels in a specific ECCS trip system.~~ LD.1

(See ITS 3.3.5.1)

A.9 ADD proposed NOTE TO SR 3.5.2.7

TABLE 3.3.3-3

EMERGENCY CORE COOLING SYSTEM RESPONSE TIMES

ECCS	RESPONSE TIME (Seconds)
1. LOW PRESSURE CORE SPRAY SYSTEM	≤ 60*, #
2. LOW PRESSURE COOLANT INJECTION MODE OF RHR SYSTEM (Pumps A, B, and C)	≤ 60*, #
3. AUTOMATIC DEPRESSURIZATION SYSTEM	NA
4. HIGH PRESSURE CORE SPRAY SYSTEM	≤ 41#
5. LOSS OF POWER	NA

LA.4

*Injection valves shall be fully OPEN within 40 seconds after receipt of the reactor vessel pressure and ECCS Injection Line Pressure Interlock signal concurrently with power source availability and receipt of an accident initiation signal.

#ECCS actuation instrumentation is eliminated from response time testing.

SR 3.5.2.7 NOTE

ADMINISTRATIVE

A.1 In the conversion of the LaSalle 1 and 2 current Technical Specifications (CTS) to the proposed plant specific Improved Technical Specifications (ITS), certain wording preferences or conventions are adopted that do not result in technical changes (either actual or interpretational). Editorial changes, reformatting, and revised numbering are adopted to make the ITS consistent with the BWR Standard Technical Specifications, NUREG-1434, Rev. 1 (i.e., the Improved Standard Technical Specifications (ISTS)).

A.2 This proposed change replaces the use of the defined term SECONDARY CONTAINMENT INTEGRITY in CTS 3.5.2 Action b and CTS 3.5.3 Action b with the essential elements of that definition. Refer also to the Discussion of Changes in the Definition section (Chapter 1.0), which addresses deletion of the Secondary Containment Integrity definition. The change is editorial in that all the individual requirements are specifically addressed by ITS 3.5.2 Required Actions D.1, D.2, and D.3. Therefore, the change is a presentation preference adopted by the BWR Standard Technical Specifications, NUREG-1434, Rev. 1, and is considered administrative only.

A.3 The CTS 3.5.2 Action b and CTS 3.5.3 Action b requirements to establish SECONDARY CONTAINMENT INTEGRITY within the next 8 hours appear to provide a period of time (8 hours) in which integrity could be violated even if capable of being maintained. Additionally, if the plant status is such that integrity is not capable of being established within 8 hours, the existing ACTIONS result in "non-compliance with the Technical Specifications" and a requirement for an LER. The intent of the ACTIONS is more appropriately presented in ITS 3.5.2 Required Actions D.1, D.2, and D.3, which require actions to be initiated immediately to restore the secondary containment boundary. With the proposed Required Actions, a significantly more conservative requirement to establish and maintain the secondary containment boundary is imposed. No longer would the provision to violate the boundary for up to 8 hours exist. However, this conservatism comes from the understanding that if best efforts to establish the boundary exceeded 8 hours, no LER would be required.

This interpretation of the ACTIONS intent is supported by the BWR Standard Technical Specifications, NUREG-1434, Rev. 1. Because this is an enhanced presentation of existing intent, the proposed change is considered administrative.

A.4 The superfluous statement in CTS 3.5.2 footnote * and CTS 3.5.3 footnote *, that the ECCS is not required to be OPERABLE provided "that the reactor vessel head is removed, the cavity is flooded" has been deleted. The footnotes also require the spent fuel pool gates to be removed and the water level maintained

ADMINISTRATIVE

A.4 (cont'd) within the limits of Specifications 3.9.8 and 3.9.9. The spent fuel pool gates can be removed and the water level maintained within the limits of CTS 3.9.8 and 3.9.9 only if the head is also removed and the cavity flooded, since CTS 3.9.8 is applicable only in MODE 5. Therefore, these additional words have been deleted.

A.5 CTS 4.5.2.1 requires the required ECCS systems/subsystems to be demonstrated OPERABLE per CTS 4.5.1. Under the new format of NUREG-1434, Revision 1, the individual Surveillance Requirements of CTS 4.5.2.1 are listed in ITS 3.5.2, the ECCS — Shutdown Specification, instead of simply referring to the Surveillances in ITS 3.5.1, the ECCS — Operating Specification. Therefore, the applicable Surveillance Requirements for CTS 4.5.1 for low pressure ECCS systems and for HPCS are also presented in the Surveillance Requirements for this Specification. In addition, the header differential pressure instrumentation is not included in the ITS, thus a reference to it is not needed. As such this rewording is merely an administrative change. The changes in these individual test requirements have been discussed in ITS 3.5.1 Surveillance Requirements discussions.

A.6 The CTS 3.5.3.a and associated Applicability, Action a, and CTS 4.5.3.1 requirements are being moved to ITS 3.6.2.2 in accordance with the format of the BWR Standard Technical Specifications, NUREG-1434, Revision 1. Any technical changes to these requirements will be addressed in the Discussion of Changes for ITS: 3.6.2.2.

A.7 CTS 4.5.3.2 requires periodic verification that the specified conditions of CTS 3.5.3 Applicability footnote * are met when the suppression pool is inoperable. Periodic verification that the unit condition remains within the Applicability and that entry into an ACTION has not occurred is not used in the BWR Standard Technical Specifications, NUREG-1434, Rev. 1 (and not typically found in current Technical Specifications). In general, this type of requirement is addressed by plant specific processes that continuously monitor plant conditions to ensure changes in MODES or other specified applicable conditions are performed in accordance with Technical Specifications and to ensure changes in the status of the plant that require entry into ACTIONS are identified in a timely manner. As a result, CTS 4.5.3.2 serves no safety purpose and is not included in ITS 3.5.2. Since this change is an enhanced presentation of existing intent, the change is considered administrative.

A.8 CTS 3/4.5.3 footnote #, which provides a cross reference to CTS 3.6.2.1, has been deleted. The format of the proposed Technical Specifications does not include providing "cross references." Proposed LCO 3.6.2.1 and LCO 3.6.2.2

ADMINISTRATIVE

A.8 (cont'd) adequately prescribe the pressure suppression requirements. Therefore, the existing reference in CTS 3/4.5.3 to "See Specifications 3.6.2.1 for pressure suppression requirements" serves no functional purpose and its removal is administrative.

A.9 A Note has been added to CTS 4.3.3.3 (proposed SR 3.5.2.7) that exempts the ECCS instrumentation associated with each ECCS injection/spray subsystem from response time testing and allows the design instrumentation response time to be used in the determination of the ECCS RESPONSE TIME. In addition, the ECCS RESPONSE TIME testing requirement is moved to ITS 3.5.2, "Emergency Core Cooling Systems - Shutdown." Deletion of the response time test for this instrumentation was previously approved by the NRC in Amendment Numbers 114 and 99 for LaSalle Units 1 and 2, respectively.

As shown on CTS Table 3.3.3-3 (footnote #) for each ECCS injection/spray subsystem, only the instrumentation is eliminated from the response time testing. The overall ECCS system response time requirement for each ECCS injection/spray subsystem, which includes diesel generator, injection valves, pumps and other components, still applies. The diesel generator and injection valve TS response time requirements are not eliminated from the testing requirement, only the requirement to perform actual testing of instrumentation is eliminated. Therefore, the addition of the Note to CTS 4.3.3.3 and movement of the ECCS RESPONSE TIME testing requirements to ITS 3.5.2 are considered to be administrative in nature.

The above change is similar to that approved by the NRC in License Amendment No. 184 for Brunswick Units 1 and 2.

TECHNICAL CHANGES - MORE RESTRICTIVE

M.1 The allowance in CTS 3.5.3 footnote * to not require the suppression pool to be OPERABLE during cavity flooding has been deleted. The ITS will require the suppression pool to be within the required limits until the cavity is completely flooded (as well as all other listed requirements met). This will ensure sufficient makeup water is available for the ECCS pumps during the cavity flooding operation. This is an additional restriction on plant operation.

TECHNICAL CHANGES - LESS RESTRICTIVE

"Generic"

LA.1 The details of CTS 3.5.2 relating to ECCS OPERABILITY (in this case what constitutes an OPERABLE ECCS subsystem) are proposed to be relocated to the Bases. ITS 3.5.2 will continue to require two ECCS subsystems to be OPERABLE. The details for system OPERABILITY are not necessary in the LCO. The definition of OPERABILITY suffices. As such, the relocated details are not required to be in the ITS to provide adequate protection of the public health and safety. Changes to the Bases will be controlled by the provisions of the proposed Bases Control Program described in Chapter 5 of the ITS.

LA.2 The suppression pool volume specified in CTS 3.5.3.b which corresponds to the level limit is proposed to be relocated to the Bases. The level limit is retained since this is the information available to the operator regarding the suppression pool. These volume and level limits are equivalent and interchangeable. Therefore, moving one of them to the Bases does not change the requirement and is a change in the presentation. As a result, the volume limit is not required to be in the ITS to provide adequate protection of public health and safety. Changes to the Bases will be controlled by the provisions of the proposed Bases Control Program described in Chapter 5 of the Technical Specifications.

LA.3 CTS 3.5.3 footnote ** and CTS 4.5.3.1.a.2 footnote **, which references the suppression pool level to a plant elevation, is proposed to be relocated to the Bases. The level limit that is retained referenced to the actual level indication that is available to the operators in the control room. This additional reference, indicating the instrument zero, is not needed to ensure the limits are maintained. The requirements of LCO 3.5.2 and SR 3.4.2.1 continue to ensure the proper suppression pool level is maintained. As a result, the reference to plant elevation of the level limit is not required to be in the ITS to provide adequate protection of public health and safety. Changes to the Bases will be controlled by the provisions of the Bases Control Program described in Chapter 5 of the Technical Specifications.

LA.4 CTS Table 3.3.3-3 denoting ECCS Response Times is proposed to be relocated to the TRM. The specific notation of these response time limits are not necessary to ensure Operability of the ECCS. Proposed SR 3.5.2.7 requires the periodic verification of the ECCS Response Time for each ECCS injection (spray subsystem). As such, these relocated requirements are not required in the ITS to provide adequate protection of the public health and safety. Changes to the TRM will be controlled by the provisions of 10 CFR 50.59.

TECHNICAL CHANGES - LESS RESTRICTIVE (continued)

LD.1 The Frequency for performing CTS 4.3.3.3 (proposed SR 3.5.2.7) has been extended from 18 months to 24 months. The ECCS response time test, CTS 4.3.3.3 (proposed SR 3.5.2.7), ensures that each ECCS injection/spray subsystem responds in a manner consistent with the values assumed in the accident analysis. The proposed change will allow these Surveillances to extend their Surveillance Frequency from the current 18 month Surveillance Frequency (i.e., a maximum of 22.5 months accounting for the allowable grace period specified in current Specification 4.0.2 and proposed SR 3.0.2) to a 24 month Surveillance Frequency (i.e., a maximum of 30 months accounting for the allowable grace period specified in current Specification 4.0.2 and proposed SR 3.0.2). This proposed change was evaluated in accordance with the guidance provided in NRC Generic Letter No. 91-04, "Changes in Technical Specification Surveillance Intervals to Accommodate a 24-Month Fuel Cycle," dated April 2, 1991. Reviews of historical maintenance and surveillance data have shown that these tests normally pass their Surveillances at the current Frequency. An evaluation has been performed using this data, and it has been determined that the effect on safety due to the extended Surveillance Frequency will be minimal. The ECCS network has built-in redundancy so that no single failure will prevent the starting of the ECCS system. Each of the ECCS injection/spray systems are tested every three months according to the ASME Section XI inservice testing program (proposed SR 3.5.2.5) to ensure that each subsystem can provide the proper flow against a specified test pressure. This test will detect significant failures in the ECCS subsystems to perform their safety function. In addition, SRs 3.5.2.1, 3.5.2.2, 3.5.2.3, and 3.5.2.4 are also performed every 12 hours or 31 days, as appropriate, to ensure the ECCS subsystems are available to perform their required function. Based on the inherent system and component reliability and the testing performed during the operating cycle, the impact, if any, from this change on system availability is minimal. The review of historical surveillance data also demonstrated that there are no failures that would invalidate this conclusion. In addition, the proposed 24 month Surveillance Frequencies, if performed at the maximum interval allowed by proposed SR 3.0.2 (30 months) do not invalidate any assumptions in the plant licensing basis.

"Specific"

L.1 The requirement of CTS 3.5.2 Action b to suspend CORE ALTERATIONS when both ECCS subsystems are inoperable and the requirement of CTS 3.5.3 Action b to suspend CORE ALTERATIONS when the suppression chamber water level requirement is not within limit have been deleted. Refueling LCOs provide requirements to ensure safe operation during CORE ALTERATIONS including required water level above the RPV flange. The ECCS function provides additional protection for loss of vessel inventory events. However,

TECHNICAL CHANGES - LESS RESTRICTIVE

L.1 (cont'd) these events are not initiated by, nor is the response of ECCS hampered by, CORE ALTERATION operations. Therefore, ITS 3.5.2 does not require this ACTION.

L.2 The CTS 3.5.3 Action b requirement to "lock" the reactor mode switch in shutdown is proposed to be deleted. The position of the reactor mode switch is adequately controlled by the MODES definition Table (proposed Table 1.1-1). Reactor mode switch positions other than Shutdown may result in the unit entering some other MODE; with the associated Technical Specification compliance requirements of that MODE and of proposed LCO 3.0.4. Only the Shutdown or Refuel position of the reactor mode switch are allowed for ITS 3.5.2 since a reactor mode switch position of other than Shutdown or Refuel results in entry into a MODE other than MODE 4 or 5. Therefore, the requirement to "lock" the reactor mode switch in Shutdown is proposed to be deleted from Technical Specifications.

L.3 CTS 3.5.3, Action b requires the establishment of Secondary Containment Integrity within 8 hours if the suppression pool water level is not within limits in MODES 4 and 5. Proposed Required Action C.2 has been added delaying this current ACTION for 4 hours to provide time to restore the limit (i.e., by restoring the affected ECCS subsystem to OPERABLE status). This 4 hour time is consistent with current LCO 3.5.2, ACTION b, which provides 4 hours to restore an inoperable ECCS subsystem with both required ECCS subsystems are inoperable, prior to the requirement to establish Secondary Containment Integrity.

L.4 A Note clarifying the alignment requirements of the LPCI subsystems is included in ITS LCO 3.5.2 (CTS LCO 3.5.2). The Note allows operation of one RHR subsystem in the shutdown cooling mode during shutdown conditions, if necessary, and clarifies that the subsystem may still be considered OPERABLE for the LPCI mode. Because manual valve positioning, required for this mode of operation, removes the capability of the subsystem to respond automatically, the subsystem would be considered inoperable without this Note. Although no specific analysis of this condition has been performed, the allowance provided by the Note is acceptable because the return to OPERABILITY entails only the repositioning of valves, either remote or locally, and the energy requiring dissipation during shutdown conditions is considerably less than that at 100% power with normal operating temperature and pressure. Further, because of the low probability of an event requiring an ECCS and the certain need for shutdown cooling, it is considered appropriate to have the RHR subsystems aligned for decay heat removal.

RELOCATED SPECIFICATIONS

None

PLANT SYSTEMS

3/4.7.3 REACTOR CORE ISOLATION COOLING SYSTEM

A.1

LIMITING CONDITION FOR OPERATION

LCO 3.5.3 3.7.3 The reactor core isolation cooling (RCIC) system shall be OPERABLE ~~with an OPERABLE flow path capable of taking suction from the suppression pool and transferring the water to the reactor pressure vessel.~~ LA.1

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2, and 3 with reactor steam dome pressure greater than 150 psig.

ACTION:

~~a. With a RCIC discharge line "keep filled" pressure alarm instrumentation channel inoperable, perform Surveillance Requirement 4.7.3.a.1 at least once per 24 hours.~~ L.2

ACTION A — b. With the RCIC system inoperable, operation may continue provided the HPCS system is OPERABLE; restore the RCIC system to OPERABLE status within 14 days or be in at least HOT SHUTDOWN within the next 12

ACTION B — hours and reduce reactor steam dome pressure to less than or equal to 150 psig within the following 24 hours.

SURVEILLANCE REQUIREMENTS

4.7.3 The RCIC system shall be demonstrated OPERABLE:

a. At least once per 31 days by:

LA.2

SR 3.5.3.1 — 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. Performance of a CHANNEL FUNCTIONAL TEST of the discharge line "keep filled" pressure alarm instrumentation, and~~ L.2

SR 3.5.3.2 — 3. 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.

~~4. Verifying that the pump flow controller is in the correct position.~~ LA.2

SR 3.5.3.3 — b. At least once per 92 days by verifying that the RCIC pump develops a flow of greater than or equal to 600 gpm in the test flow path with a system head corresponding to reactor vessel operating pressure when steam is being supplied to the turbine at 1000 + 20, - 80 psig.*

SR 3.5.3.3 NOTE — *The provisions of Specification 4.0.4 are not applicable provided the surveillance is performed within 12 hours after reactor steam pressure and flow is adequate to perform the tests. A.2

A.1

PLANT SYSTEMS

SURVEILLANCE REQUIREMENTS

c. At least once per ~~18~~ months by: 24 — LD.1

SR 3.5.3.5 — 1. Performing a system functional test which includes actual or simulated automatic actuation ~~and verifying that each automatic valve in the flow path actuates to its correct position~~, but may exclude actual injection of coolant into the reactor vessel. — L.1 — LA.2

SR 3.5.3.5 NOTE

SR 3.5.3.4 — 2. Verifying that the system ~~is~~ capable of providing a flow of greater than or equal to 600 gpm ~~to the reactor vessel~~ when steam is supplied to the turbine at a pressure of 150 ± 15 psig using the test flow path with a system head corresponding to reactor pressure. — A.4 — LA.3

~~3. Performing a CHANNEL CALIBRATION of the discharge line "keep filled" pressure alarm instrumentation and verifying the low pressure setpoint allowable value to be ≥29 psig.~~ — L.2

d. By demonstrating MCC-121y and the 250-volt battery and charger OPERABLE:

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

   a) MCC-121y is energized, and has correct breaker alignment, indicated power availability from the charger and battery, and voltage on the panel with an overall voltage of greater than or equal to 250 volts.
   b) The electrolyte level of each pilot cell is above the plates,
   c) The pilot cell specific gravity, corrected to 77°F, is greater than or equal to 1.200, and
   d) The overall battery voltage is greater than or equal to 250 volts.

2. At least once per 92 days by verifying that:

   a) The voltage of each connected battery is greater than or equal to 250 volts under float charge and has not decreased more than 12 volts from the value observed during the original test,
   b) The specific gravity, corrected to 77°F, of each connected cell is greater than or equal to 1.195 and has not decreased more than 0.05 from the value observed during the previous test, and
   c) The electrolyte level of each connected cell is above the plates.

Moved to ITS 3.8.4, 3.8.6, and 3.8.7

3. At least once per 18 months by verifying that:

   a) The battery shows no visual indication of physical damage or abnormal deterioration, and
   b) Battery terminal connections are clean, tight, free of corrosion and coated with anti-corrosion material.

A.3

SR 3.5.3.4 NOTE — *The provisions of Specification 4.0.4 are not applicably provided the surveillance is performed within 12 hours after reactor steam pressure and flow is adequate to perform the tests. — A.2

PLANT SYSTEMS A.1

3/4.7.3 REACTOR CORE ISOLATION COOLING SYSTEM

LIMITING CONDITION FOR OPERATION

LCO 3.5.3 3.7.3 The reactor core isolation cooling (RCIC) system shall be OPERABLE ~~with an OPERABLE flow path capable of taking suction from the suppression pool and transferring the water to the reactor pressure vessel.~~ LA.1

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2, and 3 with reactor steam dome pressure greater than 150 psig.

ACTION:

~~a. With a RCIC discharge line "keep filled" pressure alarm instrumentation channnel inoperable, perform Surveillance Requirement 4.7.3.a.1 at least once per 24 hours.~~ L.2

ACTION A
ACTION B
b. With the RCIC system inoperable, operation may continue provided the HPCS 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 less than or equal to 150 psig within the following 24 hours.

SURVEILLANCE REQUIREMENTS

4.7.3 The RCIC system shall be demonstrated OPERABLE:

a. At least once per 31 days by:

SR 3.5.3.1 1. Verifying ~~by venting at the high point vents~~ LA.2 that the system piping from the pump discharge valve to the system isolation valve is filled with water,

~~2. Performance of a CHANNEL FUNCTIONAL TEST of the discharge line "keep filled" pressure alarm instrumentation, and~~ L.2

SR 3.5.3.2 3. 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.

~~4. Verifying that the pump flow controller is in the correct position.~~ LA.2

SR 3.5.3.3 b. At least once per 92 days by verifying that the RCIC pump develops a flow of greater than or equal to 600 gpm in the test flow path with a system head corresponding to reactor vessel operating pressure when steam is being supplied to the turbine at 1000 + 20, - 80 psig.*

SR 3.5.3.3 NOTE *The provisions of Specification 4.0.4 are not applicable provided the surveillance is performed within 12 hours after reactor steam pressure and flow is adequate to perform the tests. A.2

A.1

SURVEILLANCE REQUIREMENTS

c. At least once per ~~18~~ months by: 24 LD.1 — actual or L.1

SR 3.5.3.5 1. 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,~~ but may exclude LA.2

SR 3.5.3.5 NOTE — actual injection of coolant into the reactor vessel.

SR 3.5.3.4 2. Verifying that the system is capable of providing a flow of greater than or equal to 600 gpm to the reactor vessel when steam is supplied to the turbine at a pressure ~~of 150 ± 15 psig~~ ~~using the test flow path.~~ with a system head corresponding to reactor pressure A.4 LA.3

~~3. Performing a CHANNEL CALIBRATION of the discharge line "keep filled" pressure alarm instrumentation and verifying the low pressure setpoint allowable value to be ≥29.0 psig.~~ L.2

d. By demonstrating MCC-221y and the 250-volt battery and charger OPERABLE:

1. At least once per 7 days by verifying that:
   a) MCC-221y is energized, and has correct breaker alignment, indicated power availability from the charger and battery, and voltage on the panel with an overall voltage of greater than or equal to 250 volts.
   b) The electrolyte level of each pilot cell is above the plates,
   c) The pilot cell specific gravity, corrected to 77°F, is greater than or equal to 1.200, and
   d) The overall battery voltage is greater than or equal to 250 volts.

2. At least once per 92 days by verifying that:
   a) The voltage of each connected battery is greater than or equal to 250 volts under float charge and has not decreased more than 12 volts from the value observed during the original test,
   b) The specific gravity, corrected to 77°F, of each connected cell is greater than or equal to 1.195 and has not decreased more than 0.05 from the value observed during the previous test, and
   c) The electrolyte level of each connected cell is above the plates.

3. At least once per 18 months by verifying that:
   a) The battery shows no visual indication of physical damage or abnormal deterioration, and
   b) Battery terminal connections are clean, tight, free of corrosion and coated with anticorrosion material.

Moved to ITS 3.8.4, 3.8.6, and 3.8.7

A.3

SR 3.5.3.4 NOTE — *The provisions of Specification 4.0.4 are not applicably provided the surveillance is performed within 12 hours after reactor steam pressure and flow is adequate to perform the tests. A.2

ADMINISTRATIVE

A.1 In the conversion of the LaSalle 1 and 2 current Technical Specifications (CTS) to the proposed plant specific Improved Technical Specifications (ITS), certain wording preferences or conventions are adopted that do not result in technical changes (either actual or interpretational). Editorial changes, reformatting, and revised numbering are adopted to make the ITS consistent with the BWR Standard Technical Specifications, NUREG-1434, Rev. 1 (i.e., the Improved Standard Technical Specifications (ISTS)).

A.2 CTS 4.7.3.b footnote * and CTS 4.7.3.c.2 footnote * allow the RCIC flow tests to be deferred until 12 hours after adequate pressure is available. Adequate pressure to perform the tests also implies adequate flow must be available to perform the tests. As such, the footnote has been modified (proposed Notes to SR 3.5.3.3 and SR 3.5.3.4) to allow deferral until adequate flow is also available. Therefore, this change is considered administrative.

A.3 The CTS 4.7.3.d requirements are being moved to ITS 3.8.4, 3.8.6, and 3.8.7 in accordance with the format of the BWR Standard Technical Specifications, NUREG-1434, Revision 1. Any technical changes to these requirements will be addressed in the Discussion of Changes for ITS 3.8.4, 3.8.6 and 3.8.7.

A.4 The requirement to verify the RCIC pump flow rate in CTS 4.7.3.c.2 is modified in ITS SR 3.5.3.4 to include the criteria of verifying pump flow against a system head corresponding to the reactor pressure. The purpose of the test is to verify that the RCIC System can maintain reactor coolant inventory during pressurized conditions with the RPV isolated. The intent of the SR is not to verify adequate RCIC pump flow at atmospheric pressure, since this does not demonstrate that the RCIC pump can deliver the required flow at reactor pressure > 150 psig. Therefore, this additional criteria has been added to clarify the true intent of the Surveillance. Since the current Surveillance would not be considered to be passed if the RCIC pump could not pump 600 gpm against a system head corresponding to reactor pressure, this change is considered administrative.

TECHNICAL CHANGES - MORE RESTRICTIVE

None

TECHNICAL CHANGES - LESS RESTRICTIVE

"Generic"

LA.1 The details of CTS 3.7.3 relating to system OPERABILITY (in this case that the RCIC System shall have a flow path capable of taking suction from the suppression pool and transferring water to the reactor pressure vessel) are proposed to be relocated to the Bases. The details for system OPERABILITY are not necessary in the LCO. The definition of OPERABILITY suffices. As such, the relocated details are not required to be in the ITS to provide adequate protection of the public health and safety. Changes to the Bases will be controlled by the provisions of the proposed Bases Control Program described in Chapter 5 of the ITS.

LA.2 The details of CTS 4.7.3.a.1, 4.7.3.a.4, and 4.7.3.c.1 relating to methods for performing Surveillances (i.e., by venting from the high point vent, verifying that the RCIC pump controller is in the correct position, and verifying that each automatic valve in the flow path actuates to the proper position during the actuation test) are proposed to be relocated to the Bases. These details are not necessary to ensure the OPERABILITY of the RCIC System. The requirements of ITS 3.5.3, RCIC System, and the associated Surveillance Requirements are adequate to ensure the RCIC System is maintained OPERABLE. As such, the relocated details are not required to be in the ITS to provide adequate protection of the public health and safety. Changes to the Bases will be controlled by the provisions of the proposed Bases Control Program described in Chapter 5 of the ITS.

LA.3 CTS 4.7.3.c.2 requires verifying RCIC System flow when steam pressure is 150 $\pm$15 psig. The minimum pressure for the test (135 psig) is proposed to be relocated to the Bases in the form of a discussion describing when adequate pressure is available to perform the test. This allowance is in the CTS as footnote * (proposed Note to SR 3.5.3.4), which describes that the test only has to be performed within 12 hours after reactor steam pressure is adequate to perform the test. The proposed Bases description provides the detail as to when adequate pressure is available. These details are not necessary to ensure the OPERABILITY of the RCIC System. The requirements of ITS 3.5.3 and SR 3.5.3.4 are adequate to ensure the RCIC System is maintained OPERABLE. As such, the relocated detail is not required to be in the ITS to provide adequate protection of the public health and safety. Changes to the Bases will be controlled by the provisions of the proposed Bases Control Program described in Chapter 5 of the ITS.

TECHNICAL CHANGES - LESS RESTRICTIVE (continued)

LD.1 The Frequencies for performing CTS 4.7.3.c.1 and 4.7.3.c.2, (proposed SRs 3.5.3.4 and 3.5.3.5) have been extended from 18 months to 24 months. The RCIC system functional test (proposed SR 3.5.3.5) ensures that a system initiation signal (actual or simulated) to the automatic initiation logic of RCIC will cause the systems or subsystems to operate as designed, including actuation of the system throughout its emergency operating sequence, automatic pump startup and actuation of all automatic valves to their required positions. The RCIC low pressure flow test (proposed SR 3.5.3.4) ensures the RCIC system is capable of performing its design function before reactor pressure is increased above the system minimum operating pressure. The proposed change will allow these Surveillances to extend their Surveillance Frequency from the current 18 month Surveillance Frequency (i.e., a maximum of 22.5 months accounting for the allowable grace period specified in current Specification 4.0.2 and proposed SR 3.0.2) to a 24 month Surveillance Frequency (i.e., a maximum of 30 months accounting for the allowable grace period specified in current Specification 4.0.2 and proposed SR 3.0.2). This proposed change was evaluated in accordance with the guidance provided in NRC Generic Letter No. 91-04, "Changes in Technical Specification Surveillance Intervals to Accommodate a 24-Month Fuel Cycle," dated April 2, 1991. Reviews of historical maintenance and surveillance data have shown that these tests normally pass their Surveillances at the current Frequency. An evaluation has been performed using this data, and it has been determined that the effect on safety due to the extended Surveillance Frequency will be minimal. This conclusion is based on the following evaluation. The increased interval between SR performances is acceptable because RCIC is not a system that is taken credit for in the safety analysis. Additionally, the functions performed by RCIC can be performed by HPCS, and Technical Specifications do not permit HPCS and RCIC to be inoperable concurrently. Therefore, the impact of this change, if any, on system availability is minimal. In addition to the low pressure flow test for which the frequency is being extended, ASME Section XI inservice testing program and SR 3.5.3.3 will still require that RCIC is tested every 3 months to ensure required flow at normal operating pressure. Although conducted at normal operating pressure, this test would detect significant failures of the RCIC turbine or pump that could lead to the failure of the RCIC system to perform its safety function at low reactor pressures. The review of historical surveillance data also demonstrated that there are no failures that would invalidate the conclusion that the impact, if any, on system availability is minimal from a change to CTS 4.7.3.c.1 and 4.7.3.c.2 as implemented in SR 3.5.3.4 and SR 3.5.3.5. In addition, the proposed 24 month Surveillance Frequencies, if performed at the maximum interval allowed by proposed SR 3.0.2 (30 months) do not invalidate any assumptions in the plant licensing basis.

TECHNICAL CHANGES - LESS RESTRICTIVE (continued)

"Specific"

L.1 The phrase "actual or," in reference to the automatic initiation signal, has been added to CTS 4.7.3.c.1 (proposed SR 3.5.3.5), the Surveillance Requirement that verifies the RCIC System actuates on a "simulated" automatic initiation signal. This allows satisfactory "actual" automatic system initiations to be used to fulfill the Surveillance Requirements. OPERABILITY is adequately demonstrated in either case since the RCIC System itself cannot discriminate between "actual" or "simulated" signals.

L.2 The CTS requirements for performing a Channel Functional Test (CTS 4.7.3.a.2), a Channel Calibration (CTS 4.7.3.c.3), and the associated Actions for inoperable instrumentation (CTS 3.7.3, Action a), on the RCIC discharge line keep fill instrumentation, are being deleted. These requirements, do not necessarily relate directly to RCIC Operability. The BWR Standard Technical Specifications, NUREG-1434, Rev. 1, does not specify alarm-only equipment to be Operable to support Operability of a system or component. Control of the availability of, and necessary compensatory activities if not available for alarms, are addressed by plant operational procedures and policies. The requirements of proposed LCO 3.5.3 and SR 3.5.3.1 will ensure the RCIC pump is maintained Operable and its discharge line filled. In addition, 10 CFR 50, Appendix B, Part XII requires that measuring devices used in activities affecting quality are properly controlled, calibrated, and adjusted to maintain accuracy within necessary limits. LaSalle Units 1 and 2 are required to comply with 10 CFR 50, thus if instrumentation is used to comply with proposed SR 3.5.3.1, it would be required to meet the 10 CFR 50, Appendix B, Part XII requirements. Therefore, this instrumentation, along with the supporting Surveillances, are proposed to be deleted.

RELOCATED SPECIFICATIONS

None

DISCUSSION OF CHANGES
ITS: SECTION 3.5 - ECCS AND RCIC SYSTEM

The Bases of the current Technical Specifications for this section (pages B 3/4.5-1 through B 3/4.5-4) have been completely replaced by revised Bases that reflect the format and applicable content of the LaSalle 1 and 2 ITS Section 3.5, consistent with the BWR ISTS, NUREG-1434, Rev. 1. The revised Bases are as shown in the LaSalle 1 and 2 ITS Bases.

<CTS>

3.5 EMERGENCY CORE COOLING SYSTEMS (ECCS) AND REACTOR CORE ISOLATION COOLING (RCIC) SYSTEM

3.5.1 ECCS—Operating

<LCO 3.5.1> LCO 3.5.1 Each ECCS injection/spray subsystem and the Automatic Depressurization System (ADS) function of ~~[eight]~~ SIX safety/relief valves shall be OPERABLE. 1

<DOC L.6> INSERT A from page 3.5-4 9

<APPL 3.5.1> <3.5.1 footnotes> APPLICABILITY: MODE 1,
MODES 2 and 3, except ADS valves are not required to be OPERABLE with reactor steam dome pressure ≤ ~~[150]~~ psig. 1

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
<3.5.1 Act a.1> <3.5.1 Act a.2> <3.5.1 Act b.1> A. One low pressure ECCS injection/spray subsystem inoperable.	A.1 Restore low pressure ECCS injection/spray subsystem to OPERABLE status.	7 days
<3.5.1 Act c.1> B. High Pressure Core Spray (HPCS) System inoperable.	B.1 Verify by administrative means RCIC System is OPERABLE when RCIC is required to be OPERABLE.	~~1 hour~~ Immediately TSTF-301
	AND	
	B.2 Restore HPCS System to OPERABLE status.	14 days

(continued)

<CTS>

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
<3.5.1 Act a.3> <3.5.1 Act b.2> <3.5.1 Act d.1> <3.5.1 Act d.2> C. Two low pressure ECCS injection/spray subsystems inoperable. ~~OR One ECCS injection and one ECCS spray subsystem inoperable.~~	C.1 Restore one low pressure ECCS injection/spray subsystem to OPERABLE status.	72 hours 8
<3.5.1 Act a.4> <3.5.1 Act b.3> <3.5.1 Act c.2> <3.5.1 Act d.3> ~~D~~ E 10 Required Action and associated Completion Time of Condition A, B, or C not met.	~~D~~ E.1 Be in MODE 3. AND ~~D~~ E.2 Be in MODE 4. 10	12 hours 36 hours
<3.5.1 Act e.1> ~~E~~ F 10 One required ADS valve inoperable. 3	~~E~~ F.1 10 Restore required ADS valve to OPERABLE status. 3	14 days
~~F. One ADS valve inoperable. AND One low pressure ECCS injection/spray subsystem inoperable.~~	~~F.1 Restore ADS valve to OPERABLE status. OR F.2 Restore low pressure ECCS injection/spray subsystem to OPERABLE status.~~	~~72 hours~~ ~~72 hours~~ 7
<3.5.1 Act e.1> <3.5.1 Act e.2> G. Two or more required ADS valves inoperable. 3 OR	G.1 Be in MODE 3. AND	12 hours
		(continued)

more from next page 4

D. ADS accumulator backup compressed gas system bottle pressure < 500 psig.	D.1 Restore ADS accumulator backup compressed gas system bottle pressure ≥ 500 psig. OR D.2 Declare associated ADS valves inoperable.	72 hours 72 hours

10

<CTS>

MOVE TO PREVIOUS PAGE 4

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
<3.5.1 Act e.1> <3.5.1 Act e.2> G. (continued) Required Action and associated Completion Time of Condition ~~E or F~~ F not met.	G.2 Reduce reactor steam dome pressure to ≤ [150] psig. 1	36 hours
<DOC A.3> H. HPCS and ~~low pressure core spray (LPCS)~~ one or more low pressure ECCS injection/spray subsystems 8 inoperable. OR Three or more ECCS injection/spray subsystems inoperable. ~~OR~~ ~~HPCS System and one or more ADS valves inoperable.~~ 7 OR ~~Two~~ One 7 or more ECCS injection/spray subsystems and one or more required 3 ADS valves inoperable.	H.1 Enter LCO 3.0.3.	Immediately

<CTS>

SURVEILLANCE REQUIREMENTS

	SURVEILLANCE		FREQUENCY	
<4.5.1.a.1>	SR 3.5.1.1	Verify, for each ECCS injection/spray subsystem, the piping is filled with water from the pump discharge valve to the injection valve.	31 days	
<4.5.1.a.3> INSERT A move to page 3.5-1	SR 3.5.1.2	-------------------NOTE------------------- Low pressure coolant injection (LPCI) subsystems may be considered OPERABLE during alignment and operation for decay heat removal with reactor ~~steam dome~~ vessel pressure less than the residual heat removal cut-in permissive pressure in MODE 3, if capable of being manually realigned and not otherwise inoperable. 2 --		9
		Verify each ECCS injection/spray subsystem manual, power operated, and automatic valve in the flow path, that is not locked, sealed, or otherwise secured in position, is in the correct position.	31 days	
<4.5.1.d.1.a>	SR 3.5.1.3	Verify ADS ~~air receiver~~ accumulator supply header pressure is ≥ 150 psig.	31 days	1
<4.5.1.d.1.b>	SR 3.5.1.4	Verify ADS accumulator backup compressed gas system bottle pressure is ≥ 500 psig.	31 days	10

(continued)

<CTS>

SURVEILLANCE REQUIREMENTS (continued)

	SURVEILLANCE	FREQUENCY
<4.5.1.b>	SR 3.5.1.~~4~~ 5 [10] Verify each ECCS pump develops the specified flow rate ~~against a system head corresponding to the specified reactor pressure~~ the specified test line. SYSTEM / FLOW RATE / Test Line ~~[SYSTEM HEAD CORRESPONDING TO A REACTOR PRESSURE OF]~~ LPCS ≥ 6350 ~~[7115]~~ gpm ≥ ~~[290]~~ psig LPCI ≥ 7200 ~~[7450]~~ gpm ≥ 130 ~~[125]~~ psig HPCS (Unit 1) ≥ ~~[7115]~~ gpm ≥ 370 ~~[445]~~ psig HPCS (Unit 2) ≥ 6200 gpm 6250 ≥ 330 ~~psig~~	~~[~~In accordance with the Inservice Testing Program ~~or 92 days]~~ [1]
<4.5.1.c.1>	SR 3.5.1.~~5~~ 6 [10] ------NOTE------ Vessel injection/spray may be excluded. ------ Verify each ECCS injection/spray subsystem actuates on an actual or simulated automatic initiation signal.	~~[18]~~ 24 months [1]
<4.5.1.d.2.a>	SR 3.5.1.~~6~~ 7 [10] ------NOTE------ Valve actuation may be excluded. ------ Verify the ADS actuates on an actual or simulated automatic initiation signal.	~~[18]~~ 24 months [1]

(continued)

<CTS>

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR ~~3.5.1.7~~ 8 10 <4.5.1.d.2.b>	~~NOTE~~ ~~Not required to be performed until 12 hours after reactor steam pressure and flow are adequate to perform the test.~~ 6 Verify each required ADS valve opens when manually actuated. 3	~~18~~ 24 months on a STAGGERED TEST BASIS for each valve solenoid 1
SR 3.5.1.9 <4.5.3.3>	NOTE Instrumentation response time may be assumed to be the design instrumentation response time. Verify the ECCS RESPONSE TIME for each ECCS injection / spray subsystem is within limits.	24 months 5

JUSTIFICATION FOR DEVIATIONS FROM NUREG-1434, REVISION 1
ITS: 3.5.1 - ECCS — OPERATING

1. The brackets have been removed and the proper plant specific information/value has been provided.

2. Changes have been made (additions, deletions, and/or changes to the NUREG) to reflect the plant specific nomenclature, number, reference, system description, analysis description, or licensing basis description.

3. The words "required" has been added consistent with its use throughout the ITS (only six of the seven installed ADS valves are required).

4. Change made to be consistent with the Writer's Guide.

5. A new Surveillance Requirement has been added to ITS 3.5.1. ITS SR 3.5.1.8 requires the ECCS RESPONSE TIME to be verified to be within the assumed limit. This SR is currently in the Instrumentation Section as ISTS SR 3.3.5.1.7. However, the instrumentation for the ECCS trip functions are currently exempt from response time testing. Deletion of the response time testing for these instruments was approved by the NRC in Amendment Numbers 114 and 99 for LaSalle Units 1 and 2, respectively. Therefore, since the current SR now essentially exempts the instrumentation portion from the test, it is more appropriately located in the system Specification; i.e., in ITS 3.5.1.

6. Experience at LaSalle 1 and 2 indicates that cycling the S/RVs under steam flow conditions actually causes damage to the S/RV valve seats and results in valve leakage. By testing the S/RVs when the plant is shutdown and at low pressure it is possible to slowly close the S/RVs and prevent this damage. As a result, LaSalle 1 and 2 have discontinued the practice of cycling S/RVs at elevated reactor pressures. Therefore, the ISTS SR 3.5.1.7 Note which allows this test to be delayed until 12 hours after reactor steam pressure and flow are adequate to perform the test has been deleted.

7. ISTS 3.5.1 ACTION F is omitted. The Bases indicate this Condition is based on the capability of the remaining ECCS to ensure adequate core cooling. The current fuel vendor analysis does not provide confirmation of this capability with only five ADS valves combined with the inoperability of a low pressure ECCS subsystem. Therefore, the Condition is not allowed. This is consistent with the CTS. Subsequent Conditions have been renumbered. As a result, ISTS 3.5.1 Condition H has been revised to include the combination of one or more ECCS injection/spray subsystems and one or more required ADS valves inoperable. Since this new Condition now covers HPCS and one required ADS valve, the specific Condition has been deleted.

8. The current LaSalle 1 and 2 licensing basis does not include Technical Specification requirements which allow continued operation with the HPCS ECCS subsystem inoperable coincident with one LPCI ECCS subsystem inoperable. The plant-specific LOCA analyses demonstrate adequate core cooling assuming potentially limiting single failures of the ECCS. Since this combination of inoperabilities represents an additional failure beyond the current analysis basis, Condition C of ISTS 3.5.1 has been revised to apply only to the low pressure ECCS subsystems. This combination of inoperabilities represents the single failure of the Division 1 diesel generator. As a result, ISTS 3.5.1 Condition H has been revised to include the combination of HPCS inoperable with any one or more of the low pressure ECCS subsystems inoperable as a condition for which immediate shutdown is required. This is consistent with CTS.

9. ISTS SR 3.5.1.2, the verification of proper valve alignment SR, has a Note that allows the LPCI subsystems to be considered Operable during alignment and operation for decay heat removal with reactor steam dome pressure less than the residual heat removal cut-in permissive pressure in Mode 3, if capable of being manually realigned and not otherwise inoperable. A similar Note is not placed above other SRs that are not met when an RHR subsystem is aligned in the shutdown cooling mode; specifically ISTS SR 3.5.1.5, the automatic actuation test, and ISTS SR 3.3.5.1.7, the ECCS Response Time test. Since the intent of the ISTS Note was to allow LPCI subsystems to be considered Operable during this condition, the ISTS Note has been moved to the LCO section of the ITS to ensure it applies to all Surveillances. This location is also consistent with similar Note allowances in the other ISTS RHR shutdown cooling Specifications (e.g., ISTS 3.4.9 and ISTS 3.4.10). Without this change, it would be interpreted that, even though the Note to ISTS SR 3.5.1.2 allows LPCI subsystems to be considered Operable during this alignment, the other two SRs do not have a similar Note, thus the affected LPCI subsystems would have to be declared inoperable due to the failure to meet the other two Srs.

10. CTS 3.5.1, Surveillance Requirement 4.5.1.d.1.b requires verification of ADS accumulator backup compressed gas system bottle pressure. With ADS accumulator backup compressed gas system bottle pressure less than the specified limit, CTS 3.5.1, Action k requires that bottle pressure be restored within 72 hours, or that the associated ADS valves be declared inoperable and appropriate Actions for the inoperable ADS valves be taken. The ISTS does not describe operability requirements for the ADS accumulator backup compressed gas system. As a result, proposed ITS 3.5.1 Required Actions D.1 and D.2 and proposed SR 3.5.1.4 have been added to address ADS accumulator backup compressed gas system bottle pressure. This change is consistent with the CTS. Subsequent Conditions and Surveillance Requirements have been renumbered.

<CTS>

3.5 EMERGENCY CORE COOLING SYSTEMS (ECCS) AND REACTOR CORE ISOLATION COOLING (RCIC) SYSTEM

3.5.2 ECCS—Shutdown

<LCO 3.5.2> LCO 3.5.2 Two ECCS injection/spray subsystems shall be OPERABLE.

<Doc L.4> ←<INSERT A from page 3.5-9> [4]

<Appl 3.5.2> <Appl 3.5.3> APPLICABILITY: MODE 4,
MODE 5 except with the spent fuel storage ~~upper containment [cavity to dryer]~~ pool ~~[gate]~~ removed and water level ≥ ~~[~~22 ft ~~[8 inches]~~ over the top of the reactor pressure vessel flange. [1]

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
<3.5.2 Act a> A. One required ECCS injection/spray subsystem inoperable.	A.1 Restore required ECCS injection/spray subsystem to OPERABLE status.	4 hours
<3.5.2 Act a> B. Required Action and associated Completion Time of Condition A not met.	B.1 Initiate action to suspend operations with a potential for draining the reactor vessel (OPDRVs).	Immediately
<3.5.2 Act b> <3.5.3 Act b> C. Two required ECCS injection/spray subsystems inoperable.	C.1 Initiate action to suspend OPDRVs. AND C.2 Restore one required [5] ECCS injection/spray subsystem to OPERABLE status.	Immediately 4 hours

(continued)

<CTS>

ACTIONS (continued)

	CONDITION	REQUIRED ACTION		COMPLETION TIME
<3.5.2 Act b> <3.5.3 Act b>	D. Required Action C.2 and associated Completion Time not met.	D.1	Initiate action to restore [secondary containment] to OPERABLE status. $\boxed{1}$	Immediately
		AND		
		D.2	Initiate action to restore one standby gas treatment subsystem to OPERABLE status.	Immediately $\boxed{1}$
		AND		
		D.3	Initiate action to restore isolation capability in each required [secondary containment] $\boxed{1}$ penetration flow path not isolated.	Immediately

SURVEILLANCE REQUIREMENTS

	SURVEILLANCE		FREQUENCY
<LCO 3.5.3.b> <4.5.3.1.a.2>	SR 3.5.2.1	Verify, for each required low pressure ECCS injection/spray subsystem, the suppression pool water level is ≥ [12.67 ft]. 12 ft 7 in $\boxed{1}$	12 hours

(continued)

<CTS>

SURVEILLANCE REQUIREMENTS (continued)

	SURVEILLANCE	FREQUENCY
<LCO 3.5.3.b> <4.5.3.1.a.2> SR 3.5.2.2	Verify, for the required High Pressure Core Spray (HPCS) System, the ~~:~~ ~~a.~~ Suppression pool water level is ≥ ~~[12.67 ft]~~ ~~; or~~ -12 ft 7 in [1] ~~b. Condensate storage tank water level is ≥ [18 ft].~~ [2]	12 hours
<4.5.2.1> SR 3.5.2.3	Verify, for each required ECCS injection/spray subsystem, the piping is filled with water from the pump discharge valve to the injection valve.	31 days
<4.5.2.1> SR 3.5.2.4	-----NOTE----- One low pressure coolant injection (LPCI) subsystem may be considered OPERABLE during alignment and operation for decay heat removal, if capable of being manually realigned and not otherwise inoperable. ----- INSERT A move to page 3.5-7 [4] Verify each required ECCS injection/spray subsystem manual, power operated, and automatic valve in the flow path, that is not locked, sealed, or otherwise secured in position, is in the correct position.	31 days

(continued)

<CTS>

SURVEILLANCE REQUIREMENTS (continued)

	SURVEILLANCE	FREQUENCY
<4.5.2.1> SR 3.5.2.5	Verify each required ECCS pump develops the specified flow rate ~~[against a system head corresponding to the specified reactor pressure]~~. the specified test line TEST LINE ~~SYSTEM HEAD CORRESPONDING TO A REACTOR~~ SYSTEM / FLOW RATE / PRESSURE ~~OF~~ LPCS ≥ ~~[7115]~~ 6350 gpm ≥ ~~[290]~~ psig LPCI ≥ ~~[7450]~~ 7200 gpm ≥ ~~[125]~~ 130 psig HPCS (unit 1) ≥ ~~[7115]~~ 6250 gpm ≥ ~~[445]~~ 370 psig HPCS (unit 2) ≥ 6200 gpm ≥ 330 psig	In accordance with the Inservice Testing Program ~~or 92 days~~ [1]
<4.5.2.1> SR 3.5.2.6	——NOTE—— Vessel injection/spray ~~may~~ be excluded. ———— Verify each required ECCS injection/spray subsystem actuates on an actual or simulated automatic initiation signal.	~~[18]~~ 24 months [1]
SR 3.5.2.7	——NOTE—— Instrumentation response time may be assumed to be the design instrumentation response time. ———— Verify the ECCS RESPONSE TIME for each required ECCS injection/spray subsystem is within limits.	24 months [3]

1. The brackets have been removed and the proper plant specific information/value has been provided.

2. Changes have been made (additions, deletions, and/or changes to the NUREG) to reflect the plant specific nomenclature, number, reference, system description, or analysis description. Specifically, the HPCS System only takes suction from the suppression pool.

3. A new Surveillance Requirement has been added to ITS 3.5.2. ITS SR 3.5.2.7 requires the ECCS RESPONSE TIME to be verified to be within the assumed limit. This SR is currently in the Instrumentation Section as ISTS SR 3.3.5.1.7. However, the instrumentation for the ECCS trip functions are currently exempt from response time testing. Deletion of the response time testing for these instruments was approved by the NRC in Amendment Number 114 and 99 for LaSalle Units 1 and 2, respectively. Therefore, since the current SR now essentially exempts the instrumentation portion from the test, it is more appropriately located in the system Specification; i.e., in ITS 3.5.2.

4. ISTS SR 3.5.2.4, the verification of proper valve alignment SR, has a Note that allows one LPCI subsystem to be considered Operable during alignment and operation for decay heat removal, if capable of being manually realigned and not otherwise inoperable. A similar Note is not placed above other SRs that are not met when an RHR subsystem is aligned in the shutdown cooling mode; specifically ISTS SR 3.5.2.6, the automatic actuation test, and ISTS SR 3.3.5.1.7, the ECCS Response Time test. Since the intent of the ISTS Note was to allow one LPCI subsystem to be considered Operable during this condition, the ISTS Note has been moved to the LCO section of the ITS to ensure it applies to all Surveillances. This location is also consistent with similar Note allowances in other ISTS RHR shutdown cooling Specifications (e.g., ISTS 3.4.9 and ISTS 3.4.10). Without this change, it could be interpreted that, even though the Note to ISTS SR 3.5.2.4 allows one LPCI subsystem to be considered Operable during this alignment, the other two SRs do not have a similar Note, thus the affected LPCI subsystem would have to be declared inoperable due to the failure to meet the other two Srs.

5. The word "required" has been added consistent with its use throughout the ITS (not all ECCS subsystems are required in MODES 4 and 5).

<CTS>

3.5 EMERGENCY CORE COOLING SYSTEMS (ECCS) AND REACTOR CORE ISOLATION COOLING (RCIC) SYSTEM

3.5.3 RCIC System

<LCO 3.7.3> LCO 3.5.3 The RCIC System shall be OPERABLE.

<Appl 3.7.3> APPLICABILITY: MODE 1,
MODES 2 and 3 with reactor steam dome pressure > [150] psig. 1

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
<3.7.3 Act b> A. RCIC System inoperable.	A.1 Verify by administrative means High Pressure Core Spray System is OPERABLE. AND A.2 Restore RCIC System to OPERABLE status.	~~1 hour~~ Immediately TSTF-301 14 days
<3.7.3 Act b> B. Required Action and associated Completion Time not met.	B.1 Be in MODE 3. AND B.2 Reduce reactor steam dome pressure to ≤ [150] psig. 1	12 hours 36 hours

<CTS>

SURVEILLANCE REQUIREMENTS

	SURVEILLANCE		FREQUENCY
<4.7.3.a.1>	SR 3.5.3.1	Verify the RCIC System piping is filled with water from the pump discharge valve to the injection valve.	31 days
<4.7.3.a.3> <4.7.3.a.4>	SR 3.5.3.2	Verify each RCIC System manual, power operated, and automatic valve in the flow path, that is not locked, sealed, or otherwise secured in position, is in the correct position.	31 days
<4.7.3.b>	SR 3.5.3.3	------------------NOTE------------------ Not required to be performed until 12 hours after reactor steam pressure and flow are adequate to perform the test. -- Verify, with ~~[RCIC steam supply~~ reactor pressure~~]~~ ≤ ~~[1045]~~ 1020 psig and ≥ ~~[945]~~ 920 psig, the RCIC pump can develop a flow rate ≥ ~~[800]~~ 600 gpm ~~[~~against a system head corresponding to reactor pressure~~]~~.	92 days $\boxed{1}$
<4.7.3.c.2>	SR 3.5.3.4	------------------NOTE------------------ Not required to be performed until 12 hours after reactor steam pressure and flow are adequate to perform the test. -- Verify, with ~~[RCIC steam supply~~ reactor pressure~~]~~ ≤ ~~[~~165~~]~~ psig, the RCIC pump can develop a flow rate ≥ ~~[800]~~ 600 gpm ~~[~~against a system head corresponding to reactor pressure~~]~~.	~~[18]~~ 24 months $\boxed{1}$

(continued)

<CTS>

SURVEILLANCE REQUIREMENTS (continued)

	SURVEILLANCE	FREQUENCY
<4.7.3.c.1> SR 3.5.3.5	-----NOTE----- Vessel injection may be excluded. ----- Verify the RCIC System actuates on an actual or simulated automatic initiation signal.	~~[18]~~ 24 months $\boxed{1}$

JUSTIFICATION FOR DEVIATIONS FROM NUREG-1434, REVISION 1
ITS: 3.5.3 - RCIC SYSTEM

1. The brackets have been removed and the proper plant specific information/value has been provided.

B 3.5 EMERGENCY CORE COOLING SYSTEMS (ECCS) AND REACTOR CORE ISOLATION COOLING (RCIC) SYSTEM

B 3.5.1 ECCS—Operating

BASES

BACKGROUND

The ECCS is designed, in conjunction with the primary and secondary containment, to limit the release of radioactive materials to the environment following a loss of coolant accident (LOCA). The ECCS uses two independent methods (flooding and spraying) to cool the core during a LOCA. The ECCS network is composed of the High Pressure Core Spray (HPCS) System, the Low Pressure Core Spray (LPCS) System, and the low pressure coolant injection (LPCI) mode of the Residual Heat Removal (RHR) System. The ECCS also consists of the Automatic Depressurization System (ADS). The suppression pool provides the required source of water for the ECCS. ~~Although no credit is taken in the safety analyses for the condensate storage tank (CST), it is capable of providing a source of water for the HPCS System.~~ 1

On receipt of an initiation signal, ECCS pumps automatically start; ~~simultaneously~~ the system aligns, and the pumps inject water, taken ~~either~~ from the ~~CST or~~ suppression pool, into the Reactor Coolant System (RCS) as RCS pressure is overcome by the discharge pressure of the ECCS pumps. 1 Although the system is initiated, ADS action is delayed, allowing the operator to interrupt the timed sequence if the system is not needed. The HPCS pump discharge pressure almost immediately exceeds that of the RCS, and the pump injects coolant into the spray sparger above the core. If the break is small, HPCS will maintain coolant inventory, as well as vessel level, while the RCS is still pressurized. If HPCS fails, it is backed up by ADS in combination with LPCI and LPCS. In this event, the ADS timed sequence would be allowed to time out and open the selected safety/relief valves (S/RVs), depressurizing the RCS and allowing the LPCI and LPCS to overcome RCS pressure and inject coolant into the vessel. If the break is large, RCS pressure initially drops rapidly, and the LPCI and LPCS systems cool the core.

Water from the break returns to the suppression pool where it is used again and again. Water in the suppression pool is circulated through a heat exchanger cooled by the ~~Standby~~ Residual Heat Removal Service Water ~~(SWS)~~ (RHRSW) System. Depending on the location and size of the break, portions of the ECCS may be ineffective;

1

(continued)

BASES

BACKGROUND (continued)

however, the overall design is effective in cooling the core regardless of the size or location of the piping break. ~~Although no credit is taken in the safety analysis for the RCIC System, it performs a similar function as HPCS but has reduced makeup capability. Nevertheless, it will maintain inventory and cool the core, while the RCS is still pressurized, following a reactor pressure vessel (RPV) isolation.~~ 2

All ECCS subsystems are designed to ensure that no single active component failure will prevent automatic initiation and successful operation of the minimum required ECCS subsystems.

The LPCS System (Ref. 1) consists of a motor driven pump, a spray sparger above the core, piping, and valves to transfer water from the suppression pool to the sparger. The LPCS System is designed to provide cooling to the reactor core when the reactor pressure is low. Upon receipt of an initiation signal, the LPCS pump is automatically started when AC power is available. When the RPV pressure drops sufficiently, LPCS flow to the RPV begins. A full flow test line is provided to route water ~~from and~~ to the suppression pool to allow testing of the LPCS System without spraying water into the RPV. 3

LPCI is an independent operating mode of the RHR System. There are three LPCI subsystems. Each LPCI subsystem (Ref. 2) consists of a motor driven pump, piping, and valves to transfer water from the suppression pool to the core. Each LPCI subsystem has its own suction and discharge piping and separate vessel nozzle that connects with the core shroud through internal piping. The LPCI subsystems are designed to provide core cooling at low RPV pressure. Upon receipt of an initiation signal, each LPCI pump is automatically started (~~C pump immediately when AC power is available~~ and A and B pumps approximately 5 ~~25~~ seconds after AC power is available). When the RPV pressure drops sufficiently, LPCI flow to the RPV begins. RHR System valves in the LPCI flow path are automatically positioned to ensure the proper flow path for water from the suppression pool to inject into the core. A ~~discharge~~ full flow test line is provided to route water ~~from and~~ to the suppression pool to allow testing of each LPCI pump without injecting water into the RPV. 1 3 1

• (If AC power is supplied by the diesel generators, C pump starts immediately when AC power is available

(continued)

BASES

BACKGROUND (continued)

The HPCS System (Ref. 3) consists of a single motor driven pump, a spray sparger above the core, and piping and valves to transfer water from the ~~suction source~~ Suppression pool to the sparger. [1] ~~Suction piping is provided from the CST and the suppression pool. Pump suction is normally aligned to the CST source to minimize injection of suppression pool water into the RPV. However, if the CST water supply is low or the suppression pool level is high, an automatic transfer to the suppression pool water source ensures a water supply for continuous operation of the HPCS System.~~ [1] The HPCS System is designed to provide core cooling over a wide range of RPV pressures (0 psid to ~~1177~~ 1200 [1] psid, vessel to suction source). Upon receipt of an initiation signal, the HPCS pump automatically starts (when AC power is available) and valves in the flow path begin to open. Since the HPCS System is designed to operate over the full range of expected RPV pressures, HPCS flow begins as soon as the necessary valves are open. A full flow test line is provided to route water ~~from and~~ [3] to the ~~CST~~ suppression pool [1] to allow testing of the HPCS System during normal operation without spraying water into the RPV.

The ECCS pumps are provided with minimum flow bypass lines, which discharge to the suppression pool. The valves in these lines automatically open to prevent pump damage due to overheating when other discharge line valves are closed or RPV pressure is greater than the LPCS or LPCI pump discharge pressures following system initiation. To ensure rapid delivery of water to the RPV and to minimize water hammer effects, the ECCS discharge line "keep fill" systems are designed to maintain all pump discharge lines filled with water.

The ADS (Ref. 4) consists of ~~8~~ 7 of the ~~20~~ 18 S/RVs for Unit 1 and 7 of the 13 S/RVs for Unit 2. [1] It is designed to provide depressurization of the primary system during a small break LOCA if HPCS fails or is unable to maintain required water level in the RPV. ADS operation reduces the RPV pressure to within the operating pressure range of the low pressure ECCS subsystems (LPCS and LPCI), so that these subsystems can provide core cooling. ~~Each ADS valve is supplied with pneumatic power from an air storage system, which consists of air accumulators and air receivers located in the drywell.~~

[1] INSERT B3.5-3(A)

(continued)

INSERT FOR B 3.5-3(A)

The Drywell Pneumatic System discharges from the air receiver (or nitrogen receiver when the primary containment is inerted) and after filtration is divided into two supply headers, one of which supplies all the ADS accumulators with approximately 175 psig air (or nitrogen). There is a check valve between each ADS accumulator and the supply. Drywell Pneumatic System low header pressure and high ADS pressure are alarmed in the control room.

The accumulators for the ADS valves are normally maintained by the Drywell Pneumatic System compressors. There are two full-capacity compressors which cycle as needed to maintain pressure in the drywell pneumatic receiver tank. Nitrogen bottle banks provide a backup source to maintain the ADS accumulators charged following isolation of the normal pneumatic supply. Each ADS accumulator is provided with a pressure switch to detect low pressure (< 150 psig). These pressure switches are provided with alarms in the control room. A control room alarm is also annunciated for low pressure in the ADS nitrogen bottle supply headers.

BASES (continued)

APPLICABLE SAFETY ANALYSES

The ECCS performance is evaluated for the entire spectrum of break sizes for a postulated LOCA. The accidents for which ECCS operation is required are presented in References 5, 6, and 7. The required analyses and assumptions are defined in 10 CFR 50 (Ref. 8), and the results of these analyses are described in Reference 9.

This LCO helps to ensure that the following acceptance criteria for the ECCS, established by 10 CFR 50.46 (Ref. 10), will be met following a LOCA assuming the worst case single active component failure in the ECCS:

a. Maximum fuel element cladding temperature is ≤ 2200°F;

b. Maximum cladding oxidation is ≤ 0.17 times the total cladding thickness before oxidation;

c. Maximum hydrogen generation from zirconium water reaction is ≤ 0.01 times the hypothetical amount that would be generated if all of the metal in the cladding surrounding the fuel, excluding the cladding surrounding the plenum volume, were to react;

d. The core is maintained in a coolable geometry; and

e. Adequate long term cooling capability is maintained.

The limiting single failures are discussed in Reference 11. For ~~a large break LOCA~~ the LOCA evaluation model which covers the entire spectrum of break sizes (large breaks to small breaks), failure of ECCS subsystems in the HPCS Division ~~1 (LPCS and LPCI-A) or Division 2 (LPCI-B and LPCI-C)~~ due to failure of its associated diesel generator is, in general, the most severe failure. ~~For a small break LOCA, HPCS System failure is the most severe failure. One ADS valve failure is analyzed as a limiting single failure for events requiring ADS operation.~~ The remaining OPERABLE ECCS subsystems, which include one spray subsystem, provide the capability to adequately cool the core and prevent excessive fuel damage under near-term and long-term conditions. 3 1

The ECCS satisfy Criterion 3 of ~~the NRC Policy Statement~~ 10 CFR 50.36(c)(2)(ii).

LCO

Each ECCS injection/spray subsystem and ~~eight~~ six ADS valves are required to be OPERABLE. The ECCS injection/spray subsystems are defined as the three LPCI subsystems, the LPCS System, and the HPCS System. The low pressure ECCS 4

(continued)

For all LOCA analyses, only six ADS valves are assumed to function. An evaluation has been performed which assumes five ADS valves function, however in this analysis all low pressure and high pressure ECCS subsystems are also assumed to be available. This event is considered bounded by other single failure analysis.

BASES

LCO (continued)

injection/spray subsystems are defined as the LPCS System and the three LPCI subsystems.

With less than the required number of ECCS subsystems OPERABLE during a limiting design basis LOCA concurrent with the worst case single failure, the limits specified in 10 CFR 50.46 (Ref. 10) could potentially be exceeded. All ECCS subsystems must therefore be OPERABLE to satisfy the single failure criterion required by 10 CFR 50.46 (Ref. 10).

As noted, LPCI subsystems 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. Alignment and operation for decay heat removal includes when the required RHR pump is not operating or when the system is realigned from or to the RHR shutdown cooling mode. This allowance is necessary since the RHR System may be required to operate in the shutdown cooling mode to remove decay heat and sensible heat from the reactor. At these low pressures and decay heat levels, a reduced complement of ECCS subsystems should provide the required core cooling, thereby allowing operation of RHR shutdown cooling when necessary. [4] [3]

APPLICABILITY

All ECCS subsystems are required to be OPERABLE during MODES 1, 2, and 3 when there is considerable energy in the reactor core and core cooling would be required to prevent fuel damage in the event of a break in the primary system piping. In MODES 2 and 3, the ADS function is not required when pressure is ≤ 150 psig because the low pressure ECCS subsystems (LPCS and LPCI) are capable of providing flow into the RPV below this pressure. ECCS requirements for MODES 4 and 5 are specified in LCO 3.5.2, "ECCS—Shutdown."

ACTIONS

A.1

If any one low pressure ECCS injection/spray subsystem is inoperable, the inoperable subsystem must be restored to OPERABLE status within 7 days. In this ~~C~~condition, the remaining OPERABLE subsystems provide adequate core cooling during a LOCA. However, overall ECCS reliability is reduced because a single failure in one of the remaining OPERABLE subsystems concurrent with a LOCA may result in the ECCS not being able to perform its intended safety function. The 7 day Completion Time is based on a reliability study (Ref. 12) that evaluated the impact on ECCS availability by assuming that various components and subsystems were taken [3]

(continued)

BASES

ACTIONS

A.1 (continued)

out of service. The results were used to calculate the average availability of ECCS equipment needed to mitigate the consequences of a LOCA as a function of allowed outage times (i.e., Completion Times).

B.1 and B.2

Immediately 4

If the HPCS System is inoperable, and the RCIC System is verified to be OPERABLE (when RCIC is required to be OPERABLE), the HPCS System must be restored to OPERABLE status within 14 days. In this ~~C~~condition, adequate core 3 cooling is ensured by the OPERABILITY of the redundant and diverse low pressure ECCS injection/spray subsystems in conjunction with the ADS. Also, the RCIC System will automatically provide makeup water at most reactor operating pressures. Verification of RCIC OPERABILITY ~~within 1 hour~~ immediately TSTF-301 is therefore required when HPCS is inoperable and RCIC is required to be OPERABLE. This may be performed by an administrative check, by examining logs or other information 3 to determine if RCIC is out of service for maintenance or other reasons. It is not necessary to perform the Surveillances needed to demonstrate the OPERABILITY of the RCIC System. However, if the OPERABILITY of the RCIC System cannot be verified and RCIC is required to be OPERABLE, Condition D must be ~~immediately~~ entered. If a single active 3 component fails concurrent with a design basis LOCA, there is a potential, depending on the specific failure, that the minimum required ECCS equipment will not be available. A 14 day Completion Time is based on the results of a reliability study (Ref. 12) and has been found to be acceptable through operating experience.

Immediate 4 Immediately

C.1

the low pressure 4 LPCS

With two ECCS injection subsystems inoperable or one ECCS injection and one ECCS spray subsystem inoperable, at least one ECCS injection/spray subsystem must be restored to OPERABLE status within 72 hours. In this ~~C~~condition, the 3 remaining OPERABLE subsystems provide adequate core cooling during a LOCA. However, overall ECCS reliability is reduced in this Condition because a single failure in one of the remaining OPERABLE subsystems concurrent with a design basis

(continued)

BASES

ACTIONS

C.1 (continued)

LOCA may result in the ECCS not being able to perform its intended safety function. Since the ECCS availability is reduced relative to Condition A, a more restrictive Completion Time is imposed. The 72 hour Completion Time is based on a reliability study, as provided in Reference 12.

Insert B D.1 and D.2 4

~~D~~.1 and ~~D~~.2 E E 4

If any Required Action and associated Completion Time of Condition A, B, or C are not met, the plant must be brought to a MODE in which the LCO does not apply. To achieve this status, the plant must be brought to at least MODE 3 within 12 hours and to MODE 4 within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

F ~~E~~.1 4

six 4 11 1 one required 4

The LCO requires ~~eight~~ ADS valves to be OPERABLE to provide the ADS function. Reference ~~13~~ contains the results of an evaluation of ~~analysis that evaluated~~ the effect of ~~one~~ ADS valve being out of service. Per this evaluation ~~analysis~~, operation of only ~~seven~~ five 4 ADS valves will provide the required depressurization. However, overall reliability of the ADS is reduced because a single failure in the OPERABLE ADS valves could result in a reduction in depressurization capability. Therefore, operation is only allowed for a limited time. The 14 day Completion Time is based on a reliability study (Ref. 12) and has been found to be acceptable through operating experience.

3

~~F.1 and F.2~~

~~If any one low pressure ECCS injection/spray subsystem is inoperable in addition to one inoperable ADS valve, adequate core cooling is ensured by the OPERABILITY of HPCS and the remaining low pressure ECCS injection/spray subsystems. However, the overall ECCS reliability is reduced because a single active component failure concurrent with a design~~ 4

(continued)

INSERT B D.1 AND D.2

With the ADS accumulator backup compressed gas system bottle pressure less than the specified limit, bottle pressure must be restored within 72 hours, or the associated ADS valves must be declared inoperable. In this condition, the remaining drywell pneumatic system and ADS accumulators are sufficient to ensure ADS valve operation. However, overall ECCS reliability is reduced in this condition because with insufficient bottle bank pressure, the capability of ADS valves to operate for long periods of time following an accident (without the drywell pneumatic system) is reduced. Each ADS valve is equipped with an individual accumulator of sufficient capacity to operate the valves in the event of a loss of air supply. The 72 hour Completion Time is based on a reliability study, as provided in Reference 12.

BASES

ACTIONS

~~F.1 and F.2 (continued)~~

~~basis LOCA could result in the minimum required ECCS equipment not being available. Since both a high pressure (ADS) and low pressure subsystem are inoperable, a more restrictive Completion Time of 72 hours is required to restore either the low pressure ECCS injection/spray subsystem or the ADS valve to OPERABLE status. This Completion Time is based on a reliability study (Ref. 12) and has been found to be acceptable through operating experience.~~ 4

G.1 and G.2

is 4 required 4

4 F If any Required Action and associated Completion Time of Condition ~~E~~ ~~or F are~~ not met or if two or more ADS valves are inoperable, the plant must be brought to a condition in which the LCO does not apply. To achieve this status, the plant must be brought to at least MODE 3 within 12 hours and reactor steam dome pressure reduced to ≤ 150 psig within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

H.1

When multiple ECCS subsystems are inoperable, as stated in Condition H, the plant is in a condition outside of the design basis ~~accident analyses~~. Therefore, LCO 3.0.3 must be entered immediately. 3

SURVEILLANCE REQUIREMENTS

SR 3.5.1.1

The flow path piping has the potential to develop voids and pockets of entrained air. Maintaining the pump discharge lines of the HPCS System, LPCS System, and LPCI subsystems full of water ensures that the systems will perform properly, injecting their full capacity into the RCS upon demand. This will also prevent a water hammer following an ECCS initiation signal. One acceptable method of ensuring the lines are full is to vent at the high points. The

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.5.1.1 (continued)

31 day Frequency is based on operating experience, on the procedural controls governing system operation, and on the gradual nature of void buildup in the ECCS piping.

SR 3.5.1.2

Verifying the correct alignment for manual, power operated, and automatic valves in the ECCS flow paths provides assurance that the proper flow paths will exist for ECCS operation. This SR does not apply to valves that are locked, sealed, or otherwise secured in position since these valves were verified to be in the correct position prior to locking, sealing, or securing. A valve that receives an initiation signal is allowed to be in a nonaccident position provided the valve will automatically reposition in the proper stroke time. This SR does not require any testing or valve manipulation; rather, it involves verification that those valves potentially capable of being mispositioned are in the correct position. This SR does not apply to valves that cannot be inadvertently misaligned, such as check valves.

The 31 day Frequency of this SR was derived from the Inservice Testing Program requirements for performing valve testing at least once every 92 days. The Frequency of 31 days is further justified because the valves are operated under procedural control and because improper valve alignment would only affect a single subsystem. This Frequency has been shown to be acceptable through operating experience.

~~This SR is modified by a Note that allows LPCI subsystems to be considered OPERABLE during alignment and operation for decay heat removal with reactor steam dome pressure less than the 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. This allows operation in the RHR shutdown cooling mode during MODE 3 if necessary.~~ 4

(continued)

BASES

SURVEILLANCE REQUIREMENTS (continued)

5

The ADS valve accumulators are sized to provide two cycles of the ADS valves upon loss of the nitrogen supply.

1

The accumulator supply header pressure verification may be accomplished by monitoring control room alarms.

SR 3.5.1.3

pneumatic

accumulator supply header

Verification every 31 days that ADS ~~air receiver~~ pressure is ≥ ~~[150]~~ psig assures adequate ~~air~~ pressure for reliable ADS operation. The accumulator on each ADS valve provides pneumatic pressure for valve actuation. ~~The designed pneumatic supply pressure requirements for the accumulator are such that, following a failure of the pneumatic supply to the accumulator, at least two valve actuations can occur with the drywell at 70% of design pressure~~ (Ref. ~~14~~ 13). The ECCS safety analysis assumes only one actuation to achieve the depressurization required for operation of the low pressure ECCS. ~~This minimum required pressure of [150] psig is provided by the ADS Instrument Air Supply System.~~ The 31 day Frequency takes into consideration ~~administrative control over operation of the Instrument Air Supply System and~~ alarms for low ~~air~~ pressure. 1

pneumatic 1

Insert B SR3.5.1.4 4

SR 3.5.1.~~4~~ 5 4

The performance requirements of the ECCS pumps are determined through application of the 10 CFR 50, Appendix K, criteria (Ref. 8). This periodic Surveillance is performed (in accordance with the ASME Code, Section XI, requirements for the ECCS pumps) to verify that the ECCS pumps will develop the flow rates required by the respective analyses. The ECCS pump flow rates ensure that adequate core cooling is provided to satisfy the acceptance criteria of 10 CFR 50.46 (Ref. 10).

1

Was determined during pre-operational testing to be

3 Under these conditions,

test line pressure 1

The pump flow rates are verified against a ~~system head~~ that ~~is~~ equivalent to the RPV pressure expected during a LOCA. The total system pump outlet pressure is adequate to overcome the elevation head pressure between the pump suction and the vessel discharge, the piping friction losses, and RPV pressure present during LOCAs. ~~These values may be established during pre-operational testing.~~ 3 ~~A 92 day~~ The Frequency for this Surveillance is in accordance with the Inservice Testing Program requirements. 4

SR 3.5.1.~~5~~ 6 4

The ECCS subsystems are required to actuate automatically to perform their design functions. This Surveillance ~~test~~ 3

(continued)

Insert B SR 3.5.1.4

SR 3.5.1.4

Verification every 31 days that ADS accumulator backup compressed gas system bottle pressure is $\geq$ 500 psig assures availability of an adequate backup pneumatic supply to the ADS accumulators following a loss of the drywell pneumatic supply. The 31 day Frequency is adequate because each ADS bottle bank is monitored by a low pressure alarm. Also, unless the normal drywell pneumatic supply is lost, the only expected losses from the bottles are due to leakage, which is minimal.

BASES

SURVEILLANCE REQUIREMENTS

SR 3.5.1.6 (continued) 4

verifies that, with a required system initiation signal (actual or simulated), the automatic initiation logic of HPCS, LPCS, and LPCI will cause the systems or subsystems to operate as designed, including actuation of the system throughout its emergency operating sequence, automatic pump startup, and actuation of all automatic valves to their required positions. This Surveillance also ensures that the ~~HPCS System~~ injection valve will automatically ~~restart~~ re-open on an RPV low water level (Level 2) signal received subsequent to an ~~RPV high water level (Level 8) trip~~ injection valve closure signal. ~~and that the suction is automatically transferred from the CST to the suppression pool.~~ The LOGIC SYSTEM FUNCTIONAL TEST performed in LCO 3.3.5.1 overlaps this Surveillance to provide complete testing of the assumed safety function. 1 3 1 1 1

The ~~18~~ 24 month Frequency is based on the need to perform this Surveillance under the conditions that apply during a plant outage and the potential for an unplanned transient if the Surveillance were performed with the reactor at power. Operating experience has shown that these components usually pass the SR when performed at the ~~18~~ month Frequency, which is based on the refueling cycle. Therefore, the Frequency was concluded to be acceptable from a reliability standpoint. 24 4 4

This SR is modified by a Note that excludes vessel injection/spray during the Surveillance. Since all active components are testable and full flow can be demonstrated by recirculation through the test line, coolant injection into the RPV is not required during the Surveillance.

SR 3.5.1.7 4

The ADS designated S/RVs are required to actuate automatically upon receipt of specific initiation signals. A system functional test is performed to demonstrate that the mechanical portions of the ADS function (i.e., solenoids) operate as designed when initiated either by an actual or simulated initiation signal, causing proper actuation of all the required components. SR 3.5.1.8 and the LOGIC SYSTEM FUNCTIONAL TEST performed in LCO 3.3.5.1 overlap this Surveillance to provide complete testing of the assumed safety function. 4

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.5.1.8 (continued) 7 4

The 18 24 month Frequency is based on the need to perform this Surveillance under the conditions that apply during a plant outage and the potential for an unplanned transient if the Surveillance were performed with the reactor at power. Operating experience has shown that these components usually pass the SR when performed at the 18 24 month Frequency, which is based on the refueling cycle. Therefore, the Frequency was concluded to be acceptable from a reliability standpoint. 4 4

This SR is modified by a Note that excludes valve actuation. This also prevents an RPV pressure blowdown. 3 Since the valves are individually tested in accordance with SR 3.5.1.8 3

SR 3.5.1.9 8 4

A manual actuation of each ADS valve, and observing the expected change in the indicated valve position, 1 is performed to verify that the valve and solenoids are functioning properly. 1 ~~and that no blockage exists in the S/RV discharge lines. This is demonstrated by the response of the turbine control or bypass valve, 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. Also, adequate steam flow must be passing through the main turbine or turbine bypass valves to continue to control reactor pressure when the ADS valves divert steam flow upon opening. Sufficient time is therefore allowed, after the required pressure and flow are achieved, to perform this test. Adequate pressure at which this test is to be performed is [950] psig (the pressure recommended by the valve manufacturer). Adequate steam flow is represented by [at least 1.25 turbine bypass valves open, or total steam flow ≥ 10^6 lb/hr]. 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. Therefore,~~ 6 ~~this SR is modified by a Note that states the Surveillance is not required to be performed until 12 hours after reactor steam pressure and flow are adequate to perform the test.~~ 4 SR 3.5.1.7 8 and the LOGIC SYSTEM FUNCTIONAL TEST performed in LCO 3.3.5.1 overlap this Surveillance to provide complete testing of the assumed safety function. 7 4

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.5.1.~~9~~ 8 4 (continued)

The Frequency of ~~18~~ 24 months on required 4 a STAGGERED TEST BASIS ensures 4 that both solenoids for each ADS valve are alternately tested. The Frequency is based on the need to perform this Surveillance under the conditions that apply just prior to or during a startup from a plant outage. Operating experience has shown that ~~these components usually pass the SR when performed at the~~ ~~18~~ ~~month Frequency, which is based~~ 24 4 on the refueling cycle. Therefore, the Frequency was concluded to be acceptable from a reliability standpoint.

4 INSERT B 3.5-13 (A)

REFERENCES

1. U FSAR, Section ~~[~~6.3.2.2.3~~]~~.
2. U FSAR, Section ~~[~~6.3.2.2.4~~]~~.
3. U FSAR, Section ~~[~~6.3.2.2.1~~]~~.
4. U FSAR, Section ~~[~~6.3.2.2.2~~]~~.
5. U FSAR, Section ~~[~~15.2.8~~]~~.
6. U FSAR, Section ~~[~~15.6.4~~]~~.
7. U FSAR, Section ~~[~~15.6.5~~]~~.
8. 10 CFR 50, Appendix K.
9. U FSAR, Section ~~[~~6.3.3~~]~~. 5
10. 10 CFR 50.46.
11. U FSAR, Section ~~[~~6.3.3.3~~]~~. 5
12. Memorandum from R.L. Baer (NRC) to V. Stello, Jr. (NRC), "Recommended Interim Revisions to LCO's for ECCS Components," December 1, 1975.

~~13. FSAR, Section [6.3.3.7.8].~~ 1

~~14~~ 13. U FSAR, Section ~~[~~7.3.1.~~1.1.4~~2~~]~~. 5

1 14. UFSAR, Table 6.3-2.
15. NEDO-32291-A, "System Analysis for the Elimination of Selected Response Time Testing Requirements," October 1995.

INSERT B 3.5-13(A)

SR 3.5.1.9

This SR ensures that the ECCS RESPONSE TIME for each ECCS injection/spray subsystem is less than or equal to the maximum value assumed in the accident analysis. Response time testing acceptance criteria are included in Reference 14. This SR is modified by a Note that allows the instrumentation portion of the response time to be assumed to be the design instrumentation response time and therefore, is excluded from the ECCS RESPONSE TIME testing. This is allowed since the instrumentation response time is a small part of the ECCS RESPONSE TIME (e.g., sufficient margin exists in the diesel generator start time when compared to the instrumentation response time) (Ref. 15).

ECCS RESPONSE TIME tests are conducted every 24 months. The 24 month Frequency is consistent with the typical industry refueling cycle and is based upon plant operating experience.

JUSTIFICATION FOR DEVIATIONS FROM NUREG-1434, REVISION 1
ITS BASES: 3.5.1 - ECCS — OPERATING

1. Changes have been made (additions, deletions, and/or changes to the NUREG) to reflect the plant specific nomenclature, number, reference, system description, or analysis description.

2. This discussion has been deleted since it discusses the RCIC System, which is not part of this LCO.

3. Editorial change made for enhanced clarity or to be consistent with similar statements in other places in the Bases.

4. Changes have been made to reflect those changes made to the Specification. The following requirements have been renumbered, where applicable, to reflect the changes.

5. The brackets have been removed and the proper plant specific information/value has been provided.

6. CTS Surveillance Requirement 4.5.1.d.2.b requires periodic manual actuation of each ADS valve to verify that the valve and solenoids are functioning properly. This verification is performed by observing the expected change in indicated valve position. ITS SR 3.5.1.7 also requires periodic manual actuation of each ADS valve to verify that the valve is functioning properly. The ISTS Bases for SR 3.5.1.7 assumes that this testing must be performed under conditions where sufficient steam flow can be assured in order to prevent S/RV seat damage. Experience at LaSalle 1 and 2 indicates that cycling the S/RVs under steam flow conditions actually causes damage to the S/RV valve seats and results in valve leakage. By testing the S/RVs when the plant is shutdown and at low pressure it is possible to slowly close the S/RVs and prevent this damage. As a result, LaSalle 1 and 2 have discontinued the practice of cycling S/RVs at elevated reactor pressures. Statements in the ISTS Bases description related to required pressure and steam flow requirements, and the verification that the S/RV discharge lines are not blocked have been removed to reflect the plant practice.

B 3.5 EMERGENCY CORE COOLING SYSTEMS (ECCS) AND REACTOR CORE ISOLATION COOLING (RCIC) SYSTEM

B 3.5.2 ECCS—Shutdown

BASES

BACKGROUND

A description of the High Pressure Core Spray (HPCS) System, Low Pressure Core Spray (LPCS) System, and low pressure coolant injection (LPCI) mode of the Residual Heat Removal (RHR) System is provided in the Bases for LCO 3.5.1, "ECCS—Operating."

APPLICABLE SAFETY ANALYSES

The ECCS performance is evaluated for the entire spectrum of break sizes for a postulated loss of coolant accident (LOCA). The long term cooling analysis following a design basis LOCA (Ref. 1) demonstrates that only one ~~low pressure~~ ECCS injection/spray subsystem is required, post LOCA, to maintain adequate reactor vessel water level in the event of an inadvertent vessel draindown. It is reasonable to assume, based on engineering judgment, that while in MODES 4 and 5, one ~~low pressure~~ ECCS injection/spray subsystem can maintain adequate reactor vessel water level. To provide redundancy, a minimum of two ~~low pressure~~ ECCS injection/spray subsystems are required to be OPERABLE in MODES 4 and 5. 1 1 1

The ECCS satisfy Criterion 3 of ~~the NRC Policy Statement~~ (10 CFR 50.36(c)(2)(ii)). 1

LCO

Two ECCS injection/spray subsystems are required to be OPERABLE. The ECCS injection/spray subsystems are defined as the three LPCI subsystems, the LPCS System, and the HPCS System. The LPCS System and each LPCI subsystem consist of one motor driven pump, piping, and valves to transfer water from the suppression pool to the RPV. The HPCS System consists of one motor driven pump, piping, and valves to transfer water from the suppression pool ~~or condensate storage tank (CST)~~ to the RPV. 1

1 The necessary portions of the Diesel Generator Cooling Water System are also required to provide appropriate cooling to each required ECCS injection/spray subsystem.

3 As noted, One LPCI subsystem (A or B) may be ~~aligned for decay heat removal in MODE 4 or 5 and~~ considered OPERABLE ~~for the ECCS function~~ if ~~it can be~~ capable of being manually realigned (remote or local) to the LPCI mode and is not otherwise inoperable. Because 2

during alignment and operation for decay heat removal

1 INSERT B 3.5-14(A)

(continued)

<u>INSERT B 3.5-14(A)</u>

Alignment and operation for decay heat removal includes when the required RHR is not operating or when the system is realigned from or to the RHR shutdown cooling mode. This allowance is necessary since the RHR System may be required to operate in the shutdown cooling mode to remove decay heat and sensible heat from the reactor.

BASES

LCO (continued)

of low pressure and low temperature conditions in MODES 4 and 5, sufficient time will be available to manually align and initiate LPCI subsystem operation to provide core cooling prior to postulated fuel uncovery.

APPLICABILITY

OPERABILITY of the ECCS injection/spray subsystems is required in MODES 4 and 5 to ensure adequate coolant inventory and sufficient heat removal capability for the irradiated fuel in the core in case of an inadvertent draindown of the vessel. Requirements for ECCS OPERABILITY during MODES 1, 2, and 3 are discussed in the Applicability section of the Bases for LCO 3.5.1. ECCS subsystems are not required to be OPERABLE during MODE 5 with the upper containment pool gate removed and the water level maintained at ≥ 22 ft ~~8 inches~~ above the RPV flange. This provides sufficient coolant inventory to allow operator action to terminate the inventory loss prior to fuel uncovery in case of an inadvertent draindown.

spent fuel storage

The Automatic Depressurization System is not required to be OPERABLE during MODES 4 and 5 because the RPV pressure is < 150 psig, and the LPCS, HPCS, and LPCI subsystems can provide core cooling without any depressurization of the primary system.

ACTIONS

A.1 and B.1

If any one required ECCS injection/spray subsystem is inoperable, the required inoperable ECCS injection/spray subsystem must be restored to OPERABLE status within 4 hours. In this Condition, the remaining OPERABLE subsystem can provide sufficient RPV flooding capability to recover from an inadvertent vessel draindown. However, overall system reliability is reduced because a single failure in the remaining OPERABLE subsystem concurrent with a vessel draindown could result in the ECCS not being able to perform its intended function. The 4 hour Completion Time for restoring the required ECCS injection/spray subsystem to OPERABLE status is based on engineering judgment that considered the availability of one subsystem and the low probability of a vessel draindown event.

(continued)

BASES

ACTIONS

A.1 and B.1 (continued)

With the inoperable subsystem not restored to OPERABLE status within the required Completion Time, action must be initiated immediately to suspend operations with a potential for draining the reactor vessel (OPDRVs) to minimize the probability of a vessel draindown and the subsequent potential for fission product release. Actions must continue until OPDRVs are suspended.

C.1, C.2, D.1, D.2, and D.3

If both of the required ECCS injection/spray subsystems are inoperable, all coolant inventory makeup capability may be unavailable. Therefore, actions must be initiated immediately to suspend OPDRVs in order to minimize the probability of a vessel draindown and the subsequent potential for fission product release. Actions must continue until OPDRVs are suspended. One required 3 ECCS injection/spray subsystem must also be restored to OPERABLE status within 4 hours. ← (MOVE FROM NEXT PAGE) 2

3 required

If at least one ECCS injection/spray subsystem is not restored to OPERABLE status within the 4 hour Completion Time, additional actions are required to minimize any potential fission product release to the environment. This includes ensuring secondary containment is OPERABLE; one standby gas treatment subsystem is OPERABLE; and secondary containment isolation capability is available (i.e., one secondary containment isolation valve and associated instrumentation are OPERABLE or other acceptable administrative controls to assure isolation capability) in each secondary containment penetration flow path not isolated that is assumed to be isolated to mitigate radioactivity releases. 2

1 The administrative controls consist of stationing a dedicated operator, who is in continuous communication with the control room, at the controls of the isolation device. In this way, the penetration can be rapidly isolated when a need for secondary containment isolation is indicated.)

This may be performed by an administrative check, by examining logs or other information, to determine if the components are out of service for maintenance or other reasons. It is not necessary to perform the Surveillances needed to demonstrate OPERABILITY of the components. If, however, any required component is inoperable, then it must be restored to OPERABLE status. In this case, the Surveillances may need to be performed to restore the component to OPERABLE status. Actions must continue until all required components are OPERABLE.

(continued)

BASES

ACTIONS

C.1, C.2, D.1, D.2, and D.3 (continued)

Move to previous page

The 4 hour Completion Time to restore at least one ECCS injection/spray subsystem to OPERABLE status ensures that prompt action will be taken to provide the required cooling capacity or to initiate actions to place the plant in a condition that minimizes any potential fission product release to the environment. 2

SURVEILLANCE REQUIREMENTS

SR 3.5.2.1 and SR 3.5.2.2

-12 ft 7 in (referenced to a plant elevation of 699 ft 11 in)

The minimum water level of ~~[12.67 ft]~~ required for the suppression pool, equivalent to a contained water volume of 70,000 ft³, is periodically verified to ensure that the suppression pool will provide adequate net positive suction head (NPSH) for the ECCS pumps, recirculation volume, and vortex prevention. With the suppression pool water level less than the required limit, all ECCS injection/spray subsystems are inoperable~~, unless they are aligned to an OPERABLE CST~~. 4 1

1

~~When the suppression pool level is < [12.67 ft], the HPCS System is considered OPERABLE only if it can take suction from the CST and the CST water level is sufficient to provide the required NPSH for the HPCS pump. Therefore, a verification that either the suppression pool water level is ≥ [12.67 ft] or the HPCS System is aligned to take suction from the CST and the CST contains ≥ [170,000] gallons of water, equivalent to 18 ft, ensures that the HPCS System can supply makeup water to the RPV.~~ 1

The 12 hour Frequency of these SRs was developed considering operating experience related to suppression pool ~~and CST~~ water level variations and instrument drift during the applicable MODES. Furthermore, the 12 hour Frequency is considered adequate in view of other indications in the control room~~, including alarms,~~ to alert the operator to an abnormal suppression pool ~~or CST~~ water level condition. 1 1 1

SR 3.5.2.3, SR 3.5.2.5, ~~and~~ SR 3.5.2.6, and SR 3.5.2.7

The Bases provided for SR 3.5.1.1, SR 3.5.1.4, ~~and~~ SR 3.5.1.5, and SR 3.5.1.8 are applicable to SR 3.5.2.3, SR 3.5.2.5, ~~and~~ SR 3.5.2.6, and SR 3.5.2.7, respectively. 3

(continued)

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.5.2.4

Verifying the correct alignment for manual, power operated, and automatic valves in the ECCS flow paths provides assurance that the proper flow paths will exist for ECCS operation. This SR does not apply to valves that are locked, sealed, or otherwise secured in position since these valves were verified to be in the correct position prior to locking, sealing, or securing. A valve that receives an initiation signal is allowed to be in a nonaccident position provided the valve will automatically reposition in the proper stroke time. This SR does not require any testing or valve manipulation; rather, it involves verification that those valves capable of potentially being mispositioned are in the correct position. This SR does not apply to valves that cannot be inadvertently misaligned, such as check valves. The 31 day Frequency is appropriate because the valves are operated under procedural control and the probability of their being mispositioned during this time period is low.

~~In MODES 4 and 5, the RHR System may operate in the shutdown cooling mode to remove decay heat and sensible heat from the reactor. Therefore, RHR valves that are required for LPCI subsystem operation may be aligned for decay heat removal. Therefore, this SR is modified by a Note that allows one LPCI subsystem of the RHR System to be considered OPERABLE for the ECCS function if all the required valves in the LPCI flow path can be manually realigned (remote or local) to allow injection into the RPV and the system is not otherwise inoperable. This will ensure adequate core cooling if an inadvertent vessel draindown should occur.~~ 3

REFERENCES

1 U

1. ~~F~~SAR, Section 6.3.3. 4

2

JUSTIFICATION FOR DEVIATIONS FROM NUREG-1434, REVISION 1
ITS BASES: 3.5.2 - ECCS — SHUTDOWN

1. Changes have been made (additions, deletions, and/or changes to the NUREG) to reflect the plant specific nomenclature, number, reference, system description, or analysis description.

2. Editorial change made for enhanced clarity or to be consistent with similar statements in other places in the Bases.

3. Changes have been made to reflect those changes made to the Specification. The following requirements have been renumbered, where applicable, to reflect the changes.

4. The brackets have been removed and the proper plant specific information/value has been provided.

B 3.5 EMERGENCY CORE COOLING SYSTEMS (ECCS) AND REACTOR CORE ISOLATION COOLING (RCIC) SYSTEM

B 3.5.3 RCIC System

BASES

BACKGROUND

The RCIC System is not part of the ECCS; however, the RCIC System is included with the ECCS section because of their similar functions.

The RCIC System is designed to operate either automatically or manually following reactor pressure vessel (RPV) isolation accompanied by a loss of coolant flow from the feedwater system to provide adequate core cooling and control of RPV water level. Under these conditions, the High Pressure Core Spray (HPCS) and RCIC systems perform similar functions. The RCIC System design requirements ensure that the criteria of Reference 1 are satisfied.

The RCIC System (Ref. 2) consists of a steam driven turbine pump unit, piping~~,~~ and valves to provide steam to the turbine, as well as piping and valves to transfer water from the suction source to the core via the ~~feedwater system~~ ~~line~~ head spray nozzle [1]. Suction piping is provided from the condensate storage tank (CST) and the suppression pool. Pump suction is normally aligned to the CST to minimize injection of suppression pool water into the RPV. However, if the CST water supply is low~~, or the suppression pool level is high,~~ [1] an automatic transfer to the suppression pool water source ensures a water supply for continuous operation of the RCIC System. The steam supply to the turbine is piped from main steam line ~~A~~ B [1], upstream of the inboard main steam line isolation valve.

The RCIC System is designed to provide core cooling for a wide range of reactor pressures, ~~[165]~~ 135 psig to ~~[1255]~~ 1185 psig. [3] Upon receipt of an initiation signal, the RCIC turbine accelerates to a specified speed. As the RCIC flow increases, the turbine control valve is automatically adjusted to maintain design flow. Exhaust steam from the RCIC turbine is discharged to the suppression pool. A full flow test line is provided to route water ~~from and~~ to the [2] CST or the suppression pool [1] to allow testing of the RCIC System during normal operation without injecting water into the RPV.

(continued)

BASES

BACKGROUND (continued)

The RCIC pump is provided with a minimum flow bypass line, which discharges to the suppression pool. The valve in this line automatically opens to prevent pump damage due to overheating when other discharge line valves are closed. To ensure rapid delivery of water to the RPV and to minimize water hammer effects, the RCIC System discharge line "keep fill" system is designed to maintain the pump discharge line filled with water.

APPLICABLE SAFETY ANALYSES

The function of the RCIC System is to respond to transient events by providing makeup coolant to the reactor. The RCIC System is not an Engineered Safety Feature System and no credit is taken in the safety analyses for RCIC System operation. Based on its contribution to the reduction of overall plant risk, ~~However,~~ the system ~~is included in the Technical Specifications as required by the NRC Policy Statement.~~ 5

Satisfies Criterion 4 of 10CFR50.36(c)(2)(ii)

LCO

The OPERABILITY of the RCIC System provides adequate core cooling such that actuation of any of the ECCS subsystems is not required in the event of RPV isolation accompanied by a loss of feedwater flow. The RCIC System has sufficient capacity to maintain RPV inventory during an isolation event.

APPLICABILITY

The RCIC System is required to be OPERABLE in MODE 1, and MODES 2 and 3 with reactor steam dome pressure > 150 psig since RCIC is the primary non-ECCS water source for core cooling when the reactor is isolated and pressurized. In MODES 2 and 3 with reactor steam dome pressure ≤ 150 psig, and in MODES 4 and 5, RCIC is not required to be OPERABLE since the ECCS injection/spray subsystems can provide sufficient flow to the vessel.

ACTIONS

A.1 and A.2

If the RCIC System is inoperable during MODE 1, or MODES 2 or 3 with reactor steam dome pressure > 150 psig, and the

(continued)

BASES

ACTIONS

A.1 and A.2 (continued)

immediately 4

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HPCS System is verified to be OPERABLE, the RCIC System must be restored to OPERABLE status within 14 days. In this Condition, loss of the RCIC System will not affect the overall plant capability to provide makeup inventory at high RPV pressure since the HPCS System is the only high pressure system assumed to function during a loss of coolant accident (LOCA). OPERABILITY of the HPCS is therefore verified immediately ~~within 1 hour~~ immediately when the RCIC System is inoperable. This may be performed as an administrative check, by examining logs or other information, to determine if the HPCS is out of service for maintenance or other reasons. Verification does not require performing the Surveillances needed to demonstrate the OPERABILITY of the HPCS System. If the OPERABILITY of the HPCS System cannot be immediately verified, however, Condition B must be ~~immediately~~ entered. For transients and certain abnormal events with no LOCA, RCIC (as opposed to HPCS) is the preferred source of makeup coolant because of its relatively small capacity, which allows easier control of RPV water level. Therefore, a limited time is allowed to restore the inoperable RCIC to OPERABLE status.

4 4

The 14 day Completion Time is based on a reliability study (Ref. 3) that evaluated the impact on ECCS availability, assuming that various components and subsystems were taken out of service. The results were used to calculate the average availability of ECCS equipment needed to mitigate the consequences of a LOCA as a function of allowed outage times (AOTs). Because of the similar functions of the HPCS and RCIC, the AOTs (i.e., Completion Times) determined for the HPCS are also applied to RCIC.

B.1 and B.2

If the RCIC System cannot be restored to OPERABLE status within the associated Completion Time, or if the HPCS System is simultaneously inoperable, the plant must be brought to a condition in which the LCO does not apply. To achieve this status, the plant must be brought to at least MODE 3 within 12 hours and reactor steam dome pressure reduced to ≤ 150 psig within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

(continued)

BASES (continued)

SURVEILLANCE REQUIREMENTS

SR 3.5.3.1

The flow path piping has the potential to develop voids and pockets of entrained air. Maintaining the pump discharge line of the RCIC System full of water ensures that the system will perform properly, injecting its full capacity into the Reactor Coolant System upon demand. This will also prevent a water hammer following an initiation signal. One acceptable method of ensuring the line is full is to vent at the high points. The 31 day Frequency is based on the gradual nature of void buildup in the RCIC piping, the procedural controls governing system operation, and operating experience.

SR 3.5.3.2

(including the RCIC pump flow controller)

2

Verifying the correct alignment for manual, power operated, and automatic valves in the RCIC flow path provides assurance that the proper flow path will exist for RCIC operation. This SR does not apply to valves that are locked, sealed, or otherwise secured in position since these were verified to be in the correct position prior to locking, sealing, or securing. A valve that receives an initiation signal is allowed to be in a nonaccident position provided the valve will automatically reposition in the proper stroke time. This SR does not require any testing or valve manipulation; rather, it involves verification that those valves capable of potentially being mispositioned are in the correct position. This SR does not apply to valves that cannot be inadvertently misaligned, such as check valves. For the RCIC System, this SR also includes the steam flow path for the turbine and the flow controller position.

The 31 day Frequency of this SR was derived from the Inservice Testing Program requirements for performing valve testing at least every 92 days. The Frequency of 31 days is further justified because the valves are operated under procedural control and because improper valve position would affect only the RCIC System. This Frequency has been shown to be acceptable through operating experience.

(continued)

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.5.3.3 and SR 3.5.3.4

The RCIC pump flow rates ensure that the system can maintain reactor coolant inventory during pressurized conditions with the RPV isolated. The flow tests for the RCIC System are performed at two different pressure ranges such that system capability to provide rated flow against a test line pressure corresponding to reactor pressure is tested both at the higher and lower operating ranges of the system. The required system head should overcome the RPV pressure and associated discharge line losses. Adequate reactor steam pressure must be available to perform these tests. Additionally, adequate steam flow must be passing through the main turbine or turbine bypass valves to continue to control reactor pressure when the RCIC System diverts steam flow. Reactor steam pressure must be ≥ [920] psig to perform SR 3.5.3.3 and ≥ ~~150~~ 135 psig to perform SR 3.5.3.4. Adequate steam flow is represented by at least ~~1.25~~ one turbine bypass valves ~~open~~ opened 50% ~~or total steam flow ≥ 10^6 lb/hr~~. Therefore, sufficient time is allowed after adequate pressure and flow are achieved to perform these SRs. Reactor startup is allowed prior to performing the low pressure Surveillance because the reactor pressure is low and the time to satisfactorily perform the Surveillance is short. The reactor pressure is allowed to be increased to normal operating pressure since it is assumed that the low pressure test has been satisfactorily completed and there is no indication or reason to believe that RCIC is inoperable. Therefore, these SRs are modified by Notes that state the Surveillances are not required to be performed until 12 hours after the reactor steam pressure and flow are adequate to perform the test. The 12 hours allowed for the flow tests after the required pressure and flow are reached are sufficient to achieve stable conditions for testing and provides a reasonable time to complete the SRs.

A 92 day Frequency for SR 3.5.3.3 is consistent with the Inservice Testing Program requirements. The ~~18~~ 24 month Frequency for SR 3.5.3.4 is based on the need to perform this Surveillance under the conditions that apply ~~just prior to or~~ during startup from a plant outage. Operating experience has shown that these components usually pass the SR when performed at the ~~18~~ 24 month Frequency, which is based on the refueling cycle. Therefore, the Frequency was concluded to be acceptable from a reliability standpoint.

SR 3.5.3.5

The RCIC System is required to actuate automatically to perform its design function. This Surveillance verifies that with a required system initiation signal (actual or simulated) the automatic initiation logic of RCIC will cause

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.5.3.5 (continued)

i.e., 2 2

the system to operate as designed, including actuation of the system throughout its emergency operating sequence, which includes automatic pump startup and actuation of all automatic valves to their required positions. This Surveillance test also 2 ensures that the RCIC System will automatically restart on an RPV low water level (Level 2) signal received subsequent to an RPV high water level (Level 8) trip and that the suction is automatically transferred from the CST to the suppression pool. The LOGIC SYSTEM FUNCTIONAL TEST performed in LCO 3.3.5.2 overlaps this Surveillance to provide complete testing of the assumed safety function.

1 actual or simulated

Shutdown signal 1

design 1

1 While this can be

~~The 18 month Frequency is based on the need to perform this Surveillance under the conditions that apply during a plant outage and the potential for an unplanned transient if the~~ Surveillance were performed with the reactor at power. ~~Operating experience~~ has shown that these components usually pass the SR when performed at the 18 month Frequency, which is based on the refueling cycle. Therefore, the Frequency was concluded to be acceptable from a reliability standpoint.

24 4

This SR is modified by a Note that excludes vessel injection during the Surveillance. Since all active components are testable and full flow can be demonstrated by recirculation through the test line, coolant injection into the RPV is not required during the Surveillance.

REFERENCES

1. 10 CFR 50, Appendix A, GDC 33.

1 U

2. FSAR, Section [5.4.6.2]. 3

3. Memorandum from R.L. Baer (NRC) to V. Stello, Jr. (NRC), "Recommended Interim Revisions to LCO's for ECCS Components," December 1, 1975.

JUSTIFICATION FOR DEVIATIONS FROM NUREG-1434, REVISION 1
ITS BASES: 3.5.3 - RCIC SYSTEM

1. Changes have been made (additions, deletions, and/or changes to the NUREG) to reflect the plant specific nomenclature, number, reference, system description, or analysis description.

2. Editorial change made for enhanced clarity or to be consistent with similar statements in other places in the Bases.

3. The brackets have been removed and the proper plant specific information/value has been provided.

4. Changes have been made to reflect those changes made to the Specification. The following requirements have been renumbered, where applicable, to reflect the changes.

5. The proper 10 CFR 50.36(c)(2)(ii) criterion has been used. The current wording was developed prior to the issuance of the change to 10 CFR 50.36, which uses criterion 4 for the current words of the NUREG.

GENERIC NO SIGNIFICANT HAZARDS EVALUATION
ITS: SECTION 3.5 - ECCS AND RCIC SYSTEM

ADMINISTRATIVE CHANGES
("A.x" Labeled Comments/Discussions)

In accordance with the criteria set forth in 10 CFR 50.92, ComEd has evaluated this proposed Technical Specifications change and determined it does not represent a significant hazards consideration. The following is provided in support of this conclusion.

1. Does the change involve a significant increase in the probability or consequences of an accident previously evaluated?

   The proposed change involves reformatting, renumbering, and rewording the existing Technical Specifications. The reformatting, renumbering, and rewording process involves no technical changes to the existing Technical Specifications. As such, this change is administrative in nature and does not impact initiators of analyzed events or assumed mitigation of accident or transient events. Therefore, this change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. Does the change create the possibility of a new or different kind of accident from any accident previously evaluated?

   The proposed change does not involve a physical alteration of the plant (no new or different type of equipment will be installed) or changes in methods governing normal plant operation. The proposed change will not impose any new or eliminate any old requirements. Thus, this change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. Does this change involve a significant reduction in a margin of safety?

   The proposed change will not reduce a margin of safety because it has no impact on any safety analyses assumptions. This change is administrative in nature. Therefore, the change does not involve a significant reduction in a margin of safety.

TECHNICAL CHANGES - MORE RESTRICTIVE
("M.x" Labeled Comments/Discussions)

In accordance with the criteria set forth in 10 CFR 50.92, ComEd has evaluated this proposed Technical Specifications change and determined it does not represent a significant hazards consideration. The following is provided in support of this conclusion.

1. Does the change involve a significant increase in the probability or consequences of an accident previously evaluated?

   The proposed change provides more stringent requirements for operation of the facility. These more stringent requirements do not result in operation that will increase the probability of initiating an analyzed event and do not alter assumptions relative to mitigation of an accident or transient event. The more restrictive requirements continue to ensure process variables, structures, systems, and components are maintained consistent with the safety analyses and licensing basis. Therefore, this change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. Does the change create the possibility of a new or different kind of accident from any accident previously evaluated?

   The proposed change does not involve a physical alteration of the plant (no new or different type of equipment will be installed) or changes in the methods governing normal plant operation. The proposed change does impose different requirements. However, these changes are consistent with the assumptions in the safety analyses and licensing basis. Thus, this change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. Does this change involve a significant reduction in a margin of safety?

   The imposition of more restrictive requirements either has no impact on or increases the margin of plant safety. As provided in the discussion of the change, each change in this category is by definition, providing additional restrictions to enhance plant safety. The change maintains requirements within the safety analyses and licensing basis. Therefore, this change does not involve a significant reduction in a margin of safety.

"GENERIC" LESS RESTRICTIVE CHANGES:
RELOCATING DETAILS TO TECHNICAL SPECIFICATION BASES, UFSAR, TRM, OR OTHER PLANT CONTROLLED DOCUMENTS
("LA.x" Labeled Comments/Discussions)

In accordance with the criteria set forth in 10 CFR 50.92, ComEd has evaluated this proposed Technical Specifications change and determined it does not represent a significant hazards consideration. The following is provided in support of this conclusion.

1. Does the change involve a significant increase in the probability or consequences of an accident previously evaluated?

   The proposed change relocates certain details from the Technical Specifications to the Bases, UFSAR, TRM, or other plant controlled documents. The Bases, UFSAR, TRM, and other plant controlled documents containing the relocated information will be maintained in accordance with 10 CFR 50.59. In addition to 10 CFR 50.59 provisions, the Technical Specification Bases are subject to the change control provisions in the Administrative Controls Chapter of the ITS. The UFSAR is subject to the change control provisions of 10 CFR 50.71(e), and the plant procedures and other plant controlled documents are subject to controls imposed by plant administrative procedures, which endorse applicable regulations and standards. Since any changes to the Bases, UFSAR, TRM, or other plant controlled documents will be evaluated per the requirements of the Bases Control Program in Chapter 5.0 of the ITS or 10 CFR 50.59, no increase (significant or insignificant) in the probability or consequences of an accident previously evaluated will be allowed. Therefore, this change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. Does the change create the possibility of a new or different kind of accident from any accident previously evaluated?

   The proposed change does not involve a physical alteration of the plant (no new or different type of equipment will be installed) or a change in the methods governing normal plant operation. The proposed change will not impose or eliminate any requirements, and adequate control of the information will be maintained. Thus, this change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. Does this change involve a significant reduction in a margin of safety?

   The proposed change will not reduce a margin of safety because it has no impact on any safety analysis assumptions. In addition, the details to be transposed from the Technical Specifications to the Bases, UFSAR, TRM, or other plant controlled

"GENERIC" LESS RESTRICTIVE CHANGES:
RELOCATING DETAILS TO TECHNICAL SPECIFICATION BASES, UFSAR, TRM, OR OTHER PLANT CONTROLLED DOCUMENTS
("LA.x" Labeled Comments/Discussions)

3. (continued)

documents are the same as the existing Technical Specifications. Since any future changes to these details in the Bases, UFSAR, TRM, or other plant controlled documents will be evaluated per the requirements of 10 CFR 50.59, no reduction (significant or insignificant) in a margin of safety will be allowed. Based on 10 CFR 50.92, the existing requirement for NRC review and approval of revisions, to these details proposed for relocation, does not have a specific margin of safety upon which to evaluate. However, since the proposed change is consistent with the BWR ISTS, NUREG-1434, Rev. 1, approved by the NRC Staff, revising the Technical Specifications to reflect the approved level of detail ensures no significant reduction in the margin of safety.

GENERIC NO SIGNIFICANT HAZARDS EVALUATION
ITS: SECTION 3.5 - ECCS AND RCIC SYSTEM

"GENERIC" LESS RESTRICTIVE CHANGES:
EXTENDING SURVEILLANCE FREQUENCIES FROM 18 MONTHS TO 24 MONTHS FOR SURVEILLANCES OTHER THAN CHANNEL CALIBRATIONS
("LD.x" Labeled Comments/Discussions)

In accordance with the criteria set forth in 10 CFR 50.92, ComEd has evaluated this proposed Technical Specifications change and determined it does not represent a significant hazards consideration. The following is provided in support of this conclusion.

1. Does the change involve a significant increase in the probability or consequences of an accident previously evaluated?

   The proposed change involves a change in the surveillance testing intervals from 18 months to 24 months. The proposed change does not physically impact the plant nor does it impact any design or functional requirements of the associated systems. That is, the proposed change does not degrade the performance or increase the challenges of any safety systems assumed to function in the accident analysis. The proposed change does not impact the Surveillance Requirements themselves nor the way in which the Surveillances are performed. Additionally, the proposed change does not introduce any new accident initiators since no accidents previously evaluated have as their initiators anything related to the frequency of surveillance testing. The proposed change does not affect the availability of equipment or systems required to mitigate the consequences of an accident because of the availability of redundant systems or equipment and because other test performed more frequently will identify potential equipment problems. Furthermore, an historical review of surveillance test results indicated that all failures identified were unique, non-repetitive, and not related to any time-based failure modes, and indicated no evidence of any failures that would invalidate the above conclusions. Therefore, the proposed change does not increase the probability or consequences of an accident previously evaluated.

2. Does the change create the possibility of a new or different kind of accident from any accident previously evaluated?

   The proposed change involves a change in the surveillance testing intervals from 18 months to 24 months. The proposed change does not introduce any failure mechanisms of a different type than those previously evaluated since there are no physical changes being made to the facility. In addition, the Surveillance Requirements themselves and the way Surveillances are performed will remain unchanged. Furthermore, an historical review of surveillance test results indicated no evidence of any failures that would invalidate the above conclusions. Therefore, the proposed change does not create the possibility of a new or different kind of accident from any previously evaluated.

"GENERIC" LESS RESTRICTIVE CHANGES:
EXTENDING SURVEILLANCE FREQUENCIES FROM 18 MONTHS TO 24 MONTHS FOR SURVEILLANCES OTHER THAN CHANNEL CALIBRATIONS
("LD.x" Labeled Comments/Discussions) (continued)

3. Does this change involve a significant reduction in a margin of safety?

Although the proposed change will result in an increase in the interval between surveillance tests, the impact on system availability is minimal based on other, more frequent testing or redundant systems or equipment, and there is no evidence of any failures that would impact the availability of the systems. Therefore, the assumptions in the licensing basis are not impacted, and the proposed change does not involve a significant reduction in a margin of safety.

NO SIGNIFICANT HAZARDS CONSIDERATION
ITS: 3.5.1 - ECCS — OPERATING

L.1 CHANGE

In accordance with the criteria set forth in 10 CFR 50.92, ComEd has evaluated this proposed Technical Specifications change and determined it does not represent a significant hazards consideration. The following is provided in support of this conclusion.

1. Does the change involve a significant increase in the probability or consequences of an accident previously evaluated?

   This change will raise the minimum pressure at which ADS is required to be OPERABLE to 150 psig. The ADS valves are not assumed to be an initiator of any analyzed event. Therefore, there is no significant increase in the probability of any previously analyzed accident. ADS is assumed in the mitigation of consequences of a loss of coolant accident which occurs at high reactor vessel pressure. ADS is not assumed in the mitigation of low pressure events since its function is to lower the pressure to within the capabilities of the low pressure makeup systems. Since this capability is not affected there is no significant increase in the consequences of any previously analyzed accident.

2. Does the change create the possibility of a new or different kind of accident from any accident previously evaluated?

   The proposed change does not introduce a new mode of plant operation and does not involve physical modification to the plant. Therefore, the proposed change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. Does this change involve a significant reduction in a margin of safety?

   The purpose of the ADS is to lower reactor pressure sufficiently to allow low pressure ECCS to inject and cool the core. Changing the minimum pressure for required OPERABILITY does not involve a significant reduction in a margin of safety since the ADS has been determined to be capable of performing its function at the higher reactor pressure (i.e., the current safety analysis shows the low pressure ECCS can provide core cooling at reactor pressures well above 150 psig).

L.2 CHANGE

In accordance with the criteria set forth in 10 CFR 50.92, ComEd has evaluated this proposed Technical Specifications change and determined it does not represent a significant hazards consideration. The following is provided in support of this conclusion.

1. Does the change involve a significant increase in the probability or consequences of an accident previously evaluated?

   The proposed change removes the requirement to submit a Special Report for ECCS actuation because the reporting requirements can be met by an LER required by 10 CFR 50.73(a)(2)(iv) and plant procedures that track ECCS actuation cycle information. The proposed change does not increase the probability of an accident because it will not involve any physical changes to plant systems, structures, or components, or the manner in which these systems, structures, or components are operated, maintained, modified, tested, or inspected. The Special Report for ECCS actuation is not assumed to be an initiator of any analyzed event. Also, the consequences of an accident are not affected by this report since it does not impact the assumptions of any design basis accident or transient.

2. Does the change create the possibility of a new or different kind of accident from any accident previously evaluated?

   The proposed change does not introduce a new mode of plant operation and does not involve physical modification to the plant. Therefore, it does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. Does this change involve a significant reduction in a margin of safety?

   The margin of safety is not reduced by removing the requirement for the submittal of a special report for ECCS actuation. This proposed change has no effect on the assumptions of the design basis accident. This change also has no impact on the safe operation of the plant because equivalent information is tracked and available or reported through the LER process. This change does not affect any plant equipment or requirements for maintaining plant equipment. Therefore, this change does not involve a significant reduction in a margin of safety.

L.3 CHANGE

In accordance with the criteria set forth in 10 CFR 50.92, ComEd has evaluated this proposed Technical Specifications change and determined it does not represent a significant hazards consideration. The following is provided in support of this conclusion.

1. Does the change involve a significant increase in the probability or consequences of an accident previously evaluated?

   The RHR System is not assumed to be the initiator of any previously analyzed event. The role of the LPCI mode of the RHR System is in mitigating and limiting consequences of analyzed events. The proposed change allows the RHR System to be aligned for RHR shutdown cooling mode while continuing to take credit for the LPCI mode. With this proposed change, LPCI is still capable of being manually realigned if needed to mitigate the consequences of an event. In addition, the allowance is applicable when the reactor is shutdown in MODE 3, with reactor pressure less than the RHR cut-in permissive pressure. The reactor heat load is considerably less in this condition than in MODE 1 (the MODE assumed in the accident analyses). Furthermore, the other subsystems of the ECCS are still required to be capable of performing their intended safety function. Therefore, this change will not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. Does the change create the possibility of a new or different kind of accident from any accident previously evaluated?

   The proposed change does not involve physical modification to the plant and the proposed change continues to provide assurance that LPCI is capable of performing its intended safety function when the reactor is shutdown and reactor pressure is less than the RHR cut-in permissive pressure. Therefore it does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. Does this change involve a significant reduction in a margin of safety?

   The proposed change has no impact on any safety analysis assumption. The clarifying Note allows the decay heat removal function to be available without the immediate shutdown requirements for two inoperable LPCI subsystems being imposed. This is in recognition that the amount of time to realign the RHR System from the shutdown cooling mode to the LPCI mode has no significant impact on the margin of safety associated with establishing LPCI injection, because the reactor heat load under these conditions is far below that assumed in the safety analysis. As a result, the margin of safety is not impacted by this change.

L.4 CHANGE

In accordance with the criteria set forth in 10 CFR 50.92, ComEd has evaluated this proposed Technical Specifications change and determined it does not represent a significant hazards consideration. The following is provided in support of this conclusion.

1. Does the change involve a significant increase in the probability or consequences of an accident previously evaluated?

   The phrase "actual or," in reference to the automatic initiation signal, has been added to the system functional test surveillance test description. This does not impose a requirement to create an "actual" signal, nor does it eliminate any restriction on producing an "actual" signal. While creating an "actual" signal could increase the probability of an event, existing procedures and 10 CFR 50.59 control of revisions to them, dictate the acceptability of generating this signal. The proposed change does not affect the procedures governing plant operations and therefore the probability of creating these signals; it simply would allow such a signal to be credited when evaluating the acceptance criteria for the system functional test requirements. Therefore, the change does not involve a significant increase in the probability of an accident previously evaluated. Since the method of initiation will not affect the acceptance criteria of the system functional test, the change does not involve a significant increase in the consequences of an accident previously evaluated.

2. Does the change create the possibility of a new or different kind of accident from any accident previously evaluated?

   The possibility of a new or different kind of accident from any accident previously evaluated is not created because the proposed change does not introduce a new mode of plant operation and does not involve physical modification to the plant.

3. Does this change involve a significant reduction in a margin of safety?

   Use of an actual signal instead of the existing requirement, which limits use to a simulated signal, will not affect the performance or acceptance criteria of the surveillance test. OPERABILITY is adequately demonstrated in either case since the system itself cannot discriminate between "actual" or "simulated" signals. Therefore, the change does not involve a significant reduction in a margin of safety.

L.5 CHANGE

In accordance with the criteria set forth in 10 CFR 50.92, ComEd has evaluated this proposed Technical Specifications change and determined it does not represent a significant hazards consideration. The following is provided in support of this conclusion.

1. Does the change involve a significant increase in the probability or consequences of an accident previously evaluated?

   The ECCS discharge line keep fill and differential pressure instrumentation are not assumed in the initiation of any analyzed event. The requirements for this instrumentation do not need to be explicitly stated in the Technical Specifications. The ECCS piping is still required to be verified filled with water per SR 3.5.1.1. In addition, the remaining Surveillance Requirements of ITS 3.5.1 are adequate to ensure the ECCS is maintained Operable. As a result, accident consequences are unaffected by this change. Therefore, this change will not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. Does the change create the possibility of a new or different kind of accident from any accident previously evaluated?

   The possibility of a new or different kind of accident from any accident previously evaluated is not created because the proposed change does not introduce a new mode of plant operation and does not involve physical modification to the plant.

3. Does the change involve a significant reduction in a margin of safety?

   The proposed deletion of the ECCS discharge line keep fill and differential pressure instrumentation requirements from Technical Specifications does not impact any margin of safety. The requirements for this instrumentation do not need to be explicitly stated in the Technical Specifications. The ECCS piping is still required to be verified filled with water per SR 3.5.1.1. The method to perform verification of SR 3.5.1.1 would require specific instrumentation to be OPERABLE. 10 CFR 50, Appendix B, Part XII requires that measuring devices used in activities affecting quality are properly controlled, calibrated, and adjusted to maintain accuracy within necessary limits. As a result, the OPERABILITY of this instrumentation will normally be maintained to satisfy SR 3.5.1.1 without the need for explicit instrumentation requirements in the Technical Specifications. In addition, the remaining Surveillance Requirements of ITS 3.5.1 are adequate to ensure the ECCS is maintained Operable. Therefore, this change does not involve a significant reduction in a margin of safety.

L.6 CHANGE

In accordance with the criteria set forth in 10 CFR 50.92, ComEd has evaluated this proposed Technical Specifications change and determined it does not represent a significant hazards consideration. The following is provided in support of this conclusion.

1. Does the change involve a significant increase in the probability or consequences of an accident previously evaluated?

   This change will increase the minimum pressure where OPERABILITY of the RCIC System is required when HPCS is inoperable to 150 psig. The RCIC System is not assumed to be an initiator of any analyzed event. Therefore, there is no significant increase in the probability of any previously analyzed accident. The function of the RCIC System is to respond to transient events by providing makeup coolant to the reactor pressure vessel (RPV). While the RCIC System is not an Engineered Safety Feature system, and no credit has been taken for it in the safety analyses, the system is important based on its contribution to the reduction of overall plant risk. Below 150 psig, the ability to provide makeup coolant to the RPV is adequately ensured through the presence of multiple low pressure ECCS. As a result, since core cooling capability is not affected, there is no significant increase in the consequences of any previously analyzed accident.

2. Does the change crease the possibility of a new or different kind of accident from any accident previously evaluated?

   The proposed change does not introduce a new mode of plant operation and does not involve physical modification to the plant. Therefore, the proposed change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. Does the change involve a significant reduction in a margin of safety?

   The HPCS and RCIC Systems are both designed to provide core cooling over a wide range of reactor pressure vessel pressures. Increasing the minimum pressure for required OPERABILITY of the RCIC System when HPCS is inoperable to 150 psig does not involve a significant reduction in a margin of safety since there are multiple low pressure ECCS that are capable of ensuring core cooling at this pressure (i.e., the current safety analysis shows the low pressure ECCS can provide core cooling at reactor pressures well above 150 psig).

NO SIGNIFICANT HAZARDS CONSIDERATION
ITS: 3.5.2 - ECCS — SHUTDOWN

L.1 CHANGE

In accordance with the criteria set forth in 10 CFR 50.92, ComEd has evaluated this proposed Technical Specifications change and determined it does not represent a significant hazards consideration. The following is provided in support of this conclusion.

1. Does the change involve a significant increase in the probability or consequences of an accident previously evaluated?

   The purpose of ECCS is to mitigate loss of core inventory accidents. Since CORE ALTERATIONS are not initiating events in LOCA analyses and the directions for suspending CORE ALTERATIONS are adequately addressed in the refueling LCOs, this change does not affect the probability or consequences of an accident previously evaluated.

2. Does the change create the possibility of a new or different kind of accident from any accident previously evaluated?

   The proposed change introduces no new mode of plant operation nor does it require physical modification to the plant. Therefore, it can not create the possibility of a new or different kind of accident.

3. Does this change involve a significant reduction in a margin of safety?

   The initiation, response, and effectiveness of ECCS do not depend upon, nor are they impacted by, CORE ALTERATIONS. Further, the necessity for suspending CORE ALTERATIONS and the margin of safety maintained thereby is appropriately addressed, initiated, and preserved by the LCOs in ITS Section 3.9 (Refueling Operations). Therefore, this change does not involve a significant reduction in a margin of safety.

L.2 CHANGE

In accordance with the criteria set forth in 10 CFR 50.92, ComEd has evaluated this proposed Technical Specification change and has determined it does not represent a significant hazards consideration. The following is provided in support of this conclusion.

1. Does the change involve a significant increase in the probability or consequences of an accident previously evaluated?

   This change does not result in any hardware or operating procedure changes. The reactor mode switch is not assumed in the initiation of any analyzed event. The requirement to "lock" the reactor mode switch in the required position was specified in the Technical Specifications to ensure that the reactor mode switch was not inadvertently moved from the Shutdown position resulting in an unauthorized MODE change. However, adequate administrative controls exist as a result of ITS Table 1.1-1, MODES, and the requirements of LCO 3.0.4 to ensure the reactor mode switch is maintained in the Shutdown position without the explicit requirement to "lock" the reactor mode switch in position. Reactor mode switch positions other than Shutdown may result in the unit entering some other MODE; with the associated Technical Specification compliance requirements of that MODE and of proposed LCO 3.0.4. As a result, the accident consequences are unaffected by this change. Therefore, this change will not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. Does the change create the possibility of a new or different kind of accident from any accident previously evaluated?

   The possibility of a new or different kind of accident from any accident previously evaluated is not created because the proposed change does not introduce a new mode of plant operation and does not involve physical modification to the plant.

3. Does this change involve a significant reduction in a margin of safety?

   The requirement to "lock" the reactor mode switch in the Shutdown position was specified in the Technical Specifications to ensure that the reactor mode switch was not inadvertently moved from the Shutdown position resulting in an unauthorized MODE change. However, adequate administrative controls exist as a result of ITS Table 1.1-1, MODES, and the requirements of LCO 3.0.4 to ensure the reactor mode switch is maintained in the Shutdown position without the explicit requirement to "lock" the reactor mode switch in Shutdown. Reactor mode switch positions other than Shutdown may result in the unit entering some other MODE; with the associated Technical Specification compliance requirements of that MODE and of proposed LCO 3.0.4. Therefore, this change does not involve a significant reduction in a margin of safety.

L.3 CHANGE

In accordance with the criteria set forth in 10 CFR 50.92, ComEd has evaluated this proposed Technical Specifications change and determined it does not represent a significant hazards consideration. The following is provided in support of this conclusion.

1. Does the change involve a significant increase in the probability or consequences of an accident previously evaluated?

   The purpose of ECCS is to mitigate loss of core inventory accidents. Providing an additional 4 hours prior to requiring Secondary Containment Integrity requirements to be met is acceptable since directions for suspending OPDRVs (i.e., actions that could cause a loss of core inventory) are adequately addressed by the ACTIONS for inoperable ECCS. There, this change does not significantly affect the probability or consequences of an analyzed accident.

2. Does the change create the possibility of a new or different kind of accident from any accident previously evaluated?

   The proposed change introduces no new mode of plant operation nor does it require physical modification to the plant. Therefore, it can not create the possibility of a new or different kind of accident.

3. Does this change involve a significant reduction in a margin of safety?

   The necessity for suspending OPDRVs and the margin of safety maintained thereby is appropriately addressed, initiated, and preserved by the ACTIONS for inoperable ECCS subsystems. Therefore, this change does not involve a significant reduction in a margin of safety.

L.4 CHANGE

In accordance with the criteria set forth in 10 CFR 50.92, ComEd has evaluated this proposed Technical Specifications change and determined it does not represent a significant hazards consideration. The following is provided in support of this conclusion.

1. Does the change involve a significant increase in the probability or consequences of an accident previously evaluated?

   The RHR System is not assumed to be the initiator of any previously analyzed event. The role of the LPCI mode of the RHR System is in mitigating and limiting consequences of analyzed events. The proposed change allows one RHR subsystem to be aligned for RHR shutdown cooling mode while continuing to take credit for the LPCI mode. With this proposed change, LPCI is still capable of being manually realigned if needed to mitigate the consequences of an event. In addition, the allowance is applicable during shutdown conditions. The reactor heat load is considerably less in this condition than in MODE 1 (the MODE assumed in the accident analyses). Furthermore, the other subsystems of the ECCS are still required to be capable of performing their intended safety function. Therefore, this change will not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. Does the change create the possibility of a new or different kind of accident from any accident previously evaluated?

   The proposed change does not involve physical modification to the plant and the proposed change continues to provide assurance that LPCI is capable of performing its intended safety function during shutdown conditions. Therefore it does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. Does this change involve a significant reduction in a margin of safety?

   The proposed change has no impact on any safety analysis assumption. The clarifying Note allows the decay heat removal function to be available without the requirements for an inoperable LPCI subsystem being imposed. This is in recognition that the amount of time to realign the RHR System from the shutdown cooling mode to the LPCI mode has no significant impact on the margin of safety associated with establishing LPCI injection, because the reactor heat load under these conditions is far below that assumed in the safety analysis. As a result, the margin of safety is not impacted by this change.

L.1 CHANGE

In accordance with the criteria set forth in 10 CFR 50.92, ComEd has evaluated this proposed Technical Specifications change and determined it does not represent a significant hazards consideration. The following is provided in support of this conclusion.

1. Does the change involve a significant increase in the probability or consequences of an accident previously evaluated?

   The phrase "actual or," in reference to the automatic initiation signal, has been added to the system functional test surveillance test description. This does not impose a requirement to create an "actual" signal, nor does it eliminate any restriction on producing an "actual" signal. While creating an "actual" signal could increase the probability of an event, existing procedures and 10 CFR 50.59 control of revisions to them, dictate the acceptability of generating this signal. The proposed change does not affect the procedures governing plant operations and therefore the probability of creating these signals; it simply would allow such a signal to be credited when evaluating the acceptance criteria for the system functional test requirements. Therefore, the change does not involve a significant increase in the probability of an accident previously evaluated. Since the method of initiation will not affect the acceptance criteria of the system functional test, the change does not involve a significant increase in the consequences of an accident previously evaluated.

2. Does the change create the possibility of a new or different kind of accident from any accident previously evaluated?

   The possibility of a new or different kind of accident from any accident previously evaluated is not created because the proposed change does not introduce a new mode of plant operation and does not involve physical modification to the plant.

3. Does this change involve a significant reduction in a margin of safety?

   Use of an actual signal instead of the existing requirement, which limits use to a simulated signal, will not affect the performance or acceptance criteria of the surveillance test. OPERABILITY is adequately demonstrated in either case since the system itself cannot discriminate between "actual" or "simulated" signals. Therefore, the change does not involve a significant reduction in a margin of safety.

L.2 CHANGE

In accordance with the criteria set forth in 10 CFR 50.92, ComEd has evaluated this proposed Technical Specifications change and determined it does not represent a significant hazards consideration. The following is provided in support of this conclusion.

1. Does the change involve a significant increase in the probability or consequences of an accident previously evaluated?

   The RCIC discharge line keep fill instrumentation is not assumed in the initiation of any analyzed event. The requirements for this instrumentation do not need to be explicitly stated in the Technical Specifications. The RCIC System piping is still required to be verified filled with water per SR 3.5.3.1. As a result, accident consequences are unaffected by this change. Therefore, this change will not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. Does the change create the possibility of a new or different kind of accident from any accident previously evaluated?

   The possibility of a new or different kind of accident from any accident previously evaluated is not created because the proposed change does not introduce a new mode of plant operation and does not involve physical modification to the plant.

3. Does this change involve a significant reduction in a margin of safety?

   The proposed deletion of the RCIC discharge line keep fill instrumentation requirements from Technical Specifications does not impact any margin of safety. The requirements for this instrumentation do not need to be explicitly stated in the Technical Specifications. The RCIC System piping is still required to be verified filled with water per SR 3.5.3.1. The method to perform verification of SR 3.5.3.1 would require specific instrumentation to be OPERABLE. 10 CFR 50, Appendix B, Part XII requires that measuring devices used in activities affecting quality are properly controlled, calibrated, and adjusted to maintain accuracy within necessary limits. As a result, the OPERABILITY of this instrumentation will normally be maintained to satisfy SR 3.5.3.1 without the need for explicit instrumentation requirements in the Technical Specifications. Therefore, this change does not involve a significant reduction in a margin of safety.

ENVIRONMENTAL ASSESSMENT
ITS: SECTION 3.5 - ECCS AND RCIC SYSTEM

In accordance with the criteria set forth in 10 CFR 50.21, ComEd has evaluated this proposed Technical Specification change for identification of licensing and regulatory actions requiring environmental assessment, determined it meets the criteria for a categorical exclusion set forth in 10 CFR 51.22(c)(9) and as such, has determined that no irreversible consequences exist in accordance with 10 CFR 50.92(b). This determination is based on the fact that this change is being proposed as an amendment to a license issued pursuant to 10 CFR which changes a requirement with respect to installation or use of a facility component located within the restricted area, as defined in 10 CFR 20, or which changes an inspection or a surveillance requirement, and the amendment meets the following specific criteria:

1. The amendment involves no significant hazards consideration.

   As demonstrated in the No Significant Hazards Consideration, this proposed amendment does not involve any significant hazards consideration.

2. There is no significant change in the type or significant increase in the amounts of any effluents that may be released offsite.

   The proposed change will not result in changes in the operation or configuration of the facility. There will be no change in the level of controls or methodology used for processing of radioactive effluents or handling of solid radioactive waste, nor will the proposal result in any change in the normal radiation levels within the plant. Therefore, there will be no change in the types or significant increase in the amounts of any effluents released offsite resulting from this change.

3. There is no significant increase in individual or cumulative occupational radiation exposure.

   The proposed change will not result in changes in the operation or configuration of the facility which impact radiation exposure. There will be no change in the level of controls or methodology used for processing of radioactive effluents or handling of solid radioactive waste, nor will the proposal result in any change in the normal radiation levels within the plant. Therefore, there will be no increase in individual or cumulative occupational radiation exposure resulting from this change.

Therefore, based upon the above evaluation, ComEd has concluded that no irreversible consequences exist with the proposed change.