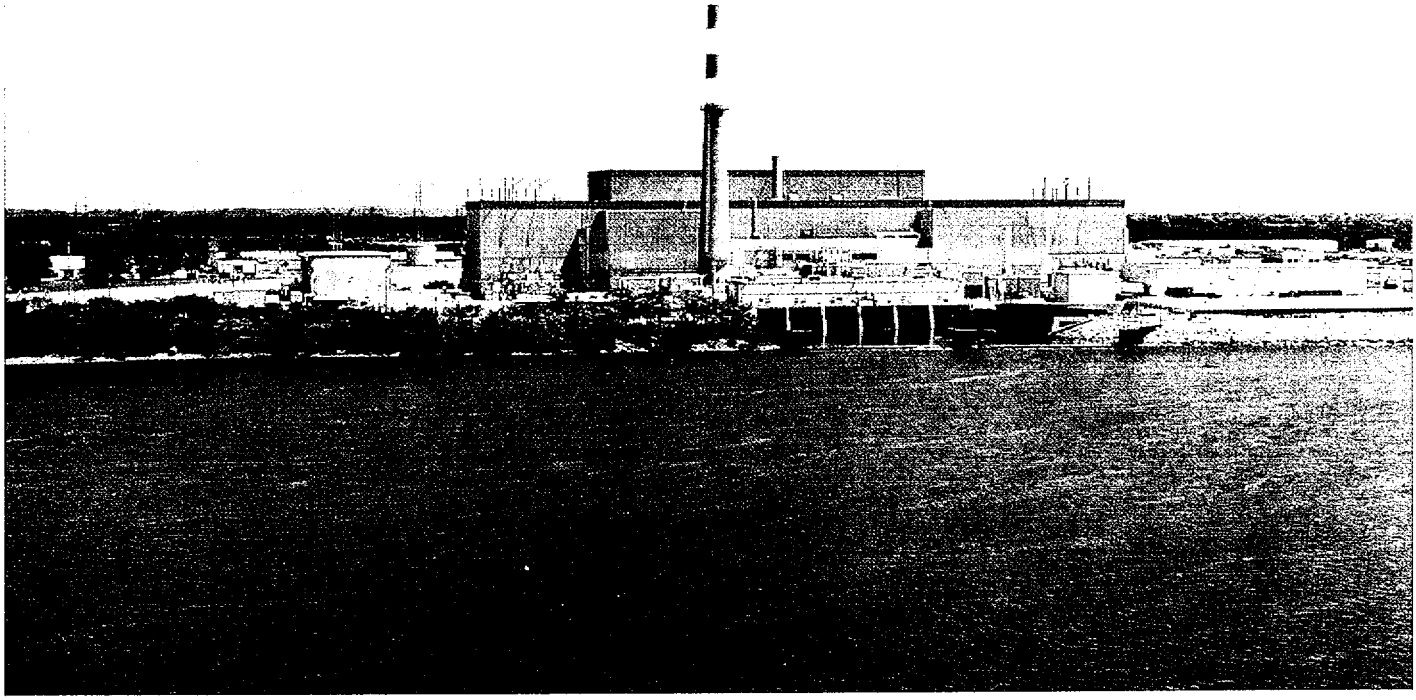
Improved Technical Specifications



Quad Cities 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,

OR

One recirculation loop shall be in operation 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 Neutron Flux—High), 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 is reset for single loop operation.

APPLICABILITY: MODES 1 and 2.

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. No recirculation loops in operation.	A.1	Be in MODE 2.	8 hours
	AND		
	A.2	Be in MODE 3.	12 hours

(continued)

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
B. Recirculation loop flow mismatch not within limits.	B.1 Declare the recirculation loop with lower flow to be "not in operation."	2 hours
C. Requirements of the LCO not met for reasons other than Condition A or B.	C.1 Satisfy the requirements of the LCO.	24 hours
D. Required Action and associated Completion Time of Condition C not met.	D.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 jet pump loop 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

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.2 Jet Pumps

LCO 3.4.2 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 3.4.2.1	----------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 one of the following criteria (a or b) is satisfied for each operating recirculation loop: a. Recirculation pump flow to speed ratio differs by ≤ 10% from established patterns. b. Each jet pump flow differs by ≤ 10% from established patterns.	24 hours

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.3 Safety and Relief Valves

LCO 3.4.3 The safety function of 9 safety valves shall be OPERABLE.

AND

The relief function of 5 relief valves shall be OPERABLE.

APPLICABILITY: MODES 1, 2, and 3.

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. One relief valve inoperable.	A.1	Restore the relief valve to OPERABLE status.	14 days
B. Required Action and associated Completion Time of Condition A not met. OR Two or more relief valves inoperable. OR One or more safety valves inoperable.	B.1 AND B.2	Be in MODE 3. Be in MODE 4.	12 hours 36 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.4.3.1	Verify the safety function lift setpoints of the safety valves are as follows: Number of Safety Valves — Setpoint (psig) 1 — 1135 ± 11.3 2 — 1240 ± 12.4 2 — 1250 ± 12.5 4 — 1260 ± 12.6	In accordance with the Inservice Testing Program
SR 3.4.3.2	----------NOTE---------- Not required to be performed until 12 hours after reactor steam pressure and flow are adequate to perform the test. ---------- Verify each relief valve opens when manually actuated.	24 months
SR 3.4.3.3	----------NOTE---------- Valve actuation may be excluded. ---------- Verify each relief valve actuates on an actual or simulated automatic initiation signal.	24 months

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.4 RCS Operational LEAKAGE

LCO 3.4.4 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. OR 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 limits. OR	4 hours (continued)

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
B. (continued)	B.2 Identify source of unidentified LEAKAGE increase is not IGSCC susceptible material.	4 hours
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
SR 3.4.4.1	Verify RCS unidentified and total LEAKAGE and unidentified LEAKAGE increase are within limits.	12 hours

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.5 RCS Leakage Detection Instrumentation

LCO 3.4.5 The following RCS leakage detection instrumentation shall be OPERABLE:

a. Drywell floor drain sump monitoring system; and

b. Primary containment atmospheric monitoring system.

APPLICABILITY: MODES 1, 2, and 3.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Drywell floor drain sump monitoring system inoperable.	----------NOTE---------- LCO 3.0.4 is not applicable. ----------	
	A.1 Restore drywell floor drain sump monitoring system to OPERABLE status.	30 days

(continued)

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
B. Required primary containment atmospheric particulate monitoring system inoperable.	----------NOTE---------- LCO 3.0.4 is not applicable. ----------		
	B.1	Analyze grab samples of primary containment atmosphere.	Once per 12 hours
	AND		
	B.2	Restore required primary containment atmospheric particulate monitoring system to OPERABLE status.	30 days
C. Required Action and associated Completion Time of Condition A or B not met.	C.1	Be in MODE 3.	12 hours
	AND		
	C.2	Be in MODE 4.	36 hours
D. All required leakage detection systems inoperable.	D.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 3.4.5.1	Verify primary containment atmospheric monitoring system operating.	12 hours
SR 3.4.5.2	Perform a CHANNEL FUNCTIONAL TEST of required leakage detection instrumentation.	31 days
SR 3.4.5.3	Perform a CHANNEL CALIBRATION of required leakage detection instrumentation.	24 months

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.6 RCS Specific Activity

LCO 3.4.6 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
	AND	
OR	B.2.1 Isolate all main steam lines.	12 hours
Reactor Coolant specific activity > 4.0 μCi/gm Dose EQUIVALENT I-131.	OR	

(continued)

ACTIONS

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

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.4.6.1	----------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)

3.4.7 Residual Heat Removal (RHR) Shutdown Cooling System – Hot Shutdown

LCO 3.4.7 Two RHR shutdown cooling subsystems shall be OPERABLE.

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

APPLICABILITY: MODE 3, with reactor steam dome 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(s) to OPERABLE status. AND	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.	1 hour
	AND		
	A.3	--------NOTE-------- Only applicable if both RHR shutdown cooling subsystems are inoperable. ----------	
		Verify reactor coolant circulation by an alternate method.	1 hour AND Once per 12 hours thereafter
	AND		
	A.4	--------NOTE-------- Only applicable if both RHR shutdown cooling subsystems are inoperable. ----------	
		Monitor reactor coolant temperature and pressure.	Once per hour
	AND		
	A.5	Be in MODE 4.	24 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.4.7.1	----------NOTE---------- Not required to be met until 2 hours after reactor steam dome pressure is less than the RHR cut-in permissive pressure. ---------- Verify each RHR shutdown cooling subsystem manual and power operated valve in the flow path, that is not locked, sealed or otherwise secured in position, is in the correct position or can be aligned to the correct position.	12 hours

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.8 Residual Heat Removal (RHR) Shutdown Cooling System – Cold Shutdown

LCO 3.4.8 Two RHR shutdown cooling subsystems shall be OPERABLE.

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

1. Not required to be met during hydrostatic testing.

2. 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 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. AND	1 hour AND Once per 24 hours thereafter

(continued)

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. (continued)	A.2 ---------NOTE-------- Only applicable if both RHR shutdown cooling subsystems are inoperable. ---------- Verify reactor coolant circulating by an alternate method.	1 hour AND Once per 12 hours thereafter
	AND	
	A.3 ---------NOTE-------- Only applicable if both RHR shutdown cooling subsystems are inoperable. ---------- Monitor reactor coolant temperature.	Once per hour

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.4.8.1	Verify each RHR shutdown cooling subsystem manual and power operated valve in the flow path, that is not locked, sealed or otherwise secured in position, is in the correct position or can be aligned to the correct position.	12 hours

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.9 RCS Pressure and Temperature (P/T) Limits

LCO 3.4.9 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.	30 minutes
	AND		
	A.2	Determine RCS is acceptable for continued operation.	72 hours
B. Required Action and associated Completion Time of Condition A not met.	B.1	Be in MODE 3.	12 hours
	AND		
	B.2	Be in MODE 4.	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 3.4.9.1	----------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 3.4.9-1, 3.4.9-2, and 3.4.9-3; b. RCS heatup and cooldown rates are $\leq$ 100°F in any 1 hour period; and c. RCS temperature change during inservice leak and hydrostatic testing is $\leq$ 20°F in any 1 hour period when the RCS pressure and RCS temperature are being maintained within the limits of Figure 3.4.9-1.	30 minutes
SR 3.4.9.2	Verify RCS pressure and RCS temperature are within the applicable criticality limits specified in Figure 3.4.9-3.	Once within 15 minutes prior to control rod withdrawal for the purpose of achieving criticality

(continued)

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.4.9.3	----------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 3.4.9.4	----------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 3.4.9.5	----------NOTE---------- Only required to be performed when tensioning the reactor vessel head bolting studs. ---------- Verify reactor vessel flange and head flange temperatures are ≥ 83°F.	30 minutes

(continued)

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.4.9.6	----------NOTE---------- Not required to be performed until 30 minutes after RCS temperature ≤ 93°F in MODE 4. ---------- Verify reactor vessel flange and head flange temperatures are ≥ 83°F.	30 minutes
SR 3.4.9.7	----------NOTE---------- Not required to be performed until 12 hours after RCS temperature ≤ 113°F in MODE 4. ---------- Verify reactor vessel flange and head flange temperatures are ≥ 83°F.	12 hours

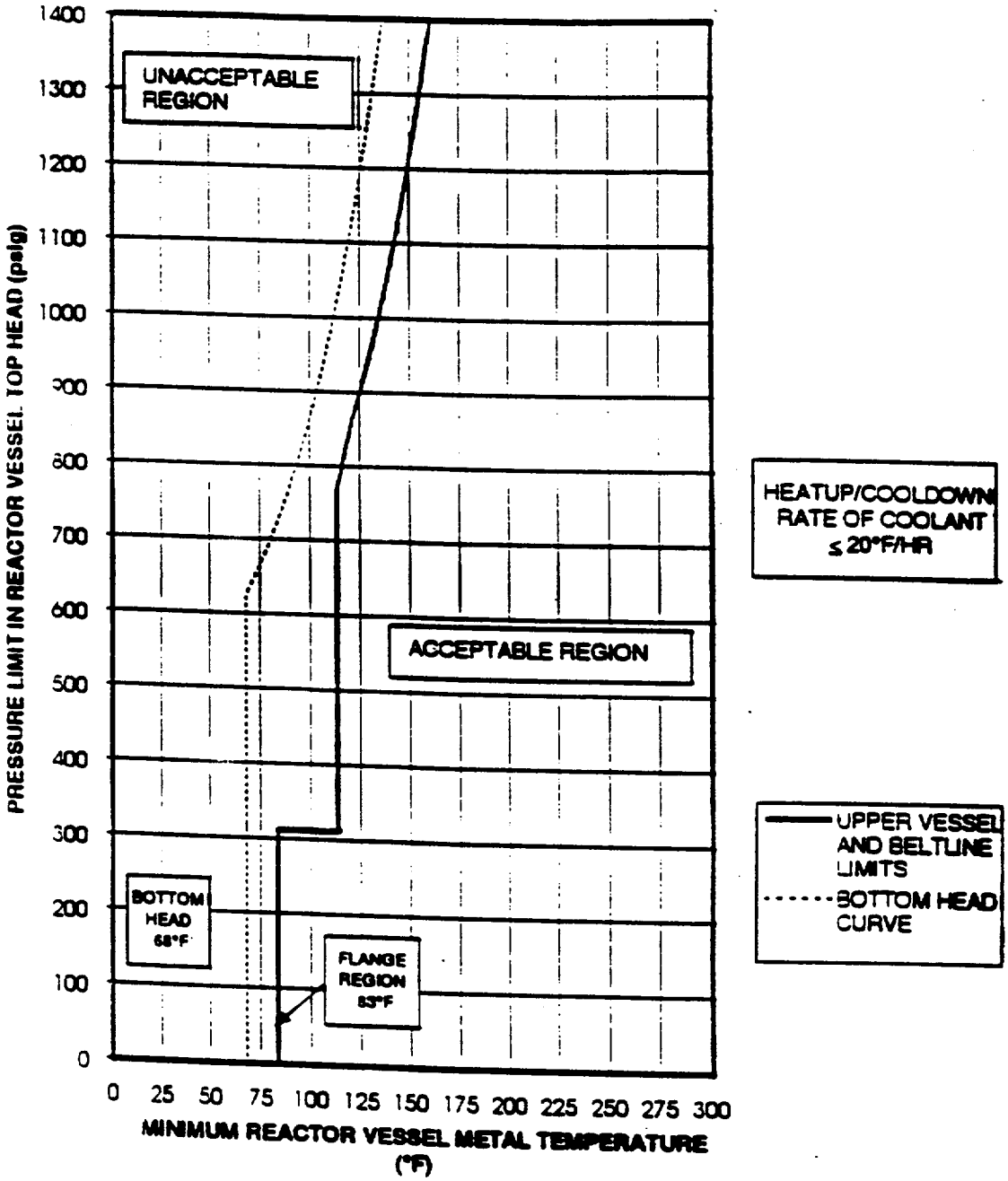


Figure 3.4.9-1 (Page 1 of 1)
Non-Nuclear Inservice Leak and Hydrostatic Testing Curve
(Valid to 32 EFPY)

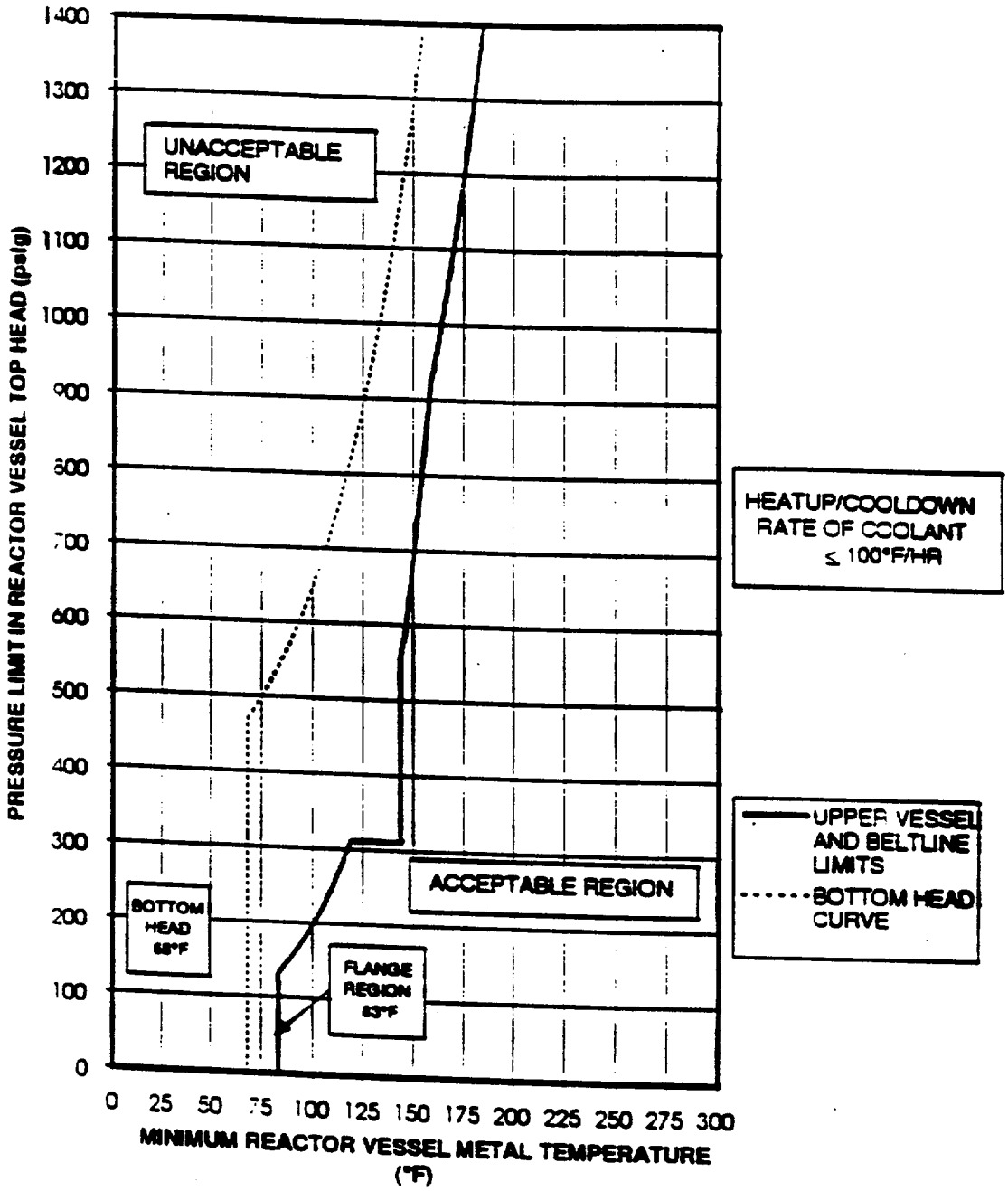


Figure 3.4.9-2 (Page 1 of 1)
Non-Nuclear Heatup/Cooldown Curve
(Valid to 32 EFPY)

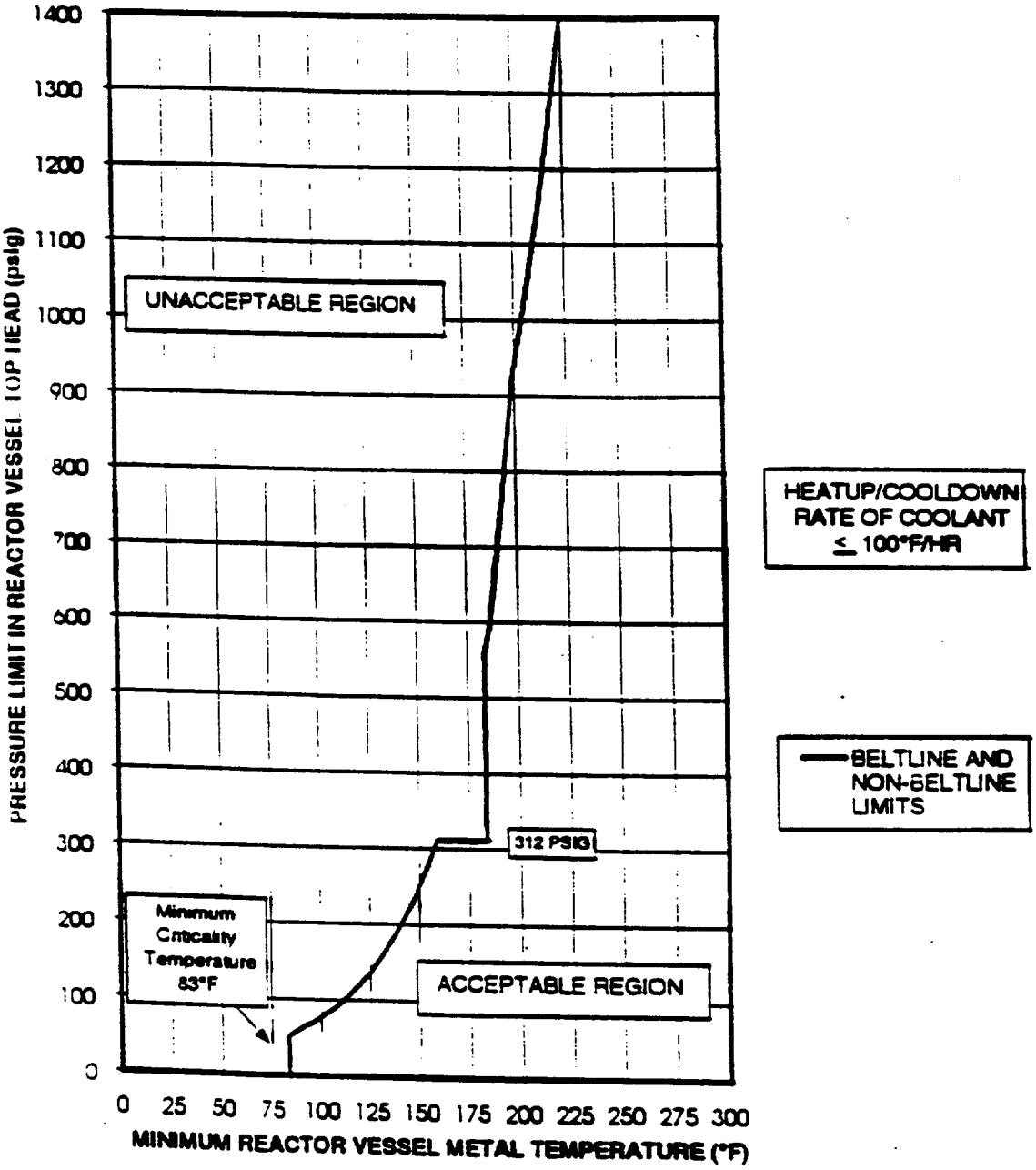


Figure 3.4.9-3 (Page 1 of 1)
Critical Operations Curve
(Valid to 32 EFPY)

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.10 Reactor Steam Dome Pressure

LCO 3.4.10 The reactor steam dome pressure shall be ≤ 1005 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 3.4.10.1 Verify reactor steam dome pressure is ≤ 1005 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 one variable speed motor driven recirculation pump, a motor generator (MG) set to control pump speed and 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. The subcooled water enters the bottom of the fuel channels and contacts the fuel cladding, where heat is transferred

(continued)

BASES

BACKGROUND (continued)

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 to 100% of RTP) without having to move control rods and disturb desirable flux patterns.

Each recirculation loop is manually started from the control room. The MG set provides regulation of individual recirculation loop drive flows. The flow in each loop is manually 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. 1). 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. 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.

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

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. 4) 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 (RPS) average power range monitor (APRM) and the Rod Block Monitor 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 Neutron Flux – High Allowable Value is in LCO 3.3.1.1, "Reactor Protection System (RPS) Instrumentation." The Rod Block Monitor-Upscale Allowable Value is in LCO 3.3.2.1, "Control Rod Block Instrumentation."

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. Alternatively, 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 Neutron Flux – High 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.

(continued)

BASES (continued)

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.

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 and A.2

With no recirculation loops in operation, the probability of thermal-hydraulic oscillations is greatly increased. Therefore, action must be taken as soon as practicable to reduce power to assure stability concerns are addressed and place the unit in at least MODE 2 within 8 hours and 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 transients and minimal dependence on the recirculation loop coastdown characteristics. 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.

B.1 and C.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 the lower flow must be declared "not in operation," as required by Required Action B.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 pump speeds to re-establish forward flow or by tripping the pump.

(continued)

BASES

ACTIONS

B.1 and C.1 (continued)

With the requirements of the LCO not met for reasons other than Condition A or B (e.g., one loop "not in operation"), the recirculation loops must be restored to operation with matched flows within 24 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 B.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 24 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.

D.1

With the Required Action and associated Completion Time of Condition C 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 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.

(continued)

BASES (continued)

SURVEILLANCE REQUIREMENTS

SR 3.4.1.1

This SR ensures the recirculation loops 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 jet pump loop 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 Surveillance 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. UFSAR, Section 6.3.3.2.

2. UFSAR, Chapter 15.

3. UFSAR, Section 6.3.3.2.2.4.

4. UFSAR, Section 15.3.6.

B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.2 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 recirculation loop contains ten 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 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.

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 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 3.4.2.1

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). Each recirculation loop must satisfy one of the performance criteria provided. 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 base-lining new "established patterns", engineering judgement of the daily surveillance results is used to detect significant abnormalities which could indicate a jet pump failure.

The recirculation pump speed operating characteristics (pump flow versus pump speed) are determined by the flow resistance from the loop suction through the jet pump nozzles. A change in the relationship may indicate a plug, flow restriction, loss in pump hydraulic performance, leakage, or new flow path between the recirculation pump discharge and jet pump nozzle. For this criterion, the pump flow versus pump speed relationship must be verified.

Individual jet pumps in a recirculation loop normally 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 pattern or relationship of one jet pump to

(continued)

BASES

SURVEILLANCE REQUIREMENTS

3.4.2.1 (continued)

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 flow patterns are established by plotting historical data as discussed in Reference 2.

Flow from a jet pump may be used to simulate the flow in the other jet pump associated with the same riser. This allowance may be used for one or two jet pumps. This is necessary since one jet pump flow indication instrument line in Unit 1 has failed. An analysis has been performed which demonstrated the acceptability of this method (Ref. 4).

The 24 hour Frequency has been shown by operating experience to be timely for detecting jet pump degradation and is consistent with the Surveillance 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 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.

2. GE Service Information Letter No. 330, including Supplement 1, "Jet Pump Beam Cracks," June 9, 1980.

(continued)

BASES

REFERENCES (continued)

3. NUREG/CR-3052, "Closeout of IE Bulletin 80-07: BWR Jet Pump Assembly Failure," November 1984.

4. EAS 07-0289, Revision 1, "Safety Evaluation to Justify Operation With Loss of Jet Pump Flow Indication for Quad Cities 1 and 2," March 1989.

B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.3 Safety and Relief Valves

BASES

BACKGROUND

The ASME Boiler and Pressure Vessel Code requires the reactor pressure vessel be protected from overpressure during upset conditions by self-actuated safety valves. As part of the nuclear pressure relief system, the size and number of safety valves are selected such that peak pressure in the nuclear system will not exceed the ASME Code limits for the reactor coolant pressure boundary (RCPB). Each unit is designed with nine safety valves, one of these valves also functions in the relief mode. This valve is a dual function Target Rock safety/relief valve (S/RV).

The safety valves and S/RV are located on the main steam lines between the reactor vessel and the first isolation valve within the drywell. The safety valves actuate in the safety mode (or spring mode of operation). In this mode, the safety valve opens when the inlet steam pressure reaches the lift set pressure. At that point, the vertical upward force generated by the inlet pressure under the valve disc balances the downward force generated by the spring. Slight steam leakage develops across the valve disc-to-seat interface and is directed into the huddle chamber. Pressure builds up rapidly in the huddle chamber developing an additional vertical lifting force on the disc and disc holder. This additional force in conjunction with the expansive characteristic of steam causes the valve to "pop" open to almost full lift. This satisfies the Code requirement. The S/RV is a dual function Target Rock valve that can actuate by either of two modes: the safety mode or the relief mode. In the safety mode (or spring mode of operation), the S/RV opens in the same manner as described above for the safety valves. In the relief mode (or power actuated mode of operation), automatic or manual switch actuation energizes a solenoid valve which pneumatically actuates a plunger located within the main valve body. Actuation of the plunger allows pressure to be vented from the top of the main valve piston. This allows reactor pressure to lift the main valve piston, which opens the main valve. The relief valves and S/RV discharge steam through a discharge line to a point below the minimum water level in the suppression pool. All other safety valves discharge directly to the drywell.

(continued)

BASES

BACKGROUND (continued)

In addition to the safety valves and S/RV, each unit is designed with four relief valves which actuate in the relief mode to control RCS pressure during transient conditions to prevent the need for safety valve actuation (except S/RV) following such transients. The relief valves are also located on the main steam lines between the reactor vessel and the first isolation valve within the drywell. These valves are sized by assuming a turbine trip, a coincident scram and a failure of the turbine bypass system. For Unit 1, four of the relief valves are of the Electromatic type, which are opened by automatic or manual switch actuation of a solenoid. The switch energizes the solenoid to actuate a plunger, which contacts the pilot valve operating lever, thereby opening the pilot valve. When the pilot valve opens, pressure under the main valve disc is vented. This allows reactor pressure to overcome main valve spring pressure, which forces the main valve disc downward to open the main valve. For Unit 2, four of the relief valves are of the Target Rock power operated relief valve type. When the solenoid is energized, a magnetic force is developed which moves a plunger upward until it contacts the moveable core. This motion is transmitted through the pilot rod to fully open two pilot discs, allowing the control pressure above the main disc to vent through the second pilot seat to the downstream side of the valve. In addition, the motion of the pilot disc partially reduces the control pressure above the main disc. When the force of the control pressure acting on the top of the main disc falls below the force of the inlet pressure acting on the lower annular area, the main disc will move to the open position. In the open position, with the moveable core positioned close to the fixed core, the magnetic force is well in excess of the closing forces due to control pressure and return spring force. This ensures that the main disc will be held firmly in the open position. The main disc can be opened even with the valve inlet pressure equal to 0 psig. Two of the five relief valves are the low set relief valves and all of the relief valves, including the S/RV, are Automatic Depressurization System (ADS) valves. The low set relief requirements are specified in LCO 3.6.1.6, "Low Set Relief Valves," and the ADS requirements are specified in LCO 3.5.1, "ECCS – Operating."

(continued)

BASES (continued)

APPLICABLE SAFETY ANALYSES

The overpressure protection system must accommodate the most severe pressurization 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. 1). For the purpose of the analyses, nine safety valves (including the S/RV) are assumed to operate in the safety mode. The relief valves are not credited to function during this event. The analysis results demonstrate that the design safety valve 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 (Refs. 2 and 3, respectively), the relief valves as well as the S/RV are assumed to function. [The opening of the relief valves during the pressurization event mitigates the increase in reactor vessel pressure, which affects the MINIMUM CRITICAL POWER RATIO (MCPR) during these events.] In these events, the operation of four of the five relief valves are required to mitigate the events. Reference 4 discusses additional events that are expected to actuate the safety and relief valves.

Safety and relief valves satisfy Criterion 3 of 10 CFR 50.36(c)(2)(ii).

LCO

The safety function of nine safety valves are required to be OPERABLE to satisfy the assumptions of the safety analysis (Ref. 1). The safety valve requirements of this LCO are applicable to the capability of the safety valves to mechanically open to relieve excess pressure when the lift setpoint is exceeded (safety function).

The safety valve setpoints are established to ensure that the ASME Code limit on peak reactor pressure is satisfied. The ASME Code specifications require the lowest safety valve

(continued)

BASES

LCO (continued)

setpoint to be at or below vessel design pressure (1250 psig) and the highest safety valve to be set so that the total accumulated pressure does not exceed 110% of the design pressure for overpressurization conditions. The transient evaluations in the UFSAR are based on these setpoints, but also include the additional uncertainties of ± 1% of the nominal 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 relief valves, including the S/RV, 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, all safety and relief valves must be OPERABLE, since considerable energy may be in the reactor core and the limiting design basis transients are assumed to occur in these MODES. The safety and relief valves 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 core heat.

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 and MCPR limits are 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 safety and relief functions are not needed during these conditions.

ACTIONS

A.1

With the relief function of one relief valve (or S/RV) inoperable, the remaining OPERABLE relief valves are capable of providing the necessary protection. However, the overall

(continued)

BASES

ACTIONS

A.1 (continued)

reliability of the pressure relief system is reduced because additional failures in the remaining OPERABLE relief valves 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 relief valve to OPERABLE status is based on the relief capability of the remaining relief valves, the low probability of an event requiring relief valve actuation, and a reasonable time to complete the Required Action.

B.1 and B.2

With less than the minimum number of required safety valves OPERABLE, a transient may result in the violation of the ASME Code limit on reactor pressure. If the relief function of the inoperable relief valves cannot be restored to OPERABLE status within the associated Completion Time of Required Action A.1, or if the relief function of two or more relief valves are inoperable, or if the safety function of one or more safety valves is 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 and to MODE 4 within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.4.3.1

This Surveillance requires that the safety valves, including the S/RV, will open at the pressures assumed in the safety analysis of Reference 1. The demonstration of the safety valve and S/RV safety lift settings must be performed during shutdown, since this is a bench test, to be done 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 safety valve and S/RV setpoints are ± 1% for OPERABILITY.

(continued)

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.4.3.2

A manual actuation of each relief valve, including the 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 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 relief valve or the S/RV diverts 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 300 psig (the pressure recommended by the valve manufacturer). Adequate steam flow is represented by at least 2 turbine bypass valves open.

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. Unit startup is allowed prior to performing this test because valve OPERABILITY is verified, per ASME Code requirements (Ref. 5), prior to valve installation. 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 S/RV fails to actuate due only to the failure of the solenoid but is capable of opening on overpressure, the safety function of the S/RV is considered OPERABLE.

The 24 month Frequency ensures that each solenoid for each relief valve is tested. The 24 month Frequency was developed based on the relief valve tests required by the ASME Boiler and Pressure Vessel Code, Section XI (Ref. 5). Operating experience has shown that these components usually pass the Surveillance when performed at the 24 month Frequency. Therefore, the Frequency was concluded to be acceptable from a reliability standpoint.

(continued)

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.4.3.3

The relief valves, including the S/RV, are required to actuate automatically upon receipt of specific initiation signals. A system functional test is performed to verify that the mechanical portions (i.e., solenoids) of the relief valve operate as designed when initiated either by an actual or simulated automatic initiation signal. The LOGIC SYSTEM FUNCTIONAL TESTs in LCO 3.3.5.1, "Emergency Core Cooling System (ECCS) Instrumentation," and LCO 3.3.6.3, "Relief Valve Instrumentation," overlap this SR to provide complete testing of the 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 these components usually pass the Surveillance when performed at the 24 month Frequency. 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 3.4.3.2.

REFERENCES

1. UFSAR, Section 5.2.2.1.
2. UFSAR, Section 15.2.3.1.
3. UFSAR, Section 15.2.2.1.
4. UFSAR, Chapter 15.
5. ASME, Boiler and Pressure Vessel Code, Section XI.

B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.4 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 UFSAR, Sections 3.1.2.4 and 3.1.3.6 (Ref. 1).

The safety significance of RCS LEAKAGE from the RCPB varies widely depending on the source, rate, and duration. Therefore, detection of LEAKAGE in the primary containment 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 that is detrimental to the safety of the facility or the public.

A limited amount of leakage inside primary containment is expected from auxiliary systems that cannot be made 100% leaktight. Leakage from these systems should be detected and isolated from the primary containment 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 that, 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. 2 and 3) shows that leakage rates of hundreds of gallons per minute will precede crack instability.

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.

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

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

The 5 gpm of unidentified LEAKAGE is allowed as a reasonable minimum detectable amount that the containment atmospheric monitoring and drywell floor drain sump 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 reasonable to properly reduce the LEAKAGE increase or identify the source before the reactor must be shut down without unduly jeopardizing plant safety.

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 safety systems.

SURVEILLANCE REQUIREMENTS

SR 3.4.4.1

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 3.4.5, "RCS Leakage Detection Instrumentation." The drywell floor drain sump flow integrator is typically monitored to determine actual LEAKAGE rates; however, an alternate method which may be used to quantify LEAKAGE is calculating flow rates using sump pump run times. In conjunction with alarms and other administrative controls, a 12 hour Frequency for this Surveillance is appropriate for identifying LEAKAGE and for tracking required trends (Ref. 4).

REFERENCES

1. UFSAR, Sections 3.1.2.4 and 3.1.3.6.

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

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

4. Generic Letter 88-01, Supplement 1, February 1992.

B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.5 RCS Leakage Detection Instrumentation

BASES

BACKGROUND

UFSAR, Sections 3.1.3.6 and 3.1.6.4 (Ref. 1), require means for detecting 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 leakage rates. The Bases for LCO 3.4.4, "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 two independently monitored variables, such as changes in flow out of the drywell floor drain sump and drywell particulate radioactivity levels. The drywell floor drain sump monitoring system provides a means of quantifying LEAKAGE in the drywell.

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 any LEAKAGE not collected in the drywell equipment drain sump. Leakage into the drywell floor drain sump is pumped through a piping header that penetrates the containment wall to the Liquid Radioactive Waste System.

(continued)

BASES

BACKGROUND (continued)

Two drywell floor drain sump pumps take suction from the drywell floor drain sump and discharge to the Liquid Radioactive Waste System. When a high level is reached in the floor drain sump, a level switch actuates to start a floor drain sump pump when the pump discharge valves are open. A flow monitor in the discharge line of the drywell floor drain sump pumps provides a flow input to a flow integrator in the control room. The flow integrator is used to quantify the amount of sump input. The pumps can also be started from the control room.

The primary containment atmospheric monitoring system continuously monitors the primary containment atmosphere for airborne particulate radioactivity. A sudden increase of radioactivity, which may be attributed to RCPB steam or reactor water LEAKAGE, is annunciated in the control room. The primary containment atmospheric particulate radioactivity monitoring system is not capable of quantifying LEAKAGE rates, but satisfies the Regulatory Guide 1.45 (Ref. 2) recommended sensitivity of 1.0E-9 μCi/cc radioactivity for airborne particulates.

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). The drywell floor drain sump monitoring system is designed with the capability of detecting LEAKAGE less than the established LEAKAGE rate limits. The primary containment atmospheric particulate radioactivity monitoring system provides indication of changes in leakage rates.

A control room alarm, provided by the primary containment atmospheric particulate radioactivity monitoring system, 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). 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).

(continued)

BASES (continued)

LCO

The drywell floor drain sump monitoring system is required to quantify the unidentified LEAKAGE from the RCS. Thus, for the system to be considered OPERABLE, the flow monitoring portion of the system must be OPERABLE. The other monitoring system (particulate) provides 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 3.4.4. This Applicability is consistent with that for LCO 3.4.4.

ACTIONS

A.1

With the drywell floor drain sump monitoring system inoperable, no other form of sampling can provide the equivalent information to quantify leakage. However, the primary containment atmospheric particulate monitoring system will provide indication of changes in leakage.

With the drywell floor drain sump monitoring system inoperable, but with RCS unidentified and total LEAKAGE being determined every 12 hours (SR 3.4.4.1), 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 monitoring system is inoperable. This allowance is provided because other instrumentation is available to monitor RCS leakage.

B.1 and B.2

With the particulate primary containment atmospheric monitoring system inoperable, grab samples of the primary containment atmosphere must be taken and analyzed to provide periodic leakage information. Provided a sample is obtained

(continued)

BASES

ACTIONS

B.1 and B.2 (continued)

and analyzed once every 12 hours, the plant may be operated for up to 30 days to allow restoration of the required monitor.

The 12 hour interval provides periodic information that is adequate to detect LEAKAGE. The 30 day Completion Time for restoration recognizes that at least one other form of leakage detection is available.

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 the particulate primary containment atmospheric monitoring channel is inoperable. This allowance is provided because other methods are available to monitor RCS leakage.

C.1 and C.2

If any Required Action and associated Completion Time of Condition A or B cannot be 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 MODE 4 within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to perform the actions in an orderly manner and without challenging plant systems.

D.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 (either the drywell floor drain sump monitoring system or the

(continued)

BASES

SURVEILLANCE REQUIREMENTS (continued)

primary containment atmospheric particulate 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.

SR 3.4.5.1

This SR requires verification that the required primary containment atmospheric monitoring system is in operation. The check ensures that the primary containment atmosphere is being continuously monitored for particulate radioactivity. The Frequency of 12 hours is based on instrument reliability and is reasonable for detecting off normal conditions.

SR 3.4.5.2

This SR is for 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 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 3.4.5.3

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

(continued)

BASES (continued)

REFERENCES

1. UFSAR, Sections 3.1.3.6 and 3.1.6.4.

2. Regulatory Guide 1.45, May 1973.

3. UFSAR, Section 5.2.5.7.

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.

B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.6 RCS Specific Activity

BASES

BACKGROUND

During circulation, the reactor coolant acquires radioactive materials due to release of fission products from fuel leaks into the reactor coolant and activation of corrosion products in the reactor coolant. These radioactive materials in the reactor 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 that 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 and control room 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

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BASES

APPLICABLE SAFETY ANALYSES (continued)

from an MSLB outside containment during steady state operation, will not exceed 10% of the dose guidelines of 10 CFR 100. The limits on the specific activity of the primary coolant also ensure the thyroid dose to control room operators, resulting from a MSLB outside containment during steady state operation will not exceed the limits of GDC 19 of 10 CFR 50, Appendix A (Ref. 3).

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 $\leq$ 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 and GDC 19 of 10 CFR 50, Appendix A (Ref. 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.1 and A.2

When the reactor coolant specific activity exceeds the LCO DOSE EQUIVALENT I-131 limit, but is $\leq$ 4.0 μCi/gm, samples must be analyzed for DOSE EQUIVALENT I-131 at least once

(continued)

BASES

ACTIONS

A.1 and A.2 (continued)

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.

A Note to the Required Actions 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 ≤ 0.2 μCi/gm within 48 hours, or if at any time it is > 4.0 μCi/gm, it must be determined at least once 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 and GDC 19 of 10 CFR 50, Appendix A (Ref. 3) during a postulated MSLB accident.

Alternatively, the plant can be placed in MODE 3 within 12 hours and in 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

(continued)

BASES

ACTIONS

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

challenging plant systems. Also, the allowed Completion Times for Required Actions B.2.2.1 and B.2.2.2 for placing the unit in MODES 3 and 4 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

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.

3. 10 CFR 50, Appendix A, GDC 19.

B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.7 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 ≤ 212°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 (loops) of the RHR System provide decay heat removal. Each loop consists of two motor driven pumps, 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 RHR Service Water System (LCO 3.7.1, "Residual Heat Removal Service Water (RHRSW) System").

APPLICABLE SAFETY ANALYSES

Decay heat removal by operation of 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 subsystems meet Criterion 4 of 10 CFR 50.36(c)(2)(ii).

LCO

Two RHR shutdown cooling subsystems are required to be OPERABLE. An OPERABLE RHR shutdown cooling subsystem consists of one OPERABLE RHR pump, one heat exchanger, and the associated piping and valves. The two subsystems have a common suction source and are allowed to have a common heat exchanger and common discharge piping. Thus, to meet the LCO, both RHR pumps (and two RHR service water pumps) in one

(continued)

BASES

LCO (continued)

loop or one RHR pump (and one RHR service water pump) in each of the two loops must be OPERABLE. Since the piping and heat exchangers are passive components that are assumed not to fail, they are allowed to be common to both subsystems. 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.

A Note allows one RHR shutdown cooling subsystem to be inoperable for up to 2 hours for the performance of Surveillance tests. These tests may be on the affected RHR shutdown cooling subsystem 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 shutdown cooling subsystems or other operations requiring RHR flow interruption and loss of redundancy.

APPLICABILITY

In MODE 3 with reactor steam dome 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 to ensure it can be operated in the shutdown cooling mode to remove decay heat to reduce or maintain coolant temperature.

In MODES 1 and 2, and in MODE 3 with reactor steam dome 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.

(continued)

BASES

APPLICABILITY (continued)

The requirements for decay heat removal in MODES 4 and 5 are discussed in LCO 3.4.8, "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.

A.1, A.2, A.3, A.4, and A.5

With one RHR shutdown cooling subsystem inoperable for decay heat removal, except as permitted by the LCO Note, the inoperable subsystem must be restored to OPERABLE status without delay (Required Action A.1). 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 (Required Action A.2).

(continued)

BASES

ACTIONS

A.1, A.2, A.3, A.4 and A.5 (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, the Reactor Water Cleanup System in the decay heat removal mode (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 relief valve(s). In addition, with both RHR shutdown cooling subsystems inoperable, an alternate method of reactor coolant circulation must be placed into service (Required Action A.3). This alternate method may be satisfied by placing a recirculation pump in operation. This will provide the necessary circulation for monitoring coolant temperature. The 1 hour Completion Time is based on the importance of the coolant circulation function. 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 an RHR shutdown cooling subsystem), the reactor coolant temperature and pressure must be periodically monitored to ensure proper function of the alternate method (Required Action A.4). The once per hour Completion Time is deemed appropriate. Required Actions A.3 and A.4 are modified by Notes that clarify that these Required Actions are only applicable when both RHR shutdown cooling subsystems are inoperable since Condition A is applicable when one or two RHR shutdown cooling subsystems are inoperable.

(continued)

BASES

ACTIONS

A.1, A.2, A.3, A.4, and A.5 (continued)

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 (Required Action A.5).

SURVEILLANCE REQUIREMENTS

SR 3.4.7.1

Verifying the correct alignment for manual and power operated valves in the two RHR shutdown cooling subsystems' flow paths provides assurance that the proper flow paths will exist for RHR 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 can be manually (from the control room or locally) aligned is allowed to be in a non-RHR shutdown cooling position provided the valve can be repositioned. 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 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., verification that valves are aligned or can be aligned 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 3.4.8 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 $\leq$ 212°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 (loops) of the RHR System provide decay heat removal. Each loop consists of two motor driven pumps, 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 RHR Service Water System.

APPLICABLE SAFETY ANALYSES

Decay heat removal by operation of 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. An OPERABLE RHR shutdown cooling subsystem consists of one OPERABLE RHR pump, one heat exchanger, the associated piping and valves, and the necessary portions of the RHR Service Water System capable of providing cooling water to the heat exchanger. The two subsystems have a common suction source and are allowed to have a common heat exchanger and common discharge piping. Thus, to meet the LCO, both RHR pumps in one loop (and two RHR service water pumps) or one RHR pump (and one RHR service water pump) in

(continued)

BASES

LCO (continued)

each of the two loops must be OPERABLE. Since the piping and heat exchangers are passive components that are assumed not to fail, they are allowed to be common to both subsystems. 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 4, one RHR shutdown cooling subsystem can provide the required cooling, but two subsystems are required to be OPERABLE to provide redundancy.

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 allows one RHR shutdown cooling subsystem to be inoperable for up to 2 hours for the 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 shutdown cooling subsystem 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 shutdown cooling subsystems or other operations requiring RHR flow interruption and loss of redundancy.

APPLICABILITY

In MODE 4, the RHR Shutdown Cooling System must be OPERABLE to ensure it can be operated in the shutdown cooling mode to remove decay heat to maintain coolant temperature below 212°F.

In MODES 1 and 2, and in MODE 3 with reactor steam dome 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

(continued)

BASES

APPLICABILITY (continued)

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 3.4.7, "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 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.

A.1

With one of the two RHR shutdown cooling subsystems inoperable, except as permitted by LCO Notes 1 and 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

(continued)

BASES

ACTIONS

A.1 (continued)

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 in the decay heat removal mode (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 a safety/relief valve.

A.2 and A.3

With both RHR shutdown cooling subsystems inoperable, except as permitted by LCO Note 1, an alternate method of reactor coolant circulation must be placed into service. This alternate method may be satisfied by placing a recirculation pump in operation. This will provide the necessary circulation for monitoring coolant temperature. The 1 hour Completion Time is based on the importance of the coolant circulation function. 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 one of the required RHR shutdown cooling subsystems), 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.

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BASES

ACTIONS

A.2 and A.3 (continued)

Required Actions A.2 and A.3 are modified by Notes that clarify that these Required Actions are only applicable when both RHR shutdown cooling subsystems are inoperable since Condition A is applicable when one or two RHR shutdown cooling subsystems are inoperable.

SURVEILLANCE REQUIREMENTS

SR 3.4.8.1

Verifying the correct alignment for manual and power operated valves in the two RHR shutdown cooling subsystems' flow paths provides assurance that the proper flow paths will exist for RHR 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 can be manually (remote or local) aligned is allowed to be in a non-RHR shutdown cooling position provided the valve can be repositioned. 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 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 3.4.9 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, and 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 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 Appendix H of 10 CFR 50 (Ref. 4). The operating P/T limit curves will be adjusted,

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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 non-critical cooldown curve and not lower than the minimum permissible temperature for the inservice leak and hydrostatic testing. Reference 1 also allows boiling water reactors to operate with the core critical below the minimum permissible temperature allowed for the inservice hydrostatic pressure test (i.e., inservice leak and hydrostatic testing) when the water level is within the normal range for power operation and the pressure is less than 20% of the preservice system hydrostatic test pressure (for Quad Cities 1 and 2, this pressure is 312 psig). Under these conditions, the minimum temperature is 60°F above the RT_{NDT} of the closure flange regions which are stressed by the bolt preload (for Quad Cities 1 and 2, this temperature is 83°F).

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

(continued)

BASES

BACKGROUND (continued)

RCPB components. 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 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 3.4.9-1, 3.4.9-2, and 3.4.9-3, heatup and cooldown rates are $\leq$ 100°F in any 1 hour period during RCS heatup, cooldown, and inservice leak and hydrostatic testing, and the RCS temperature change during inservice leak and hydrostatic testing is $\leq$ 20°F in any 1 hour period when the RCS temperature and pressure are being maintained within the limits of Figure 3.4.9-1;

b. The temperature difference between the reactor vessel bottom head coolant and the reactor pressure vessel (RPV) coolant is $\leq$ 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 $\leq$ 50°F during recirculation pump startup in MODES 1, 2, 3, and 4;

(continued)

BASES

LCO (continued)

d. RCS pressure and temperature are within the criticality limits specified in Figure 3.4.9-3, prior to achieving criticality; and

e. The reactor vessel flange and the head flange temperatures are $\geq$ 83°F when tensioning the reactor vessel head bolting studs and 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.

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.

(continued)

BASES (continued)

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 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 placed in a lower MODE because either the RCS remained in an unacceptable P/T region for an extended period of increased stress, or a

(continued)

BASES

ACTIONS

B.1 and B.2 (continued)

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 placing the plant in at least MODE 3 within 12 hours and in 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.

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 > 212°F. Several methods may be used, including comparison with pre-analyzed transients, 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 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.

(continued)

BASES (continued)

SURVEILLANCE REQUIREMENTS

SR 3.4.9.1

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 a reasonable time for assessment and correction of minor deviations.

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 with a Note that requires this Surveillance to be performed only during system heatup and cooldown operations and inservice leak and hydrostatic testing.

SR 3.4.9.2

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.

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 3.4.9.3 and SR 3.4.9.4

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 3.4.9.3 and SR 3.4.9.4 (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 3.4.9.3 is to compare the temperatures on the bottom head drain line and the saturation temperature corresponding to reactor steam dome pressure. An acceptable means of demonstrating compliance with the temperature differential requirement in SR 3.4.9.4 is to compare the temperatures of the operating recirculation loop and the idle loop.

SR 3.4.9.3 and SR 3.4.9.4 have been modified by a Note that requires the Surveillance to be performed only in MODES 1, 2, 3, and 4. In MODE 5, the overall stress on limiting components is lower. Therefore, ΔT limits are not required. The Notes also state the SRs are only required to be met during a recirculation pump startup since this is when the stresses occur.

SR 3.4.9.5, SR 3.4.9.6, and SR 3.4.9.7

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 $\leq$ 93°F, 30 minute checks of the flange temperatures are required

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.4.9.5, SR 3.4.9.6, and SR 3.4.9.7 (continued)

because of the reduced margin to the limits. When in MODE 4 with RCS temperature ≤ 113°F, monitoring of the flange temperature is required every 12 hours to ensure the temperature is 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 3.4.9.5 is modified by a Note that requires the Surveillance to be performed only when tensioning the reactor vessel head bolting studs. SR 3.4.9.6 is modified by a Note that requires the Surveillance to be initiated 30 minutes after RCS temperature ≤ 93°F in MODE 4. SR 3.4.9.7 is modified by a Note that requires the Surveillance to be initiated 12 hours after RCS temperature ≤ 113°F 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 185-82, July 1982.

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. Letter from S.N. Bailey (NRC) to ComEd, "Quad Cities - Issuance of Amendments - Revised Pressure-Temperature Limits," dated February 4, 2000.

8. UFSAR, Section 15.4.4.3.

B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.10 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 and transients.
APPLICABLE SAFETY ANALYSES	The reactor steam dome pressure of $\leq$ 1005 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 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 analyses are conserved. Reference 2 also assumes an initial reactor steam dome pressure for the analyses of design basis accidents and transients used to determine the limits for fuel cladding integrity (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)," LCO 3.2.3, "LINEAR HEAT GENERATION RATE (LHGR)," and LCO 3.2.4, "Average Power Range Monitor (APRM) Gain and Setpoint"). 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 ≤ 1005 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

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 occurring while pressure is greater than the limit is minimized.

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.

(continued)

BASES (continued)

SURVEILLANCE REQUIREMENTS	SR 3.4.10.1 Verification that reactor steam dome pressure is $\leq$ 1005 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 5.2.2.1. 2. UFSAR, Chapter 15.

A.1

ITS 3.4.1

A.2 general organization

3.6 - LIMITING CONDITIONS FOR OPERATION

A. Recirculation Loops

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

APPLICABILITY:

OPERATIONAL MODE(s) 1 and 2.

ACTION:

ACTION C

1. With only one reactor coolant system recirculation loop in operation, within 24 hours either, restore both loops to operation or:

A.2

LCO 3.4.1

a. Increase the MINIMUM CRITICAL POWER RATIO (MCPR) Safety Limit by 0.01 per Specification 2.1.B, and — A.3

b. Increase the MINIMUM CRITICAL POWER RATIO (MCPR) Operating Limit by 0.01 per Specification 3.11.C, and — LA.3

c. Reduce the Average Power Range Monitor (APRM) Flow Biased Neutron Flux Scram and Rod Block and Rod Block Monitor Trip Setpoints to those applicable to single recirculation loop operation per Specifications 2.2.A and 3.2.E. — A.4 — Allowable Values

d. Reduce the AVERAGE PLANAR LINEAR HEAT GENERATION RATE (APLHGR) to single loop operation limits as specified in the CORE OPERATING LIMITS REPORT (COLR).

4.6 - SURVEILLANCE REQUIREMENTS

A. Recirculation Loops

~~Each pump motor generator (MG) set scoop tube mechanical and electrical stop shall be demonstrated OPERABLE with the overspeed setpoints specified in the CORE OPERATING LIMITS REPORT at least once per 18 months.~~

LA.1

Page 1 of 3

3.6 - LIMITING CONDITIONS FOR OPERATION | 4.6 - SURVEILLANCE REQUIREMENTS

e. ~~Electrically prohibit the idle recirculation pump from starting~~[4]. L.1

ACTION D Otherwise, be in at least HOT SHUTDOWN within the next 12 hours.

ACTION A 2. With no reactor coolant system recirculation loops in operation, ~~immediately initiate measures to place the unit in at least~~ STARTUP within 8 hours and in HOT SHUTDOWN within the next 6 hours. LA.2 M.2

[4] ~~Except to permit testing in preparation for returning the pump to service.~~ L.1

L

ITS 3.4.1

A.1

3.6 - LIMITING CONDITIONS FOR OPERATION

C. Recirculation Pumps

LCO 3.4.1 Recirculation pump speed shall be maintained within:

SR 3.4.1.1

1. 10% of each other with THERMAL POWER ≥80% of RATED THERMAL POWER.

2. 15% of each other with THERMAL POWER <80% of RATED THERMAL POWER.

M.1

4.6 - SURVEILLANCE REQUIREMENTS

C. Recirculation Pumps

SR 3.4.1.1 Recirculation pump speed shall be verified to be within the limits at least once per 24 hours.

add proposed SR 3.4.1.1 Note

L.3

APPLICABILITY:

OPERATIONAL MODE(s) 1 and 2 ~~during two recirculation loop operation.~~

A.2

ACTION:

ACTION B With the recirculation pump speeds different by more than the specified limits, either:

1. ~~Restore the recirculation pump speeds to within the specified limit~~ within 2 hours, or

A.5

L.2

2. ~~Trip one of the recirculation pumps~~ and take the ACTION required by Specification 3.6.A.1.

A.6

DISCUSSION OF CHANGES
ITS: 3.4.1 - RECIRCULATION LOOPS OPERATING

ADMINISTRATIVE

A.1 In the conversion of the Quad Cities 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-1433, Rev. 1 (i.e., the Improved Standard Technical Specifications (ISTS)).

A.2 CTS 3.6.A requires both recirculation loops to be in operation. When one loop is inoperable, CTS 3.6.A Action 1 provides requirements that allow continued operation with only one recirculation loop in operation. CTS 3.6.A 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 3.6.A ACTIONS 1.b, 1.c, and 1.d. Since these specific requirements are now part of the LCO, CTS 3.6.A Action 1 (ITS 3.4.1 ACTION C) has been modified to require compliance with the requirements of the LCO. Similarly, the Applicability of CTS 3.6.C has been changed from OPERATIONAL MODE(s) 1 and 2 during two loop operation to MODES 1 and 2 (ITS 3.4.1) since the first option in proposed ITS LCO 3.4.1 requires two recirculation loops with match flows to be in operation. The explicit reference to "two recirculation loop operation" in the Applicability is no longer needed since it is part of the current and proposed LCO. This change is for ease of use and understanding only, and thus is administrative.

A.3 CTS 3.6.A Action 1.a requires an increase of the MCPR safety limit per CTS 2.1.B when only one recirculation loop is in operation. The Safety Limit requirement (CTS 2.1.B) is currently specified as the single loop limit; thus, when the plant is in single loop, the limit applies immediately, not in 24 hours as allowed by CTS 3.6.A Action 1.a. The ITS maintains the single loop MCPR safety limit in ITS 2.1.1.2. The COLR also provides the required MCPR operating limits based on the number of loops operating; thus when operation is shifted from two loop to single loop, a new MCPR operating limit is required, regardless of whether this Specification requires it.

A.4 The requirements in CTS 3.6.A Action 1.c to reduce the Average Power Range Monitor (APRM) Rod Block Trip 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 Flow Biased Neutron Flux 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

ADMINISTRATIVE (continued)

A.4 (cont'd) instrumentation. Reference to the Trip Setpoints has been eliminated in the referenced Specifications 2.2.A and 3.2.E (ITS 3.3.1.1 and ITS 3.3.2.1) and replaced with Allowable Values, therefore, this change is considered administrative. The elimination of Trip Setpoints, and replacement with Allowable Values, will be addressed in the Discussion of Changes for ITS 3.3.1.1 and ITS 3.3.2.1.

A.5 CTS 3.6.C Action 1 requires restoration of the recirculation pump speeds (i.e., jet pump loop flow in ITS) to within the limits if they are not within the limits. The revised presentation of ITS ACTIONS (based on the BWR ISTS, NUREG-1433, 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 3.6.C Action 1 (ITS 3.4.1 ACTION B) provides an alternate action that allows continued operation, deleting CTS 3.6.C Action 1 is purely editorial.

A.6 CTS 3.6.C Action 2 requires action to be taken per CTS 3.6.A.1 when recirculation pump speeds differ by more than the specified limits. The format of the ITS does not include providing "cross references." CTS 3.6.A.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 3.6.A.1" in CTS 3.6.C Action 2 serves no functional purpose, and its removal is purely an administrative difference in presentation.

TECHNICAL CHANGES - MORE RESTRICTIVE

M.1 CTS 3.6.C requires the recirculation pump speeds to be maintained within prescribed limits. With THERMAL POWER ≥ 80% of RATED THERMAL POWER the recirculation pump speeds must be within 10% of each other, and with THERMAL POWER < 80% of RATED THERMAL POWER, recirculation pump speeds must be within 15% of each other. In proposed SR 3.4.1.1, the jet pump loop flow mismatch with both recirculation loops in operation is: ≤ 10% of rated core flow when operating at < 70% of rated core flow; and ≤ 5% of rated core flow when operating at ≥ 70% of rated core flow. The proposed values are consistent with the loss of coolant accident (LOCA) analysis and a small mismatch has been determined to be acceptable based on engineering judgement. Since the required mismatch tolerance is smaller (although based on core flow), this change is considered to be an additional restriction on plant operation but necessary to ensure the LOCA analysis assumption is satisfied.

TECHNICAL CHANGES - MORE RESTRICTIVE (continued)

M.2 With no reactor coolant system recirculation loops in operation, CTS 3.6.A Action 2 requires the unit to be in at least STARTUP (MODE 2) within 8 hours and in HOT SHUTDOWN (MODE 3) within the next 6 hours. Under the same conditions ITS 3.4.1 Required Action A.1 will require the unit to be in MODE 2 in 8 hours and Required Action A.2 will require the unit to be in MODE 3 in 12 hours (next 4 hours). The change has been made for consistency with other conditions in the CTS and ITS which require the units to be in MODE 3. This change is more restrictive since the total time required to be in MODE 3 has decreased from 14 to 12 hours. This proposed time period is still adequate to achieve the required plant conditions in an orderly manner and without challenging plant systems.

TECHNICAL CHANGES - LESS RESTRICTIVE

"Generic"

LA.1 CTS 4.6.A requires the recirculation pump MG set scoop tube stop settings specified in the COLR to be verified at least once per 18 months. As indicated in the CTS requirement, the scoop tube stop settings are currently specified in the COLR. The details related to these operational settings are proposed to be relocated to Technical Requirements Manual (TRM). The MCPR operating limit is dependent on the MG set scoop tube stop settings as indicated in the Bases of ITS 3.2.2, "MINIMUM CRITICAL POWER RATIO (MCPR). Therefore, with the MG set scoop tube stop settings not within limit, the MCPR operating limit may not be valid and therefore MCPR must be declared not within limits in accordance with proposed ITS 3.2.2 Required Action A.1 and action must be taken to restore MCPR to within limits within 2 hours or the THERMAL POWER must be reduced below 25% RTP (ITS 3.2.2 Required Action B.1). SR 3.2.2.1 requires the MCPRs to be verified to be greater than the limits specified in the COLR once within 12 hours after THERMAL POWER is $\geq$ 25% RTP and once per 24 hours thereafter. The MCPR limits specified in the COLR are based on MG set scoop tube settings. Therefore, if the MG set scoop tube settings are not set in accordance with the relocated requirement, the MCPR must declared not within limits. These controls are considered adequate to ensure that MCPR will be within limits during normal and transient conditions. During transients initiated at reduced core flow the transient analysis assumes a failed speed rate (not speed limit) controller which results in an infinitely slow recirculation pump run-up rate which results in the most limiting MCPR. Most failures in the recirculation flow control system would actually result in a faster transient which will be mitigated by the Average Power Range Monitor Flow Biased Neutron Flux scram instrumentation required in proposed ITS 3.3.1.1,

TECHNICAL CHANGES - LESS RESTRICTIVE

LA.1 (cont'd) Reactor Protection System Instrumentation." Therefore, the relocated details are not required 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 the TRM will be controlled by the provisions of 10 CFR 50.59. Additionally, a discussion of the scoop tube stop settings and verification requirements will be included in the UFSAR, with changes controlled by the provisions of 10 CFR 50.59.

LA.2 The CTS 3.6.A Action 2 requirement to "immediately initiate measures to place the unit in at least STARTUP" when no recirculation loops are in operation is relocated to the Bases in the form of a discussion that "action must be taken as soon as practicable" to be in MODE 2. Immediate action may not always be the conservative method to assure safety. The 8 hour Completion Time of ITS 3.4.1 Required Action A.1 ensure appropriate actions are taken in a timely manner to place the unit in MODE 2. Therefore, the relocated requirement 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.

LA.3 The detail of the actual MCPR correction factor for the MCPR operating limit for single loop operation ("0.01") in CTS 3.6.A Action 1.b 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 C to satisfy the requirements of the LCO within 24 hours are adequate to ensure the current requirement is performed during single loop operation. Since all the requirements of CTS 3.6.A Action 1.b (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.

"Specific"

L.1 The explicit requirement in CTS 3.6.A Action 1.e to electrically prohibit the idle recirculation pump from starting except to permit testing in preparation for returning the pump to service has been deleted. This requirement is not necessary to minimize the consequences of any design basis accident. Plant operating practice and procedures are adequate to ensure the pumps are not

TECHNICAL CHANGES - LESS RESTRICTIVE

L.1 (cont'd) inadvertently started. In addition, the requirements in CTS 3.6.D (ITS 3.4.9, "RCS Pressure and Temperature (P/T) Limits") will help ensure that thermal stresses resulting from the startup of an idle recirculation pump will not exceed design allowances.

L.2 The required action of CTS 3.6.C Action 2 to trip one of the recirculation pumps when the speed mismatch (i.e. flow mismatch) is not within limits has been deleted. It has been replaced with a requirement (ITS 3.4.1 ACTION B) to declare the loop with the low flow "not in operation." Once the declaration has been made, the appropriate actions for single loop operation must be taken in accordance with CTS 3.6.A.1 (ITS 3.4.1). While a shutdown of the loop may be preferred under some conditions, declaring a pump not in operation will ensure the proper actions are taken in accordance with the single loop analysis.

L.3 CTS 4.6.C requires the recirculation pump speed mismatch (i.e., jet pump loop flow mismatch in ITS) to be verified within the limits once per 24 hours when in Operational MODES 1 and 2 during two recirculation loop operation. CTS 4.0.D requires the Surveillances to be met prior to entry into the applicable Mode or other specified conditions. CTS 4.6.C 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 added (proposed SR 3.4.1.1 Note) providing an allowance for time to initiate and complete the Surveillance 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

3.6 - LIMITING CONDITIONS FOR OPERATION

B. Jet Pumps

LCO 3.4.2 All jet pumps shall be OPERABLE and flow indication shall be OPERABLE on at least 18 jet pumps[a]. L.1

APPLICABILITY:

OPERATIONAL MODE(s) 1 and 2.

ACTION:

ACTION A

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

2. With flow indication inoperable for three or more jet pumps, flow indication shall be restored such that at least 18 jet pumps have OPERABLE flow indication within 12 hours or be in at least HOT SHUTDOWN within the next 12 hours.

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

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

4.6 - SURVEILLANCE REQUIREMENTS

B. Jet Pumps

All jet pumps shall be demonstrated OPERABLE as follows: L.2

Add proposed SR 3.4.2.1 NOTE 1

1. During two loop operation, at least once per 24 hours while greater than 25% of RATED THERMAL POWER by determining recirculation loop flow, total core flow and individual jet pump flow for each jet pump and verifying that no two of the following conditions occur when both recirculation pumps are operating in accordance with Specification 3.6.C: M.1 A.2

SR 3.4.2.1.a

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

b. The indicated total core flow differs by >10% from the established total core flow value derived from established core plate ΔP/core flow relationships. M.1

SR 3.4.2.1.b

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

SR 3.4.2.1 NOTE 2

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

---

a Inoperable flow indication shall not be allowed on both jet pumps sharing a jet pump riser, nor on both calibrated jet pumps on the same recirculation loop. L.1

3.6 - LIMITING CONDITIONS FOR OPERATION

4.6 - SURVEILLANCE REQUIREMENTS

L.2

Add proposed SR 3.4.2.1 NOTE 1

2. During single recirculation loop operation, at least once per 24 hours while greater than 25% of RATED THERMAL POWER by verifying that no two of the following conditions occur: A.2

SR 3.4.2.1.a

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

M.1

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

SR 3.4.2.1.b

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

SR3.4.2.1 NOTE 2

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

DISCUSSION OF CHANGES
ITS: 3.4.2 - JET PUMPS

ADMINISTRATIVE

A.1 In the conversion of the Quad Cities 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-1433, Rev. 1 (i.e., the Improved Standard Technical Specifications (ISTS)).

A.2 The wording in CTS 4.6.B.1 and CTS 4.6.B.2 (ITS SR 3.4.2.1) was changed to require verification that one of the criteria be met, rather than require verification that no two of the conditions exist. This change is consistent with NUREG-1433, Revision 1, which is written in a positive mode, such that conditions must exist, rather than not exist. Since this change does not modify any technical requirements, it is considered administrative.

TECHNICAL CHANGES - MORE RESTRICTIVE

M.1 CTS 4.6.B.1.b and CTS 4.6.B.2.b require the verification that indicated core flow differs by ≤ 10% from the established total core flow derived from the core plate delta-P/core flow relationship. These requirements have not been included in ITS 3.4.2. Guidance for the performance and evaluation criteria to detect jet pump inoperability is provided in GE SIL 330, June 9, 1980. According to the SIL, the core flow calculation based on core plate differential pressure/core flow relationship is not sensitive to jet pump performance degradation and is not recommended as one of the alternative methods for demonstrating jet pump OPERABILITY. As a result, the alternate core flow calculation method for demonstrating jet pump OPERABILITY as specified in CTS 4.6.B.1.b and CTS 4.6.B.2.b has not been included in proposed ITS SR 3.4.2.1. This is consistent with BWR ISTS, NUREG-1433, Rev. 1, in that, ISTS SR 3.4.2.1 also does not include the core flow calculation alternative. Since this change removes a method of demonstrating jet pump OPERABILITY, the number of acceptable methods for demonstrating OPERABILITY is reduced. Therefore, this change is more restrictive on plant operation.

TECHNICAL CHANGES - LESS RESTRICTIVE

L.1 The requirements of CTS 3.6.B and associated Actions 2, 3, and 4, concerning jet pump flow indication, do not necessarily relate directly to the structural integrity of the jet pumps. The BWR ISTS 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, details associated with jet pump flow indication is currently contained in the UFSAR. The requirement to demonstrate jet pump OPERABILITY is maintained in proposed SR 3.4.2.1 and provides for timely evaluation and detection of jet pump degradation. Jet pump indication is required to be OPERABLE to satisfy proposed SR 3.4.2.1. If the Surveillance Requirement cannot be satisfied, proposed SR 3.0.1 provides the appropriate guidance. As such, these requirements are not required to be in the ITS to provide adequate protection of the public health and safety and are proposed to be deleted from the Technical Specifications.

L.2 CTS 4.6.B.1 and CTS 4.6.B.2 requires the jet pump surveillance to be performed every 24 hours when > 25% RTP. This change adds a Note to CTS 4.6.B.1 and CTS 4.6.B.2 (proposed SR 3.4.2.1 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 the Surveillance not to be performed until four hours 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 an acceptable time to establish conditions appropriate for data collection and evaluation.

RELOCATED SPECIFICATIONS

None

A.1

ITS 3.4.3

A.2 (general organization)

3.6 - LIMITING CONDITIONS FOR OPERATION

E. Safety Valves

LCO 3.4.3

The safety valve function of the 9 reactor coolant system safety valves shall be OPERABLE in accordance with the specified code safety valve function lift settings[a] established as: LA.1

LA.3

SR 3.4.3.1

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

APPLICABILITY:

OPERATIONAL MODE(s) 1, 2 and 3.

ACTION:

ACTION B

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

2. Deleted.

4.6 - SURVEILLANCE REQUIREMENTS

E. Safety Valves

1. Deleted.

SR 3.4.3.1

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

In accordance with the Inservice Testing Program

LA.1

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

b Target Rock combination safety/relief valve. LA.3

Page 1 of 4

3.6 - LIMITING CONDITIONS FOR OPERATION

E. Safety Valves

LCO 3.4.3

The safety valve function of the 9 reactor coolant system safety valves shall be OPERABLE in accordance with the specified code safety valve function lift settings[a] established as: LA.3

SR 3.4.3.1

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

APPLICABILITY:

OPERATIONAL MODE(s) 1, 2 and 3

ACTION:

ACTION B

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

2. Deleted

4.6 - SURVEILLANCE REQUIREMENTS

E. Safety Valves

1. Deleted

A.3

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

In accordance with Inservice Testing Program — LA.2

---

a The lift setting pressure shall correspond to ambient conditions of the valves at nominal operating temperatures and pressures. — LA.1

b Target Rock combination safety/relief valve. — LA.3

c The surveillance interval has been extended to 24 months for Unit 1, cycle 16 only. The provisions of Specification 4.0.B are applicable. — A.3

A.1

A.2 (general organization)

3.6 - LIMITING CONDITIONS FOR OPERATION

F. Relief Valves

LCO 3.4.3

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

(see ITS 3.3.6.3)

Relief Function Setpoint (psig)	
Open	
≤1115 psig	
≤1115 psig	
≤1135 psig	
≤1135 psig	
≤1135 psig[a]	

(see ITS 3.3.6.3)

4.6 - SURVEILLANCE REQUIREMENTS

F. Relief Valves

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

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

   b. CHANNEL CALIBRATION and LOGIC SYSTEM FUNCTIONAL TEST of the entire system at least once per 18 months.

2. Deleted.

(see ITS 3.3.6.3)

APPLICABILITY:

OPERATIONAL MODE(s) 1, 2 and 3.

ACTION:

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

add proposed SRs 3.4.3.2 and 3.4.3.3

A.4

(see ITS 3.3.6.3)

a Target Rock combination safety/relief valve.

3.6 - LIMITING CONDITIONS FOR OPERATION

4.6 - SURVEILLANCE REQUIREMENTS

ACTION A

2. With the relief valve function and/or the reactuation time delay of one of the above required reactor coolant system relief valves inoperable, restore the inoperable relief valve function and the reactuation time delay function to OPERABLE status within 14 days or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

ACTION B

(see ITS 3.3.6.3)

ACTION B

3. With the relief valve function and/or the reactuation time delay of more than one of the above required reactor coolant system relief valves inoperable, be in at least HOT SHUTDOWN within 12 hours and in COLD SHUTDOWN within the next 24 hours.

4. Deleted.

DISCUSSION OF CHANGES
ITS: 3.4.3 - SAFETY AND RELIEF VALVES

ADMINISTRATIVE

A.1 In the conversion of the Quad Cities 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-1433, Rev. 1 (i.e., the Improved Standard Technical Specifications (ISTS)).

A.2 The organization of the Safety and Relief Valves requirements of CTS 3/4.6.E and CTS 3/4.6.F, respectively, are proposed to be included within one Specification in the ITS (ITS 3.4.3). The current licensing basis is similar to that of the BWR 6 Standard Technical Specifications, NUREG-1434, Rev. 1. Since this change does not alter any technical requirements, this change is considered administrative.

A.3 CTS 4.6.E.2 Note c provides a one time extension of the safety valve pressure testing surveillance interval from 18 months to 24 months for Unit 1, Cycle 16 only. This Note is deleted from the CTS since the one time extension will have expired prior to ITS implementation and will no longer be applicable. Since the change does not alter any technical requirements, it is considered to be administrative.

A.4 Two new Surveillance Requirements are proposed to be added. Proposed SR 3.4.3.2 ensures the relief valves open when manually actuated. This ensures that the valves and solenoids are functioning properly and that no blockage exists in the lines. Proposed SR 3.4.3.3 ensures that the relief valves will actuate on an actual or simulated automatic initiation signal. These proposed Surveillance Requirements are consistent with the current testing requirements in CTS 4.5.A.4.a and b (for ADS) as modified in the Discussion of Changes for ITS 3.5.1, "ECCS — Operating." Since inoperabilities associated with the mechanical portions of the ADS valves (which are also relief valves) require entry into both the Actions of CTS 3.6.F, "Relief Valves," as well as the Action of CTS 3.5.A, "ECCS — Operating," the duplication of these Surveillance Requirements in ITS 3.4.3 is considered to be administrative.

TECHNICAL CHANGES - MORE RESTRICTIVE

None

TECHNICAL CHANGES - LESS RESTRICTIVE

"Generic"

LA.1 The details of CTS 3.6.E footnote (a), relating to lift setting pressure of the safety 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 3.4.3.1 are adequate to ensure safety valve lift setpoints are within required settings. As a result, the details relocated to the Bases are not necessary for ensuring safety 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.

LA.2 The testing requirements of CTS 4.6.E.2 for safety valve setting verification are proposed to be relocated to the Inservice Testing (IST) Program. These testing requirements do demonstrate the Reactor Coolant System (RCS) safety valves are OPERABLE. However, the IST Program, required by 10 CFR 50.55a, provides requirements for the testing of all ASME Code Class 1, 2, and 3 valves in accordance with applicable codes, standards, and relief requests, endorsed by the NRC for Quad Cities 1 and 2. Compliance with 10 CFR 50.55a, and as a result the IST Program and implementing procedures, is required by the Quad Cities 1 and 2 Operating Licenses. These controls are adequate to ensure the required testing to demonstrate OPERABILITY is performed. 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 relocated requirements in the IST Program will be controlled by the provisions of 10 CFR 50.55a.

LA.3 The detail in CTS 3.6.E LCO footnote (b) that the lowest setting safety valve is a "Target Rock" combination safety/relief valve is proposed to be relocated to the Bases. Proposed LCO 3.4.3 will continue to require 9 safety valves to be OPERABLE and proposed SR 3.4.3.1 will continue to specify the settings of the safety valves. In addition, the Bases provides sufficient description of both the safety valves and the safety/relief valve. This detail is not necessary to ensure lift settings are maintained properly. As such, this 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 3.6.F Action 1 requires an open relief valve to be closed provided the suppression pool temperature is <110°F. If unable to close the open relief valve, or if suppression pool temperature is ≥ 110°F, the reactor mode switch must be placed in shutdown. This Action is not included in the ITS. Required Actions for open relief valves are implicit in the Actions of CTS 3.7.K and ITS 3.6.2.1. Required Action D.1 of ITS 3.6.2.1 will also require that the reactor mode switch be immediately placed in shutdown if the suppression pool average temperature is ≥ 110°F. Action 1 of CTS 3.6.F is anticipatory of this requirement in the event of an open relief valve, and preemptive in all cases. This Action represents detailed methods of responding to an event and not necessarily a compensatory action for failure to meet this LCO. As such it is not appropriate for the ITS and is adequately addressed in Quad Cities 1 and 2 Emergency Operating Procedures and by ITS 3.6.2.1, the Suppression Pool Temperature LCO. Therefore, CTS 3.6.F, Action 1 is proposed to be deleted from Technical Specifications.

RELOCATED SPECIFICATIONS

None

ITS 3.4.4

A.1

3.6 - LIMITING CONDITIONS FOR OPERATION

H. Operational Leakage

LCO 3.4.4 Reactor coolant system leakage shall be limited to:

1. No PRESSURE BOUNDARY LEAKAGE.
2. ≤25 gpm total leakage averaged over ~~any~~ the previous 24 hour surveillance period. A.2
3. ≤5 gpm UNIDENTIFIED LEAKAGE.
4. ≤2 gpm increase in UNIDENTIFIED LEAKAGE within ~~any~~ the previous period of 24 hours ~~or less~~ (Applicable in OPERATIONAL MODE 1 only).

4.6 - SURVEILLANCE REQUIREMENTS

H. Operational Leakage

The reactor coolant system leakage shall be demonstrated to be within each of the limits ~~by~~:

1. Sampling the primary containment atmospheric particulate radioactivity at least once per 12 hours[a] ~~and~~ — Moved to ITS 3.4.5 — A.3
2. ~~Determining the primary containment sump flow rate at least once per 8 hours, not to exceed 12 hours.~~ — LA.1 — L.1

SR 3.4.4.1

APPLICABILITY:

OPERATIONAL MODE(s) 1, 2 and 3.

ACTION:

ACTION C

1. 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

2. With the reactor coolant system UNIDENTIFIED LEAKAGE or total leakage rate(s) greater than the above limit(s), 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 C

---

a Not a means of quantifying leakage. — Moved to ITS 3.4.5 — A.3

3.6 - LIMITING CONDITIONS FOR OPERATION

4.6 - SURVEILLANCE REQUIREMENTS

ACTION B

3. With an increase in reactor coolant system UNIDENTIFIED LEAKAGE of >2 gpm within ~~any~~ the previous period ~~of~~ 24 hours ~~or less~~ in OPERATIONAL MODE 1: A.2

add proposed Required Action B.1

A.4

a. Identify the source of leakage as not IGSCC susceptible material within 4 hours, or

ACTION C

b. Be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

DISCUSSION OF CHANGES
ITS: 3.4.4 - RCS OPERATIONAL LEAKAGE

ADMINISTRATIVE

A.1 In the conversion of the Quad Cities 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-1433, Rev. 1 (i.e., the Improved Standard Technical Specifications (ISTS)).

A.2 In CTS 3.6.H.2, CTS 3.6.H.4, and CTS 3.6.H Action 3, 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 The CTS 4.6.H.1 requirement for sampling of primary containment particulate and the associated footnote a are being moved to proposed ITS 3.4.5, "RCS Leakage Detection Instrumentation," in accordance with the format of BWR ISTS, NUREG-1433, Rev. 1. Any technical changes to this requirement will be addressed in the Discussion of Changes for ITS 3.4.5.

A.4 If Reactor Coolant System unidentified LEAKAGE increases by greater than 2 gpm in a 24 hour period, CTS 3.6.H Action 3.a requires identification of the source of the leakage as not IGSCC susceptible material within 4 hours. ITS 3.4.4 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 as not IGSCC susceptible material. This change is considered administrative since restoring compliance with the LCO is always an option (per CTS 3.0.B), whether or not it is specifically stated in the Actions.

TECHNICAL CHANGES - MORE RESTRICTIVE

None

TECHNICAL CHANGES - LESS RESTRICTIVE

"Generic"

LA.1 Details of the CTS 4.6.H.2 method for performing the reactor coolant system leakage Surveillance (by determining the primary containment sump flow rate) is proposed to be relocated to the Bases. The requirements of proposed SR 3.4.4.1 are adequate to determine reactor coolant system leakage is within required limits. As a result, the details relocated to the Bases are not necessary for ensuring reactor coolant system leakage is determined 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 4.6.H.2 requires the primary containment sump flow rate (RCS unidentified and total LEAKAGE and unidentified LEAKAGE increase) to be determined at least once per 8 hours, not to exceed 12 hours. The Surveillance Frequency for CTS 4.6.H.2 has been changed to 12 hours in ITS SR 3.4.4.1. This change essentially allows the 25% extension specified in CTS 4.0.B (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 drywell sump pump flow integrators.

RELOCATED SPECIFICATIONS

None

A.1

3.6 LIMITING CONDITIONS FOR OPERATION

G. Leakage Detection Systems

LCO 3.4.5 The following reactor coolant system leakage detection systems shall be OPERABLE:

1. The primary containment atmosphere particulate radioactivity sampling system, and

2. The drywell floor drain sump monitoring system.

APPLICABILITY:

OPERATIONAL MODE(s) 1, 2 and 3.

ACTION:

add proposed NOTE to ACTION B — L.1

1. With the primary containment atmosphere particulate radioactivity sampling system inoperable, restore the inoperable leak detection radioactivity sampling system to OPERABLE status within 24 hours; otherwise, be in HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

ACTION B

ACTION C

add proposed Required Action B.1 — M.2

30 days — L.2

add proposed NOTE to ACTION A — L.1

2. With the drywell floor drain sump monitoring system inoperable, within 8 hours establish an alternate manual method of determining drywell floor drain sump flow rates by calculating flow rates using sump pump run times, AND within 30 days restore the drywell floor drain sump monitoring system to an OPERABLE status; otherwise, be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

ACTION A

A.3

ACTION C

A.4

add proposed ACTION D

4.11 - SURVEILLANCE REQUIREMENTS

L.3

G. Leakage Detection Systems

add proposed NOTE to Surveillance Table

The reactor coolant system leakage detection systems shall be demonstrated OPERABLE by:

A.5

1. ~~Performing the leakage determinations of Specification 4.6.H.~~

2. Performing a CHANNEL CALIBRATION of required leakage detection instrumentaion at least once per ~~18~~ 24 months.

SR 3.4.5.3

add proposed SR 3.4.5.2 — M.1

LE.1

A.1

3.6 - LIMITING CONDITIONS FOR OPERATION

H. Operational Leakage

Reactor coolant system leakage shall be limited to:

1. No PRESSURE BOUNDARY LEAKAGE.
2. ≤25 gpm total leakage averaged over any 24 hour surveillance period.
3. ≤5 gpm UNIDENTIFIED LEAKAGE.
4. ≤2 gpm increase in UNIDENTIFIED LEAKAGE within any period of 24 hours or less (Applicable in OPERATIONAL MODE 1 only).

APPLICABILITY:

OPERATIONAL MODE(s) 1, 2 and 3.

ACTION:

1. With any PRESSURE BOUNDARY LEAKAGE, be in at least HOT SHUTDOWN within 12 hours and in COLD SHUTDOWN within the next 24 hours.
2. With the reactor coolant system UNIDENTIFIED LEAKAGE or total leakage rate(s) greater than the above limit(s), 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.

4.6 - SURVEILLANCE REQUIREMENTS

H. Operational Leakage

The reactor coolant system leakage shall be demonstrated to be within each of the limits by:

1. Sampling the primary containment atmospheric particulate radioactivity at least once per 12 hours[a], and (SR 3.4.5.1) A.2 LA.1
2. Determining the primary containment sump flow rate at least once per 8 hours, not to exceed 12 hours.

<see ITS 3.4.4>

LA.1

a Not a means of quantifying leakage.

DISCUSSION OF CHANGES
ITS: 3.4.5 - RCS LEAKAGE DETECTION INSTRUMENTATION

ADMINISTRATIVE

A.1 In the conversion of the Quad Cities 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-1433, Rev. 1 (i.e., the Improved Standard Technical Specifications (ISTS)).

A.2 CTS 4.6.H.1 requires sampling the primary containment atmosphere particulate radioactivity at least once every 12 hours. Quad Cities 1 and 2 currently continuously monitor the primary containment atmosphere to satisfy this requirement. The CTS 4.6.H.1 requirement has been moved to proposed SR 3.4.5.1 requiring a verification of the operation of the continuous monitoring system. Therefore, this change is considered administrative.

A.3 The CTS 3.6.G Action 2 requirement to establish an alternate method of determining drywell floor drain sump flow rates within 8 hours has been deleted from the proposed ITS 3.4.5 ACTIONS. CTS 4.6.H.2 (proposed ITS SR 3.4.4.1) already requires the determination of drywell floor drain sump flow rate at this specified frequency and 3.6.H (proposed ITS 3.4.5) provides the appropriate ACTIONS for conditions when LEAKAGE cannot be determined. In addition, proposed SR 3.4.4.1 Bases address acceptable methods to determine LEAKAGE. As such, the deleted information is unnecessary and this change is considered administrative. Any technical changes to CTS 4.6.H.2 will be addressed in the Discussion of Changes for ITS 3.4.5.

A.4 Currently, no Actions are provided in CTS 3.6.G if all required RCS leakage detection systems are inoperable. Therefore, CTS 3.0.C must be entered. The revised presentation of ACTIONS is proposed to explicitly identify that LCO 3.0.3 (CTS 3.0.C) is required to be entered if all required RCS leakage detection systems are inoperable. Therefore ITS 3.4.5 ACTION D has been added to be consistent with the current requirements and is considered a presentation preference. Therefore, this change is administrative.

A.5 The requirement in CTS 4.6.G.1 to perform the leakage determinations of CTS 4.6.H has been deleted since it duplicates the requirement of CTS 4.6.H.2 (proposed ITS SR 3.4.4.1). Therefore, this change is considered administrative.

TECHNICAL CHANGES - MORE RESTRICTIVE

M.1 ITS SR 3.4.5.2 has been added to CTS 4.6.G to require a CHANNEL FUNCTIONAL TEST on the drywell floor drain sump monitoring system and the primary containment atmospheric particulate monitoring system on a 31 day frequency. This requirement ensures the primary containment atmospheric particulate alarm is functioning and the flow channel, including the integrator of the drywell floor drain sump monitoring system is functioning properly. This is an added requirement necessary to help ensure the RCS leakage detection instrumentation is maintained OPERABLE and therefore is considered more restrictive.

M.2 CTS 3.6.G Action 1 provides the actions for when the primary containment atmosphere particulate monitoring is inoperable. A new requirement has been added to CTS 3.6.G Action 1, which will require an analysis of a primary containment atmosphere grab sample once every 12 hours (ITS 3.4.5 Required Action B.1). This added requirement provides information which helps to detect LEAKAGE. This requirement is more restrictive on plant operation, however necessary to provide the information normally provided by the primary containment atmosphere particulate monitoring system. This change is consistent with NUREG-1433, Rev. 1.

TECHNICAL CHANGES - LESS RESTRICTIVE

"Generic"

LA.1 The detail in CTS 4.6.H Note a (that the primary containment atmospheric particulate radioactivity does not provide a means of quantifying leakage) is proposed to be relocated to the Bases. This detail is not necessary to be included in the Technical Specification to ensure the equipment remains OPERABLE. The requirements of proposed LCO 3.4.5 and the associated Surveillances will ensure the instrumentation remains OPERABLE. 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.

LE.1 The Frequency for performing the CHANNEL CALIBRATION required by CTS 4.6.G.2 (proposed SR 3.4.5.3) has been extended from 18 to 24 months to facilitate a change to the Quad Cities 1 and 2 fuel cycle from 18 months to 24 months including the 25% grace period. This SR ensures that the required primary containment atmospheric particulate monitoring system and drywell

TECHNICAL CHANGES - LESS RESTRICTIVE

LE.1 (cont'd) floor drain sump monitoring systems are OPERABLE and within the established calibration requirements. The proposed change will allow this Surveillance to extend the 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.B 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.B) 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. Drywell sumps are monitored to indirectly determine Reactor Coolant System leakage rates. Rate measurements look at the change in condition over a relatively short period of time. Since drift is a small change over a long duration (months) and leakage is determined over a short duration (hours) as a differential, the Reactor Coolant Leakage Detection Systems will not be affected by instrument drift resulting from an increase in calibration interval. Additionally, two fixed-measurement interval methods exist for determining unidentified drywell leakage rates. First, the leakage rate can be calculated using the flow integrator data as indicated in the control room, monitoring the volume pumped out of the drywell floor drain sump. The second fixed-measurement method involves monitoring the drywell floor sump drain pump run time. By determining pump run time over a given period, the leakage rate can be determined if the pump rate is known or can be conservatively estimated. Therefore, the extension of the surveillance interval is justified. The primary containment atmospheric particulate monitoring system provides backup to the drywell floor drain sump monitoring system, in that they both will alert the operators to unanticipated unidentified leakage from the RCPB. 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. Proposed SRs 3.4.5.1 and 3.4.5.2 require that the CHANNEL CHECK and CHANNEL FUNCTIONAL TEST be performed on a more frequent basis. 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 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 Currently, CTS 3.0.D (ITS LCO 3.0.4) would preclude a change in MODES with leakage detection monitoring system inoperable. A statement that LCO 3.0.4 is not applicable for the condition of the drywell floor drain sump monitoring system inoperable or the primary containment atmospheric particulate monitoring system inoperable has been added as a Note to ITS 3.4.5 ACTION A and ACTION B. When this allowance is used, either the drywell floor drain sump monitoring system or the required primary containment atmospheric particulate monitoring system remains available, and the compensatory actions for the inoperable system (or the requirement that unidentified leakage be quantified in accordance with ITS 3.4.4) will provide adequate indication of RCS leakage. Since 1) probabilities have determined a 30 day allowed out of service time for one leakage detection system is acceptable; 2) a leakage detection system is still OPERABLE; and 3) compensatory measures will still ensure leakage is being quantified, the LCO 3.0.4 exception is considered to provide no significant impact on safety and is acceptable.

L.2 CTS 3.6.G Action 1 requires the restoration of the inoperable primary containment atmospheric particulate radioactivity sampling system leakage detection system to OPERABLE status within 24 hours. Proposed ITS 3.4.5, Required Action B.2 requires an inoperable primary containment atmospheric monitoring system be restored to OPERABLE status within 30 days. This is a relaxation of requirements, and therefore less restrictive. This 30 day Completion Time is allowed provided primary containment atmospheric particulate radioactivity is determined once every 12 hours, by analyses of grab samples of the primary containment atmosphere, as required in proposed ITS 3.4.5, Required Action B.1. These changes are acceptable based on operating experience, considering there is another method of leakage detection still available to monitor and assess RCS Operational LEAKAGE (analysis of grab samples of the primary containment atmosphere) and RCS unidentified and total LEAKAGE can be quantified.

L.3 A Note has been added to CTS 4.6.G (Note to ITS 3.4.5 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 instrumentation is OPERABLE. This change will minimize unnecessary entry/exit into the actions for OPERABLE instrumentation. Operational history indicates that the performance of the Surveillances associated with RCS Leakage Detection Instrumentation normally demonstrate that the instrumentation is in

TECHNICAL CHANGES - LESS RESTRICTIVE

L.3 (cont'd) fact OPERABLE. In addition, CTS 4.0.C (and as modified in the Discussion of Changes for Section 3.0) allows entry into the actions to be delayed for missed Surveillances for up to a period of 24 hours, therefore this short time period of 6 hours for instrumentation tested within the required Surveillance Frequency should also be permitted. 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 RCS leakage since the other channel must be OPERABLE for this allowance to be used.

RELOCATED SPECIFICATIONS

None

3.6 - LIMITING CONDITIONS FOR OPERATION

J. Specific Activity

LCO 3.4.6 The specific activity of the reactor coolant shall be limited to ≤0.2 μCi/gram DOSE EQUIVALENT I-131.

APPLICABILITY:

OPERATIONAL MODE(s) 1, 2 and 3, with any main steam line not isolated.

ACTION:

ACTION A

1. With the specific activity of the reactor coolant >0.2 μCi/gram DOSE EQUIVALENT I-131 but ≤4.0 μCi/gram DOSE EQUIVALENT I-131, determine DOSE EQUIVALENT I-131 once per 4 hours and restore DOSE EQUIVALENT I-131 to within limits within 48 hours[a].

ACTION B

2. With the specific activity of the reactor coolant >0.2 μCi/gram DOSE EQUIVALENT I-131 for greater than 48 hours, or with the specific activity of the reactor coolant >4.0 μCi/gram DOSE EQUIVALENT I-131, determine DOSE EQUIVALENT I-131 once per 4 hours, and isolate all main steam lines within 12 hours, or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

4.6 - SURVEILLANCE REQUIREMENTS

J. Specific Activity

SR 3.4.6.1 In OPERATIONAL MODE 1, the specific activity of the reactor coolant shall be verified to be ≤0.2 μCi/gram DOSE EQUIVALENT I-131 once per 7 days.

Required Actions A.1 and A.2 Note

LCO 3.0.4

[a] The provisions of ~~Specification 3.0.B~~ are not applicable.

DISCUSSION OF CHANGES
ITS: 3.4.6 - RCS SPECIFIC ACTIVITY

ADMINISTRATIVE

A.1 In the conversion of the Quad Cities 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-1433, Rev. 1 (i.e., the Improved Standard Technical Specifications (ISTS)).

TECHNICAL CHANGES - MORE RESTRICTIVE

None

TECHNICAL CHANGES - LESS RESTRICTIVE

None

RELOCATED SPECIFICATIONS

None

ITS 3.4.7

 A.1

add proposed SR 3.4.7 Note L.2

3.6 - LIMITING CONDITIONS FOR OPERATION

O. Residual Heat Removal - HOT SHUTDOWN

LCO 3.4.7

Two shutdown cooling mode subsystems of the residual heat removal (RHR) system shall be OPERABLE[a] and, unless at least one recirculation pump is in operation, at least one shutdown cooling mode subsystem shall be capable of circulating reactor coolant[b] with each subsystem consisting of at least:

A.2 A.3

1. One OPERABLE RHR pump, and
2. One OPERABLE RHR heat exchanger.

LA.1

4.6 - SURVEILLANCE REQUIREMENTS

O. Residual Heat Removal - HOT SHUTDOWN

Two M.1 SR 3.4.7.1

At least one shutdown cooling mode subsystem of the residual heat removal system, recirculation pump or alternate method shall be verified to be capable of circulating reactor coolant at least once per 12 hours.

LA.2 A.2 moved to Required Action A.3 Required Action A.3 A.2

L.1

add proposed LCO Note

APPLICABILITY:

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

add proposed ACTIONS Note 2 A.4

ACTION:

ACTION A

1. With less than the above required RHR shutdown cooling mode subsystems OPERABLE, immediately initiate corrective action to return the required subsystems to OPERABLE status as soon as possible. Within 1 hour and at least once per 24 hours thereafter, demonstrate the operability of at least one alternate method capable of decay heat removal for each inoperable RHR shutdown cooling mode subsystem. Be in at least COLD SHUTDOWN within 24 hours.[a] A.5 A.6

A.2 LA.1

SR 3.4.7.1 ACTIONS NOTE 1

a. Each shutdown cooling subsystem is considered OPERABLE if it can be manually aligned (remote or local) in the shutdown cooling mode for removal of decay heat. The provisions of Specification 3.0.D are not applicable.

b. The RHR shutdown cooling subsystem may be removed from operation during hydrostatic testing. A.3

c. Whenever the two required RHR SDC mode subsystems are inoperable, if unable to attain COLD SHUTDOWN as required by this ACTION, maintain reactor coolant temperature as low as practical by use of alternate heat removal methods. A.6

3.6 - LIMITING CONDITIONS FOR OPERATION

ACTION A

Required Actions A.3 and A.4 Note

2. With no RHR shutdown cooling mode subsystem OPERABLE, immediately initiate corrective action to return at least one subsystem to OPERABLE status as soon as possible. Within 1 hour establish reactor coolant circulation with a recirculation pump or by an alternate method and monitor reactor coolant temperature and pressure at least once per hour.

L A.2

4.6 - SURVEILLANCE REQUIREMENTS

DISCUSSION OF CHANGES
ITS: 3.4.7 - RHR SHUTDOWN COOLING SYSTEM — HOT SHUTDOWN

ADMINISTRATIVE

A.1 In the conversion of the Quad Cities 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-1433, Rev. 1 (i.e., the Improved Standard Technical Specifications (ISTS)).

A.2 CTS 3.6.O requires two RHR shutdown cooling systems to be OPERABLE and, unless at least one recirculation pump is in operation, at least one shutdown cooling mode subsystem capable of circulating reactor coolant. CTS 3.6.O footnote (a) clarifies that an OPERABLE shutdown cooling subsystem is one that can be manually aligned in the shutdown cooling mode for removal of decay heat. This latter requirement (i.e., the subsystem can be manually aligned in the shutdown cooling mode for decay heat removal) also ensures proper circulation of reactor coolant since water obtained from the reactor vessel (via a reactor recirculation suction line) is pumped through an RHR heat exchanger (or heat exchange bypass valve) and routed to the reactor vessel via a recirculation discharge line. This will ensure the capability to circulate reactor coolant. Therefore, the detail in CTS 3.6.O that an RHR shutdown cooling subsystem must be capable of circulating reactor coolant is duplicative to the requirements in CTS 3.6.O footnote (a) and the definition of OPERABLE-OPERABILITY and has been deleted. Proposed LCO 3.4.7 will simply require both RHR shutdown cooling subsystems to be OPERABLE. Since this change does not result in any technical changes, the removal of the redundant requirement is considered administrative.

The content of CTS 3.6.O footnote(a) has been incorporated in proposed SR 3.4.7.1 (CTS 4.6.O) and will require the verification that each (refer to Discussion of Change M.1 below for further discussion) RHR shutdown cooling subsystem manual and power operated valve in the flow path, that is not locked, sealed or otherwise secured in position, is in the correct position, or can be aligned to the correct position. This proposed Surveillance is consistent with current requirements in CTS 4.6.O and is consistent with the format used in other Specifications in the CTS/ITS (e.g., ITS 3.1.7 for the Standby Liquid Control System) which require a system to be in standby. CTS 3/4.6.O does not require continuous reactor coolant circulation by a recirculation pump, an RHR shutdown cooling subsystem, or an alternate method as required by NUREG-0123 and NUREG-1433, Rev. 1. This deviation from the Standard Technical Specifications was approved by the NRC in the SER for Amendments 162 (Unit 1) and 158 (Unit 2) from John F. Stang (NRC) to D.L. Farrer (ComEd), dated September 21, 1995, since the RHR System and the RHR

ADMINISTRATIVE

A.2 (cont'd) Service Water System flow at Quad Cities 1 and 2 can not be throttled sufficiently to maintain constant temperature. Therefore, the system must be used continuously or intermittently to control temperature to be within the required cooldown/heatup rates. In MODE 3 below the RHR cut in permissive pressure, an RHR shutdown cooling subsystem is turned on and off as necessary to control temperature and to ensure the required cooldown or heatup rates of CTS 3/4.6.K (ITS 3.4.9) are not exceeded. The requirement to have a recirculation pump, RHR shutdown cooling subsystem or alternate method in operation is governed by plant procedures, except when both RHR shutdown cooling subsystems are inoperable. In this situation, circulation is required per CTS 3.6.O Action 2 (proposed ITS 3.4.7 ACTION A). Since, this change simply represents a change in presentation, this portion of the change is also administrative.

CTS 4.6.O requires the verification of the capability to circulate reactor coolant via an RHR shutdown cooling subsystem, a recirculation pump, or an alternate method every 12 hours. As indicated above, the proposed Surveillance has been written to ensure the requirements of the LCO are met. Since there are no actual requirements for recirculation pumps or an alternate method in the LCO, the option to verify the alternate method has been included as ITS 3.4.7 Required Action A.3 (refer to Discussion of Change LA.2 below for relocation of the recirculation pump as an alternate method of decay heat removal to the Bases) for the condition when both RHR shutdown cooling subsystems are inoperable since this is the only time an alternate method is required to be in operation. As indicated in the Bases, the alternate method must be functioning to help ensure temperature monitoring. This portion of the change is also considered administrative since CTS 3.6.O Action 2 currently requires an alternate method (or recirculation pump) to be established within 1 hour (ITS 3.4.7 Required Action A.3 first Completion Time).

A.3 The CTS 3.6.O footnote (b) allowance that the RHR shutdown cooling subsystem may be removed from operation, during hydrostatic testing has been deleted since these tests are not performed during MODE 3 operations. Since the footnote does not provide any additional allowance, its removal is considered administrative.

A.4 The proposed ACTION Note 2, "Separate Condition entry is allowed for each...," has been added to CTS 3.6.O Actions (ITS 3.4.7 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.

ADMINISTRATIVE (continued)

A.5 The requirement of CTS 3.6.O Action 1 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 since the reactor is currently required to be in MODE 4 within 24 hours (i.e., exit this Specification). Once in MODE 4, CTS 3.6.P and the ITS 3.4.8 both require the periodic verification of the availability of an alternate decay heat removal method. Since the frequency of the requirement in CTS 3.6.O Action 1 is of no consequence, its omission is considered an administrative change.

A.6 The CTS 3.6.O Action 1 footnote (c) requirement that if unable to attain cold shutdown when two 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 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

M.1 CTS 3.6.O (LCO 3.4.7) requires the OPERABILITY of both RHR shutdown cooling subsystems. However, CTS 4.6.O does not explicitly require a verification of the OPERABILITY of a RHR shutdown cooling subsystem. The Surveillance only requires the verification of one RHR shutdown cooling subsystem, a recirculation pump or an alternate method. The Surveillance Requirement has been changed to match the requirements of the LCO. Therefore, proposed SR 3.4.7.1 will require the verification that each RHR shutdown cooling subsystem is OPERABLE (refer to Discussion of Change A.2 above for further discussion). Since this change will require the verification every 12 hours that each RHR shutdown cooling subsystem is OPERABLE, this change is considered more restrictive on plant operations.

TECHNICAL CHANGES - LESS RESTRICTIVE

"Generic"

LA.1 The details in CTS 3.6.O.1 and CTS 3.6.O.2 of what constitutes an OPERABLE RHR shutdown cooling subsystem and the detail in CTS 3.6.O footnote (a) describing how the system can be manually aligned (remote or local) 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. Proposed SR 3.4.7.1 indicates that a system can be considered OPERABLE if it can be aligned to the correct position. The details for subsystem OPERABILITY are not necessary in ITS 3.4.7. 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 an acceptable alternate method in CTS 4.6.O and CTS 3.6.O Action 2 for circulating reactor coolant (i.e., a recirculation pump) is proposed to be relocated to the Bases. This detail is not necessary for assuring circulation of the reactor coolant when necessary. Proposed ITS 3.4.7 Required Action A.3 requires verification of reactor coolant circulation by an alternate method. 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 3.6.O does not allow one RHR shutdown cooling subsystem to be inoperable for performance of Surveillances. An allowance has been added (proposed LCO 3.4.7 Note) which allows one RHR shutdown cooling subsystem to be inoperable up to 2 hours to perform Surveillances. This change is necessary since tests are required that necessitate leaving the RHR Shutdown Cooling System in an inoperable status during the performance. This is acceptable because the core heat generation can be low enough and the heatup rate slow enough to allow some change to the RHR shutdown cooling subsystems or other operation requiring RHR flow interruption and loss of redundancy. This change is consistent with NUREG-1433, Rev. 1.

TECHNICAL CHANGES - LESS RESTRICTIVE (continued)

L.2 CTS 3.6.O requires two RHR shutdown cooling subsystems to be OPERABLE in Operational MODE 3, with reactor vessel pressure less than the RHR cut-in permissive setpoint. CTS 4.6.O requires a verification that a loop is capable of operation. CTS 3.0.D and 4.0.D require the LCO and Surveillances to be met prior to entry into the applicable mode or other specified conditions. The RHR System cannot be verified to be capable of circulating reactor coolant 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. CTS 3.0.D (LCO 3.0.4) is not applicable as specified in CTS 3.6.O Note a. Therefore, entry into the actions is permitted. However, both CTS 3.0.D and 4.0.D must be "not applicable" to perform the necessary verification following the reduction of pressure to below the cut-in permissive pressure setpoint. Therefore, a Note to CTS 4.6.O (proposed SR 3.4.7.1) has been added. Without this change, certain entries into the applicable operating conditions would result in intentional temporary noncompliance until the system is verified to be in operation.

RELOCATED SPECIFICATIONS

None

ITS 3.4.8

3.6 - LIMITING CONDITIONS FOR OPERATION

P. Residual Heat Removal - COLD SHUTDOWN

LCO 3.4.8

Two shutdown cooling mode subsystems of the residual heat removal (RHR) system shall be OPERABLE[a][b] and, unless at least one recirculation pump is in operation, at least one shutdown cooling mode subsystem shall be capable of circulating reactor coolant[b] with each subsystem consisting of at least: A.2

1. One OPERABLE RHR pump, and
2. One OPERABLE RHR heat exchanger. LA.1

APPLICABILITY:

add proposed LCO 3.4.8 Note L.1

OPERATIONAL MODE 4.

ACTION:

add proposed ACTION Note A.4

ACTION A

1. With less than the above required RHR shutdown cooling mode subsystems OPERABLE, within 1 hour and at least once per 24 hours thereafter, demonstrate the operability of at least one alternate method capable of decay heat removal for each inoperable RHR shutdown cooling mode subsystem.

4.6 - SURVEILLANCE REQUIREMENTS

P. Residual Heat Removal - COLD SHUTDOWN

SR 3.4.8.1

At least one shutdown cooling mode subsystem of the residual heat removal system, recirculation pump or alternate method shall be verified to be capable of circulating reactor coolant at least once per 12 hours. LA.2 A.2 moved to Required Action A.2

Required Action A.2

LA.1 A.2

a Each shutdown cooling subsystem is considered OPERABLE if it can be manually aligned (remote or local) in the shutdown cooling mode for removal of decay heat.

LCO 3.4.8 Note 1

b The RHR shutdown cooling subsystem(s) is not required to be OPERABLE and may be removed from operation during hydrostatic testing.

3.6 - LIMITING CONDITIONS FOR OPERATION

4.6 - SURVEILLANCE REQUIREMENTS

ACTION A

2. With no RHR shutdown cooling mode subsystem OPERABLE, immediately initiate corrective action to return at least one subsystem to OPERABLE status as soon as possible. Within 1 hour establish reactor coolant circulation with a recirculation pump or by an alternate method and monitor reactor coolant temperature and pressure at least once per hour.

L.2

LA.2

ADMINISTRATIVE

A.1 In the conversion of the Quad Cities 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-1433, Rev. 1 (i.e., the Improved Standard Technical Specifications (ISTS)).

A.2 CTS 3.6.P requires two RHR shutdown cooling systems to be OPERABLE and, unless at least one recirculation pump is in operation, at least one shutdown cooling mode subsystem capable of circulating reactor coolant. CTS 3.6.P footnote (a) clarifies that an OPERABLE shutdown cooling subsystem is one than can be manually aligned in the shutdown cooling mode for removal of decay heat. This latter requirement (i.e., the subsystem can be manually aligned in the shutdown cooling mode for decay heat removal) also ensures proper circulation of reactor coolant since water obtained from the reactor vessel (via a reactor recirculation suction line) is pumped through an RHR heat exchanger (or heat exchange bypass valve) and routed to the reactor vessel via a recirculation discharge line. This will ensure the capability to circulate reactor coolant. Therefore, the detail in CTS 3.6.P that an RHR shutdown cooling subsystem must be capable of circulating reactor coolant is duplicative to the requirement in CTS 3.6.P footnote (a) and the definition of OPERABLE-OPERABILITY and has been deleted. Proposed LCO 3.4.8 will simply require both RHR shutdown cooling subsystems to be OPERABLE. Since this change does not result in any technical changes, the removal of the redundant requirement is considered administrative.

The content of CTS 3.6.P footnote (a) has been incorporated in proposed SR 3.4.8.1 (CTS 4.6.P) and will require the verification that each (refer to Discussion of Change M.1 below for further discussion) RHR shutdown cooling subsystem manual and power operated valve in the flow path, that is not locked, sealed, or otherwise secured in position, is in the correct position, or can be aligned to the correct position. This proposed Surveillance is consistent with current requirements in CTS 4.6.P and is consistent with the format used in other Specifications in the CTS/ITS (e.g., ITS 3.1.7 for the Standby Liquid Control System) which require a system to be in standby. CTS 3/4.6.P does not require continuous reactor coolant circulation by a recirculation pump, an RHR shutdown cooling subsystem, or an alternate method as required by NUREG-0123 and NUREG-1433, Rev. 1. This deviation from the Standard Technical Specifications was approved by the NRC in the SER for Amendments 162 (Unit 1) and 158 (Unit 2) from John F. Stang (NRC) to D.L. Farrer (ComEd), dated September 21, 1995, since the RHR System and the RHR Service Water System flow at Quad Cities 1 and 2 can not be throttled sufficiently to maintain constant

ADMINISTRATIVE

A.2 (cont'd) temperature. Therefore, the system must be used continuously or intermittently to control temperature and be within the required cooldown/heatup rates. In MODE 4 RHR shutdown cooling subsystem is turned on and off as necessary to control temperature and to ensure the required cooldown or heatup rates of CTS 3/4.6.K (ITS 3.4.9) are not exceeded. The requirement to have a recirculation pump, RHR shutdown cooling subsystem or alternate method in operation is governed by plant procedures, except when both RHR shutdown cooling subsystems are inoperable. In this situation, circulation is required per CTS 3.6.P Action 2 (proposed ITS 3.4.8 ACTION A). Since this change simply represents a change in presentation, this portion of the change is also administrative.

CTS 4.6.P requires the verification of the capability to circulate reactor coolant via an RHR shutdown cooling subsystem, a recirculation pump, or an alternate method every 12 hours. As indicated above, the proposed Surveillance has been written to ensure the requirements of the LCO are met. Since there are no actual requirements for recirculation pumps or an alternate method in the LCO, the option to verify the alternate method has been included as ITS 3.4.8 Required Action A.3 (refer to Discussion of Change LA.2 below for relocation of the recirculation pump as an alternate method of decay heat removal to the Bases) for the condition when both RHR shutdown cooling subsystems are inoperable since this is the only time an alternate method is required to be in operation. As indicated in the Bases, the alternate method must be functioning to help ensure temperature monitoring. This portion of the change is also considered administrative since CTS 3.6.P Action 2 currently requires an alternate method (or recirculation pump) to be established within 1 hour (ITS 3.4.8 Required Action A.2 first Completion Time).

A.3 Not used.

A.4 The proposed ACTION Note, "Separate Condition entry is allowed for each RHR shutdown cooling subsystem" has been added to CTS 3.6.P Actions (ITS 3.4.8 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.

TECHNICAL CHANGES - MORE RESTRICTIVE

None

TECHNICAL CHANGES - LESS RESTRICTIVE

"Generic"

LA.1 The details in CTS 3.6.P.1 and CTS 3.6.P.2 of what constitutes an OPERABLE RHR shutdown subsystem and the detail in CTS 3.6.P footnote (a) describing how the system can be aligned "remote or local" 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. Proposed SR 3.4.8.1 indicates that a system can be considered OPERABLE if it can be aligned to the correct position. The details for subsystem OPERABILITY are not necessary in ITS 3.4.8. 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 an acceptable alternate method in CTS 4.6.P and CTS 3.6.P Action 2 for circulating reactor coolant (i.e., a recirculation pump) is proposed to be relocated to the Bases. This detail is not necessary for assuring circulation of the reactor coolant when necessary. Proposed ITS 3.4.8 Required Action A.2 requires verification of reactor coolant circulation by an alternate method. 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 3.6.P does not allow one RHR shutdown cooling subsystem to be inoperable for performance of Surveillances. An allowance has been added (proposed LCO 3.4.8 Note) which allows one RHR shutdown cooling subsystem to be inoperable up to 2 hours to perform Surveillances. This change is necessary since tests are required that necessitate placing the RHR Shutdown Cooling System in an inoperable status. This is acceptable because the core heat

TECHNICAL CHANGES - LESS RESTRICTIVE

L.1 (cont'd) generation can be low enough and the heatup rate slow enough to allow some changes to the RHR shutdown cooling subsystems or other operations requiring RHR flow interruptions and loss of redundancy. This change is consistent with BWR ISTS, NUREG-1433, Rev. 1.

L.2 With no RHR shutdown cooling subsystem OPERABLE, CTS 3.6.P Action 2 requires the plant to immediately initiate corrective action to return at least one subsystem to OPERABLE status as soon as possible. In addition, the same action requires reactor coolant circulation to be established with a recirculation pump or by an alternate method within 1 hour and to monitor reactor coolant temperature and pressure at least once per hour. During MODE 4 operations, the core decay heat may be low enough such that alternative methods of decay heat removal are sufficient to remove core decay heat to maintain or reduce temperature. CTS 3.6.P Action 1 (ITS 3.4.8 Required Action A.1) requires the verification that an alternate method of decay heat removal is available for each inoperable RHR shutdown cooling subsystem within one hour and once per 24 hours thereafter. In this MODE, it is acceptable to take both RHR subsystems out of service to perform required maintenance operations since the Technical Specifications provide sufficient requirements to ensure decay heat removal methods are available. Therefore, the requirement in CTS 3.6.P Action 2 to immediately initiate corrective action to return at least one subsystem to OPERABLE status as soon as possible has been deleted. This allowance has been approved at both Dresden and LaSalle and is consistent with BWR ISTS, NUREG-1433, Rev. 1.

RELOCATED SPECIFICATIONS

None

3.6 - LIMITING CONDITIONS FOR OPERATION

K. Pressure/Temperature Limits

LCO 3.4.9 The primary system coolant system temperature and reactor vessel metal temperature and pressure shall be limited as specified below:

1. Pressure Testing:

SR 3.4.9.1 a. The reactor vessel metal temperature and pressure shall be maintained within the Acceptable Regions as shown on Figure 3.6.K-1 with the rate of change of the primary system coolant temperature ≤ 20°F per hour, or

SR 3.4.9.1 b. The rate of change of the primary system coolant temperature shall be ≤100°F per hour when reactor vessel metal temperature and pressure is maintained within the Acceptable Regions as shown on Figure 3.6.K-2.

2. Non-Nuclear Heatup and Cooldown and low power PHYSICS TESTS:

SR 3.4.9.1 a. The reactor vessel metal temperature and pressure shall be maintained within the Acceptable Regions as shown on Figure 3.6.K-2, and

SR 3.4.9.1 b. The rate of change of the primary system coolant temperature shall be ≤100°F per hour.

4.6 - SURVEILLANCE REQUIREMENTS

K. Pressure/Temperature Limits

1. During non-nuclear heatup or cooldown, and pressure testing operations, at least once per 30 minutes,

SR 3.4.9.1 a. The rate of change of the primary system coolant temperature shall be determined to be within the heatup and cooldown rate limits, and

b. The reactor vessel metal temperature and pressure shall be determined to be within the Acceptable Regions on Figures 3.6.K-1 through 3.6.K-2.

2. For reactor critical operation, determine within 15 minutes prior to the withdrawal of control rods and at least once per 30 minutes during primary system heatup or cooldown, L.2

SR 3.4.9.1 a. The rate of change of the primary system coolant temperature to be within the limits, and

SR 3.4.9.2 b. The reactor vessel metal temperature and pressure to be within the Acceptable Region on Figure 3.6.K-3.

~~3. The reactor vessel material surveillance specimens shall be removed and examined, to determine changes in reactor pressure vessel material properties in accordance with 10CFR Part 50, Appendix H.~~

A.4

A.1

ITS 3.4.9

A.5

add proposed SR 3.4.9.7 Note

3.6 - LIMITING CONDITIONS FOR OPERATION

3. Nuclear Heatup and Cooldown:

SR 3.4.9.2 a. The reactor vessel metal temperature and pressure shall be maintained within the Acceptable Region as shown on Figure 3.6.K-3, and

SR 3.4.9.1 b. The rate of change of the primary system coolant temperature shall be ≤100°F per hour.

SR 3.4.9.5 SR 3.4.9.6 SR 3.4.9.7 4. The reactor vessel flange and head flange temperature ≥83 °F when reactor vessel head bolting studs are under tension.

4.6 - SURVEILLANCE REQUIREMENTS

4. The reactor vessel flange and head flange temperature shall be verified to be ≥83°F:

a. In OPERATIONAL MODE 4 when the reactor coolant temperature is:

SR 3.4.9.7 1) ≤113°F, at least once per 12 hours.

SR 3.4.9.6 2) ≤93°F, at least once per 30 minutes.

SR 3.4.9.5 b. Within 30 minutes prior to and at least once per 30 minutes during tensioning of the reactor vessel head bolting studs. A.6

add proposed SR 3.4.9.6 Note

A.5

add proposed Notes for Condition A and C

A.2

APPLICABILITY:

At all times.

ACTION:

With any of the above limits exceeded,

ACTION A

1. Restore the reactor vessel metal temperature and/or pressure to within the limits within 30 minutes without exceeding the applicable primary system coolant temperature rate of change limit, and A.3

ACTION C

2. Perform an engineering evaluation to determine the effects of the out-of-limit condition on the structural integrity of the reactor coolant system and determine that the reactor coolant system remains acceptable for continued operations within 72 hours, or LA.2 L.1

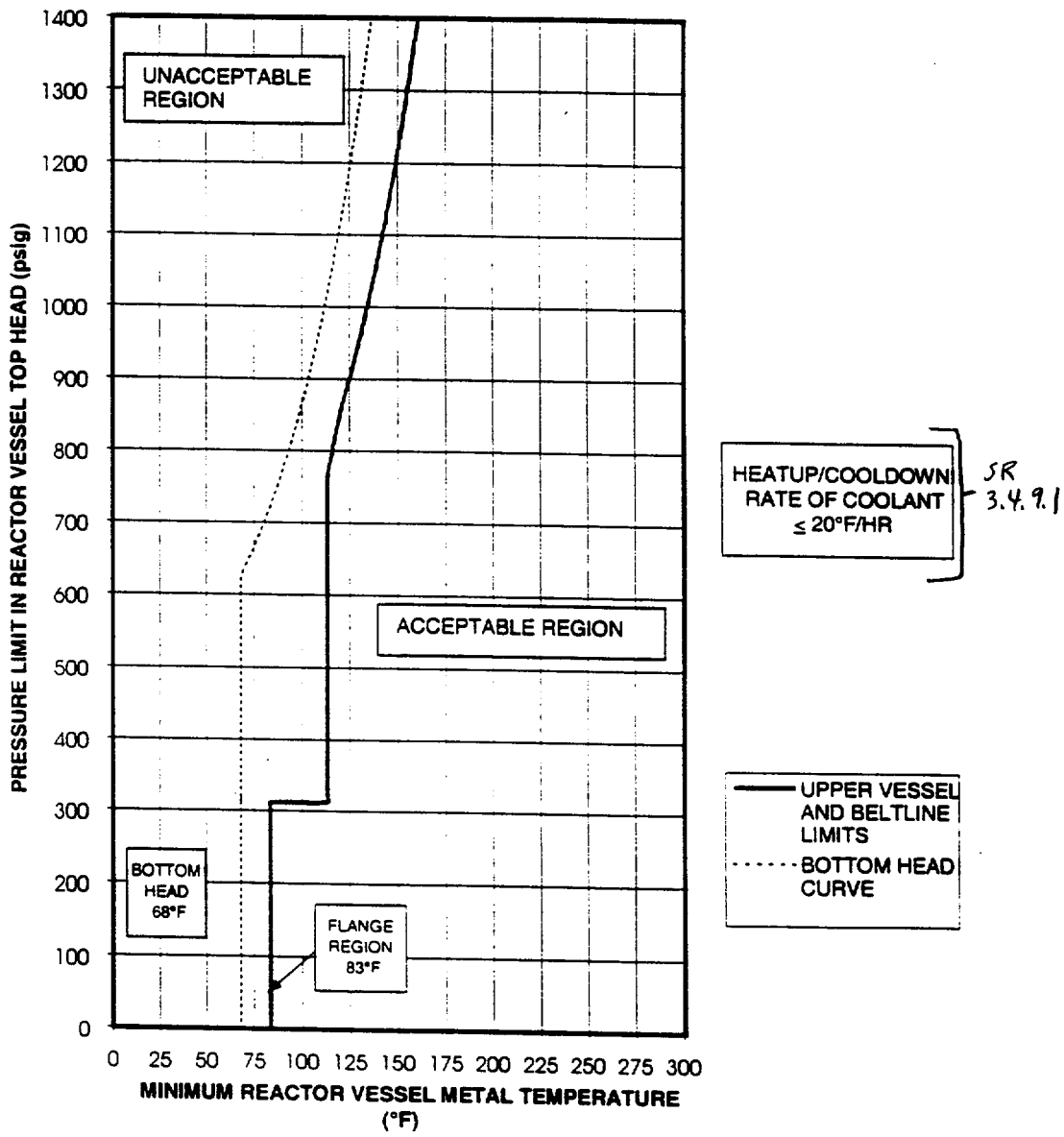
ACTION B

3. Be in at least HOT SHUTDOWN within 12 hours and in COLD SHUTDOWN within the following 24 hours.

3.4.9-1

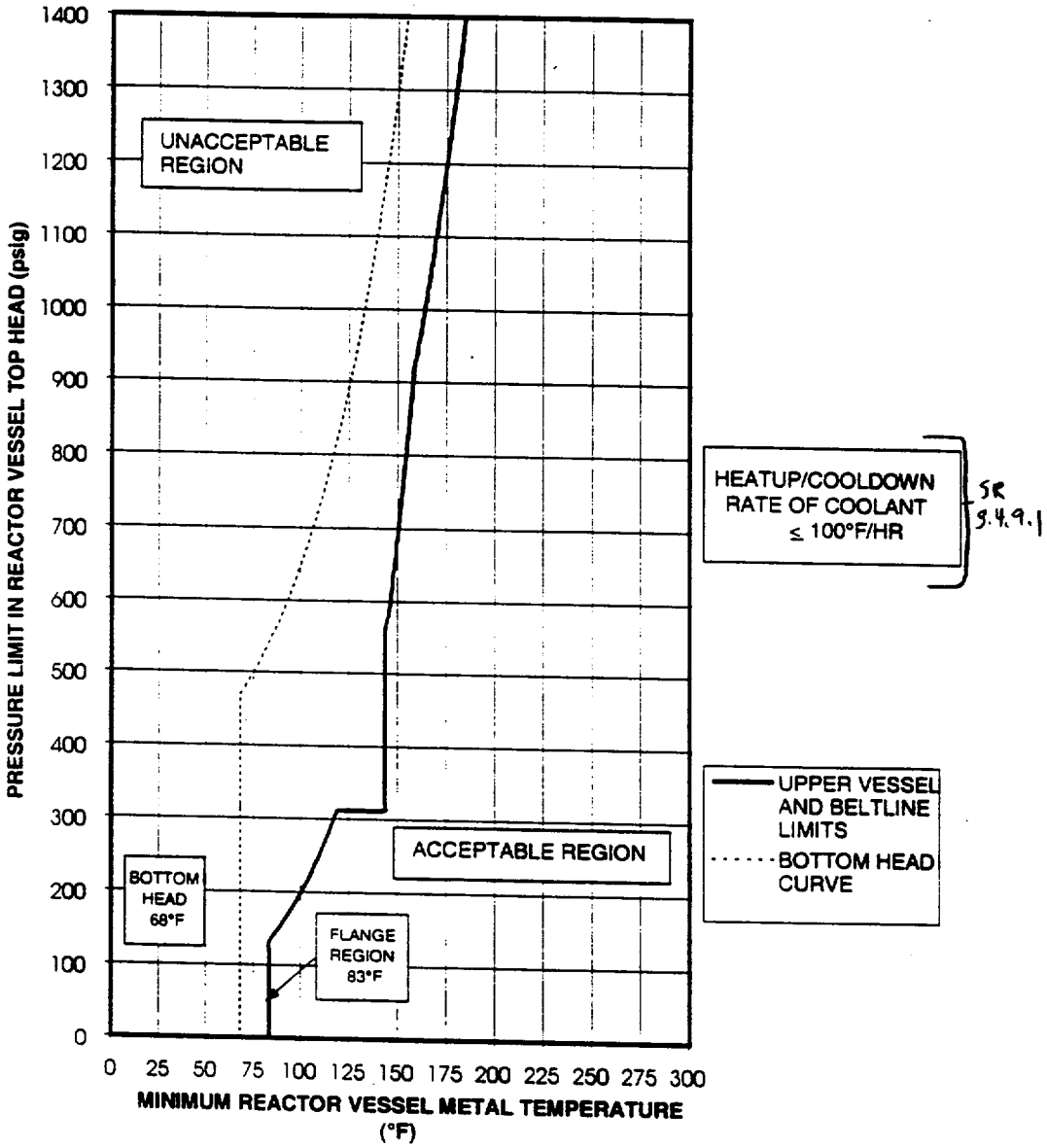
FIGURE 3.6.K-1

PRESSURE – TEMPERATURE LIMITS FOR PRESSURE TESTING – VALID TO 32 EFPY



3.4.9-2

FIGURE 3.6.K-2
PRESSURE – TEMPERATURE LIMITS FOR NON-NUCLEAR
HEATUP/COOLDOWN – VALID TO 32 EFPY

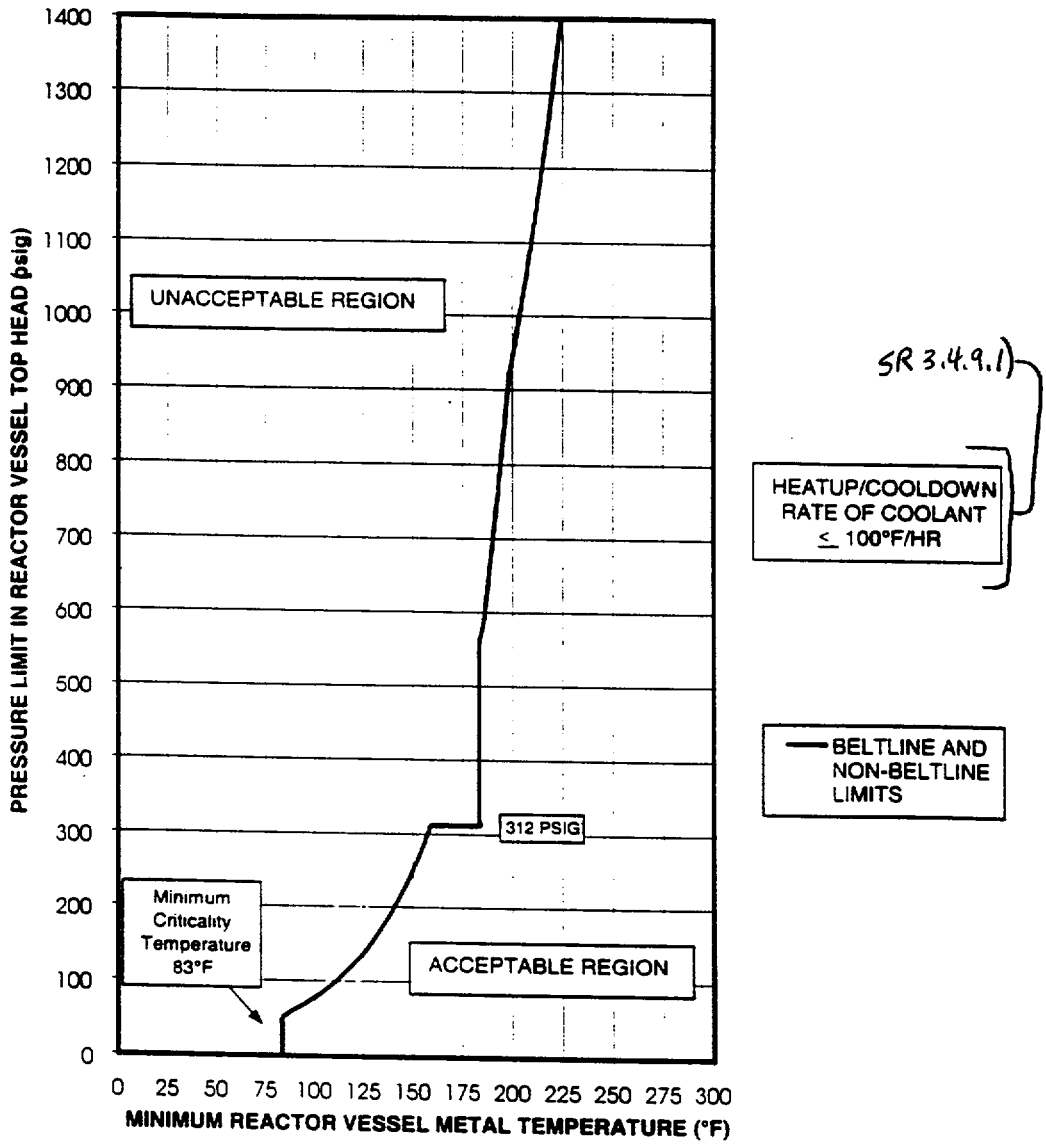


A.1

ITS 3.4.9

3.4.9-3

FIGURE 3.6.K-3
PRESSURE – TEMPERATURE LIMITS FOR CRITICAL CORE OPERATIONS – VALID TO 32 EFPY



3.6 - LIMITING CONDITIONS FOR OPERATION

D. Idle Recirculation Loop Startup

SR 3.4.9.3 — An idle recirculation loop shall not be started unless the temperature differential between the reactor pressure vessel steam space coolant and the bottom head drain line coolant is ≤145°F[a], and:

LA.3

1. When both loops have been idle, unless the temperature differential between the reactor coolant within the idle loop to be started up and the coolant in the reactor pressure vessel is ≤50°F, or — A.7

2. When only one loop has been idle, unless the temperature differential between the reactor coolant within the idle and operating recirculation loops is ≤50°F and the speed of the operating pump is ≤45% of rated pump speed. — A.8 — SR 3.4.9.4 — LA.1

APPLICABILITY:

SR 3.4.9.3
SR 3.4.9.4 OPERATIONAL MODE(s) 1, 2, 3 and 4 — during recirculation pump startup — A.7

ACTION:

With temperature differences and/or flow rates exceeding the above limits, suspend startup of any idle recirculation loop. — LA.1

Add proposed ACTIONS A, B, and C — M.1

4.6 - SURVEILLANCE REQUIREMENTS

D. Idle Recirculation Loop Startup — LA.1

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.

SR 3.4.9.3
SR 3.4.9.4

---

a Below 25 psig reactor pressure this temperature differential is not applicable. — M.2

DISCUSSION OF CHANGES
ITS: 3.4.9 - RCS PRESSURE AND TEMPERATURE (P/T) LIMITS

ADMINISTRATIVE

A.1 In the conversion of the Quad Cities 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-1433, Rev. 1 (i.e., the Improved Standard Technical Specifications (ISTS)).

A.2 The CTS 3.6.K Action 2 to "perform an engineering evaluation..." 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 3.4.9 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 ISTS, NUREG-1433, Rev. 1. Because this is an enhanced presentation of the existing intent, the proposed change is administrative.

A.3 The CTS 3.6.K Action 1 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 3.4.9 Required Action C.1. This interpretation of the intent is supported by the BWR ISTS, NUREG-1433, Rev. 1. Because this is an enhanced presentation of the existing intent, the proposed change is administrative.

A.4 CTS 4.6.K.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 and eliminating the Technical Specification details that are also found in Appendix H, is considered a presentation preference which is administrative.

ADMINISTRATIVE (continued)

A.5 CTS 4.6.K.4.a requires periodic verification that reactor vessel and head flange temperatures are ≥ 83°F. The Frequency of this verification change is based on reactor coolant system temperature. Notes have been provided in proposed SR 3.4.9.6 and 3.4.9.7 to clarify the current intent in CTS 4.6.K.4.a of allowing entry into the applicable conditions (i.e., ≤ 113°F and ≤ 93°F) without having performed these Surveillance Requirements. Since this requirement is currently only performed during the specified conditions (i.e., when ≤ 113°F and ≤ 93°F), these changes (the addition of the two Notes) are considered administrative.

A.6 The CTS 4.6.K.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.A and proposed SR 3.0.1, which require the Surveillance to be current when in the applicable Mode or condition. CTS 4.0.C 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. CTS 3.6.K Action 1 (ITS 3.4.9 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 when CTS 4.6.K.4.b (proposed SR 3.4.9.5) is not current. Therefore, this change is considered administrative.

A.7 The CTS 3.6.D 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 3.4.9 LCO statement. The actual description of the requirements and the limits are found in proposed SR 3.4.9.3 and SR 3.4.9.4. 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 3.6.D.1 and 3.6.D.2 (proposed SR 3.4.9.4) ensure the temperature difference between the idle loop and the RPV coolant is acceptable. The CTS 3.6.D.2 requirement to monitor the temperature difference between an idle loop and an operating loop is unnecessary and has been deleted since it is redundant to the loop-to-coolant requirement of CTS 3.6.D.1 (proposed SR 3.4.9.4). However, the loop-to-coolant temperature check may use the operating loop temperature as representative of "coolant temperature."

TECHNICAL CHANGES - MORE RESTRICTIVE

M.1 The CTS 3.6.D Action required to be taken when a recirculation pump is started without having met the temperature requirements has been changed. Currently, the CTS 3.6.D Action only states to suspend the startup of a recirculation loop. This however, does not provide an action if the loop is already operating. Proposed ITS 3.4.9 adds ACTIONS A, B, and C 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.

M.2 The CTS 3.6.D footnote a allowance that the differential temperature between the reactor pressure vessel steam space coolant and the bottom head drain line coolant is not applicable below 25 psig reactor pressure has been deleted. Therefore, ITS SR 3.4.9.3 will require the differential temperature requirement between the reactor pressure vessel coolant and the bottom head coolant to be within limits (< 145°F) in MODES 1, 2, 3, and 4 during a recirculation pump startup. Since, the limit must be met at all times in these MODES, this change is more restrictive. This change is necessary to minimize thermal stresses resulting from the startup of an idle recirculation pump.

TECHNICAL CHANGES - LESS RESTRICTIVE

"Generic"

LA.1 The details of CTS 3.6.D (and its associated Action) and CTS 4.6.D relating to operational limits (loop flow and pump speed) 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 and pump speed are considered operational limits since they are not directly related to the ability of the system to perform its safety analysis functions. The flow rate and pump speed are 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.

TECHNICAL CHANGES - LESS RESTRICTIVE (continued)

LA.2 The detail in CTS 3.6.K Action 2 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 3.4.9 Required Action 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 3.6.D 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 3.4.9.3 to verify the difference between the bottom head coolant temperature and the reactor pressure vessel coolant temperature is sufficient to ensure that the differential temperature is met prior to a startup of a recirculation pump. The Bases for SR 3.4.9.3 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.

"Specific"

L.1 CTS 3.6.K Action 2 specifies a Completion Time of 72 hours for the required engineering evaluation with an LCO applicability of "at all times." Proposed ITS 3.4.9, Required Action C.2, (applicable when in conditions other than MODES 1, 2, and 3) requires completion "prior to entering MODE 2 or 3." While Required Action C.2 is intended to be initiated without delay, it is not restricted to a specified Completion Time, only by a restriction on returning to (entering) operating MODES (i.e., 1, 2, or 3) where additional stresses (heatup/criticality) may be imposed. This change is consistent with BWR ISTS, NUREG-1433, Rev. 1, and is considered acceptable since continued plant operation is prohibited until RCS integrity is assured.

TECHNICAL CHANGES - LESS RESTRICTIVE (continued)

L.2 CTS 4.6.K.2.a requires the rate of change of primary system coolant temperature to be determined within limits 15 minutes prior to withdrawal of control rods and at least once per 30 minutes during primary system heatup or cooldown. The requirement to verify the rate of change during the 15 minute period prior to withdrawal of control rods has been deleted, however, the Frequency of once every 30 minutes has been retained as proposed in SR 3.4.9.1 during heatup and cooldown. The primary coolant temperature is not expected to change significantly until the reactor becomes critical, therefore, this Surveillance Requirement is not necessary. CTS 4.6.K.2.b, the requirement to verify the reactor vessel metal temperature and pressure to be within the Acceptable Region of the critical core operation curve (CTS Figure 3.6.K-3) once within 15 minutes prior to control rod withdrawal for the purpose of achieving criticality, is being retained in ITS SR 3.4.9.2. The proposed Frequencies of proposed SR 3.4.9.1 and 3.4.9.2 are considered acceptable to ensure the RCS P/T limits are met during critical operations. This change is consistent with BWR ISTS, NUREG-1433, Rev. 1.

RELOCATED SPECIFICATIONS

None

A.1

ITS 3.4.10

3.6 - LIMITING CONDITIONS FOR OPERATION

L. Reactor Steam Dome Pressure

LCO 3.4.10 The pressure in the reactor steam dome shall be ≤1005 psig.

4.6 - SURVEILLANCE REQUIREMENTS

L. Reactor Steam Dome Pressure

SR 3.4.10.1 The reactor steam dome pressure shall be verified to be ≤1005 psig at least once per 12 hours.

APPLICABILITY:

OPERATIONAL MODE(s) 1[a] and 2[a]

M.1

ACTION:

ACTION A With the reactor steam dome pressure >1005 psig, reduce the pressure to ≤1005 psig within 15 minutes or be in at least

ACTION B HOT SHUTDOWN within 12 hours.

M.1

a Not applicable during anticipated transients.

DISCUSSION OF CHANGES
ITS: 3.4.10 - REACTOR STEAM DOME PRESSURE

ADMINISTRATIVE

A.1 In the conversion of the Quad Cities 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-1433, Rev. 1 (i.e., the Improved Standard Technical Specifications (ISTS)).

TECHNICAL CHANGES - MORE RESTRICTIVE

M.1 The CTS 3.6.L footnote (a) 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 3.4.10 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

None

RELOCATED SPECIFICATIONS

None

3.6 - LIMITING CONDITIONS FOR OPERATION

N. Structural Integrity

The structural integrity of ASME Code Class 1, 2 and 3 components shall be maintained in accordance with Specification 4.6.N.

APPLICABILITY:

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

ACTION:

1. 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 limits 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.

2. 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).

3. 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.

4.6 - SURVEILLANCE REQUIREMENTS

N. Structural Integrity

No additional Surveillance Requirements other than those required by Specification 4.0.E.

DISCUSSION OF CHANGES
CTS: 3/4.6.N - STRUCTURAL INTEGRITY

ADMINISTRATIVE

None

TECHNICAL CHANGES - MORE RESTRICTIVE

None

TECHNICAL CHANGES - LESS RESTRICTIVE

None

RELOCATED SPECIFICATIONS

R.1 The CTS 3/4.6.N 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 3/4.6.N did not satisfy the NRC Policy Statement Technical Specification screening criteria as documented in the Application of Selection Criteria to the Quad Cities 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 3/4.6-1 through B 3/4.6-8) have been completely replaced by revised Bases reflecting the format and applicable content of the Quad Cities 1 and 2 ITS Section 3.4, consistent with the BWR ISTS, NUREG-1433, Rev. 1. The revised Bases are as shown in the Quad Cities 1 and 2 ITS Bases. In addition, blank pages 3/4.6-13, 3/4.6-14, 3/4.6-15, 3/4.6-17, and 3/4.6-18 have been deleted.

3.4 REACTOR COOLANT SYSTEM (RCS)

⟨CTS⟩ 3.4.1 Recirculation Loops Operating

LCO 3.4.1 Two recirculation loops with matched flows shall be in operation,

⟨3.6.A⟩

⟨3.6.A Act 1.b, 1.c, 1.d⟩

⟨3.6.C⟩⟨M.1⟩

OR

One recirculation loop ~~may~~ shall be in operation ~~provided~~ with the following limits ~~are~~ 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~~}~~; 1

b. LCO 3.2.2, "MINIMUM CRITICAL POWER RATIO (MCPR)," single loop operation limits ~~{~~specified in the COLR~~}~~; ~~and~~ 2

c. LCO 3.3.1.1, "Reactor Protection System (RPS) Instrumentation," Function 2.b (Average Power Range Monitors Flow Biased ~~Simulated Thermal Power~~ Neutron Flux—High), Allowable Value of Table 3.3.1.1-1 is reset for single loop operation~~.~~; and 3

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 is reset for single loop operation 1 2

APPLICABILITY: MODES 1 and 2.

ACTIONS

⟨3.6.A Act 1⟩

CONDITION	REQUIRED ACTION	COMPLETION TIME
A* Requirements of the LCO not met. for reasons other than Condition A or B C 2	~~A~~.1 C Satisfy the requirements of the LCO.	24 hours

*Pending resolution of stability issue. 2

Insert Actions

(continued)

2

<CTS>

Insert ACTIONS

<3.6.A Act 2>	A.	No recirculation loops in operation.	A.1 Be in MODE 2. AND A.2 Be in MODE 3.	8 hours 12 hours
<3.6.C Act>	B.	Recirculation loop flow mismatch not within limits.	B.1 Declare the recirculation loop with lower flow to be "not in operation."	2 hours

⟨CTS⟩

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
D. Required Action and associated Completion Time of Condition A not met. ⟨3.6.A Act 1.⟩ OR No recirculation loops in operation.	D.1 Be in MODE 3.	12 hours 2

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.4.1.1 ⟨4.6.C⟩	----------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. ≤ ~~{10}~~% of rated core flow when operating at < ~~{70}~~% of rated core flow; and 1 b. ≤ ~~{5}~~% of rated core flow when operating at ≥ ~~{70}~~% of rated core flow. 1	24 hours 3

JUSTIFICATION FOR DEVIATIONS FROM NUREG-1433, REVISION 1
ITS: 3.4.1 - RECIRCULATION LOOPS OPERATING

1. The brackets have been removed and the proper plant specific information/value has been provided.

2. The "Recirculation Loops Operating" Specification has been revised to reflect Current Technical Specifications requirements except where justified in the Discussion of Changes. When ComEd completes resolution of the long-term stability issue, the ITS will be revised appropriately.

3. Changes have been made to reflect plant specific design requirements related to flow mismatch.

⟨CTS⟩

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.2 Jet Pumps

⟨3.6.B⟩ LCO 3.4.2 All jet pumps shall be OPERABLE.

APPLICABILITY: MODES 1 and 2.

ACTIONS

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

⟨CTS⟩

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.4.2.1	----------NOTES---------- 1. Not required to be performed until 4 hours after associated recirculation loop is in operation. ⟨9.6.B.1⟩ 2. Not required to be performed until 24 hours after > 25% RTP. ⟨9.6.B.2⟩ ----------	
	Verify at least one of the following criteria (a, b, or c) is satisfied for each operating recirculation loop: a. Recirculation pump flow to speed ratio differs by ≤ 5% from established patterns, and jet pump loop flow to recirculation pump speed ratio differs by ≤ 5% from established patterns. (10) b. Each jet pump diffuser to lower plenum differential pressure differs by ≤ 20% from established patterns. c. Each jet pump flow differs by ≤ 10% from established patterns.	24 hours [1]

Reviewer's Note: An acceptable option to these criteria for jet pump OPERABILITY can be found in the BWR/6 ITS, NUREG-1434. [2]

JUSTIFICATION FOR DEVIATIONS FROM NUREG-1433, REVISION 1
ITS: 3.4.2 - JET PUMPS

1. The specific criteria of BWR ISTS, NUREG-1433, Rev. 1, second part of SR 3.4.2.1.a and SR 3.4.2.1.b, which are methods of verifying the jet pumps are OPERABLE, are not included in the Quad Cities 1 and 2 ITS 3.4.2. In addition, the limit in the first part of SR 3.4.2.1.a has been increased from 5% to 10%. These changes are consistent with the current Quad Cities 1 and 2 licensing basis. The subsequent requirement is relabeled to reflect this change.

2. This Reviewer's Note has been deleted. This Note provides the location of an alternative set of criteria that is not used at Quad Cities 1 and 2. This is not meant to be retained in the final version of the plant specific submittal.

⟨CTS⟩ 3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.3 Safety and Relief Valves (S/RVs)

⟨3.6.E⟩ ⟨3.6.F⟩ LCO 3.4.3 The safety function of ~~[0]~~ 9 ~~S/RVs~~ safety valves shall be OPERABLE.

AND

The relief function of 5 relief valves shall be OPERABLE.

⟨Appl 3.6.E⟩ ⟨Appl 3.6.F⟩ APPLICABILITY: MODES 1, 2, and 3.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
⟨3.6.E Act 2⟩ A. One ~~or two~~ ~~required~~ ~~S/RVs~~ relief valve inoperable.	A.1 Restore the relief valve ~~required~~ ~~S/RVs~~ to OPERABLE status.	14 days
⟨3.6.E Act 1⟩ B. Required Action and associated Completion Time of Condition A not met.	B.1 Be in MODE 3.	12 hours
	AND	
	B.2 Be in MODE 4.	36 hours
OR		
⟨3.6.F Act 2⟩ ⟨3.6.F Act 3⟩ ~~Three~~ One or more ~~required~~ ~~S/RVs~~ safety valves inoperable.		

Two or more relief valves inoperable

OR

<CTS>

5 TSTF-298 changes not shown

safety valves 1

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
SR 3.4.3.1 Verify the safety function lift setpoints of the ~~required~~ S/RVs are as follows: 3 1 safety valves Number of S/RVs — Setpoint (psig) [4] — [1090 ± 32.7] [4] — [1100 ± 33.0] [3] — [1110 ± 33.3] Following testing, lift settings shall be within ± 1%. 4 (3.6.E) (4.6.E)	~~[In accordance with the Inservice Testing Program or [18] months]~~ 2
SR 3.4.3.2 ----------NOTE---------- Not required to be performed until 12 hours after reactor steam pressure and flow are adequate to perform the test. ---------- 3 Verify each ~~required~~ S/RV opens when manually actuated. relief valve 1 (DOC A.4)	~~[18]~~ 24 months ~~[on a STAGGERED TEST BASIS for each valve solenoid]~~ 2 3

1	1135	± 11.3
2	1240	± 12.4
2	1250	± 12.5
4	1260	± 12.6

2

(DOC A.4)

SR 3.4.3.3 ----- NOTE ----
Valve actuation may be excluded.

Verify each relief valve actuates on an actual or simulated automatic initiation signal | 24 months

1

JUSTIFICATION FOR DEVIATIONS FROM NUREG-1433, REVISION 1
ITS: 3.4.3 - SAFETY AND RELIEF VALVES

1. The current Quad Cities 1 and 2 licensing basis includes requirements for both safety and relief valves. Therefore, the relief valve requirements have been added.

2. The brackets have been removed and the proper plant specific information/value has been provided.

3. The brackets have been removed and the information/value deleted since it does not apply.

4. The requirement in ISTS SR 3.4.3.1, that lift settings shall be within $\pm$ 1% following testing, has been deleted since the tolerance specified for OPERABILITY is $\pm$ 1%.

5. TSTF-298 provides an allowance, in ISTS SR 3.4.3.1, for safety/relief valves to be replaced with spare Operable safety/relief valves having lower setpoints. TSTF-298 has not been adopted since Quad Cities 1 and 2 do not currently have analyses to support this allowance.

⟨CTS⟩

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.4 RCS Operational LEAKAGE

⟨3.6.H⟩

LCO 3.4.4 RCS operational LEAKAGE shall be limited to:

a. No pressure boundary LEAKAGE;

b. ≤ 5 gpm unidentified LEAKAGE; ~~and~~

c. ≤ ~~30~~ 25 gpm total LEAKAGE averaged over the previous 24 hour period; ~~and~~

d. ≤ 2 gpm increase in unidentified LEAKAGE within the previous ~~[4]~~ 24 hour period in MODE 1.

1

APPLICABILITY: MODES 1, 2, and 3.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
⟨3.6.H Act 2⟩ A. Unidentified LEAKAGE not within limit. OR Total LEAKAGE not within limit.	A.1 Reduce LEAKAGE to within limits.	4 hours
⟨3.6.H Act 3⟩ B. Unidentified LEAKAGE increase not within limit.	B.1 Reduce unidentified LEAKAGE increase to within limits. OR	4 hours (continued)

2

⟨CTS⟩

ACTIONS

	CONDITION	REQUIRED ACTION	COMPLETION TIME
⟨3.6.H Act 3⟩	B. (continued)	B.2 Identify ~~Verify~~ source of unidentified LEAKAGE increase is not ~~service sensitive type 304 or type 316 austenitic stainless steel~~ IGSCC susceptible material	4 hours 3
⟨3.6.H Act 1⟩ ⟨3.6.H Act 2⟩ ⟨3.6.H Act 3.b⟩	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
⟨4.6.H⟩	SR 3.4.4.1	Verify RCS unidentified and total LEAKAGE and unidentified LEAKAGE increase are within limits.	~~8~~ 12 hours 4

JUSTIFICATION FOR DEVIATIONS FROM NUREG-1433, REVISION 1
ITS: 3.4.4 - RCS OPERATIONAL LEAKAGE

1. The brackets have been removed and the proper plant specific information/value has been provided.

2. Editorial change has been made to achieve consistency with the Writer's Guide.

3. Changes have been made to reflect the plant specific nomenclature.

4. 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.

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.5 RCS Pressure Isolation Valve (PIV) Leakage

LCO 3.4.5 The leakage from each RCS PIV shall be within limit.

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.

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 must have been verified to meet SR 3.4.5.1 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, de-activated automatic, or check valve.	4 hours
	AND		
	A.2	Isolate the high pressure portion of the affected system from the low pressure portion by use of a second closed manual, de-activated automatic, or check valve.	72 hours
B. Required Action and associated Completion Time not met.	B.1	Be in MODE 3.	12 hours
	AND		
	B.2	Be in MODE 4.	36 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.4.5.1	----------NOTE---------- Not required to be performed in MODE 3. ---------- 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 ≥ [ ] and ≤ [ ] psig.	[In accordance with the Inservice Testing Program or [18] months]

JUSTIFICATION FOR DEVIATIONS FROM NUREG-1433, REVISION 1
ISTS: 3.4.5 - RCS PRESSURE ISOLATION VALVE (PIV) LEAKAGE

1. NUREG-1433, Specification 3.4.5, sets forth Limiting Conditions for Operation and Surveillance Requirements for Reactor Coolant System (RCS) pressure isolation valve (PIV) leakage. PIVs are defined as any two valves in series within the reactor coolant pressure boundary (RCPB) which separate the high pressure RCS from an attached low pressure system. These valves are normally closed during power operation.

   The Reactor Safety Study (WASH-1400) identified the potential intersystem loss of coolant accident (Event V) in a PWR as a significant contributor to the risk of core melt. In this scenario, check valves fail in the injection lines of the RHR or low pressure injection systems, allowing high pressure reactor coolant to enter low pressure piping outside containment. Subsequent failure of this low pressure piping would result in loss of reactor coolant outside containment and subsequent core meltdown. Similar scenarios were also determined to be possible in BWRs.

   All plants licensed since 1979 have PIVs listed in their Technical Specifications, along with testing intervals, acceptance criteria, and limiting conditions for operation. Certain older plants were required to periodically leak test, on an individual basis, only those PIVs which were listed in an Order dated April 20, 1981 (Event V Order). That Order was sent to 32 operating PWRs and 2 operating BWRs. Other older plants have had no specific requirements imposed to individually leak test any of their PIVs.

   Quad Cities 1 and 2 were licensed prior to 1979, and was not a recipient of the Event V Order to perform periodic leak tests of PIVs. Therefore, the requirements of NUREG-1433 Specification 3.4.5 do not currently apply to Quad Cities 1 and 2, and are not incorporated in the ITS. Subsequent Specifications are renumbered accordingly.

⟨CTS⟩

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.6 5 RCS Leakage Detection Instrumentation

LCO 3.4.6 5 The following RCS leakage detection instrumentation shall be OPERABLE: 1

⟨3.6.G⟩

a. Drywell floor drain sump monitoring system; ~~and~~ 2

b. One channel of either primary containment atmospheric particulate or atmospheric gaseous monitoring system; [and 3

c. Primary containment air cooler condensate flow rate monitoring system]. 4

⟨Appl 3.6.G⟩

APPLICABILITY: MODES 1, 2, and 3.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Drywell floor drain sump monitoring system inoperable.	----------NOTE---------- LCO 3.0.4 is not applicable. ---------- 5	
	A.1 Restore drywell floor drain sump monitoring system to OPERABLE status.	30 days

⟨3.6.G Act 1⟩

⟨3.6.G Act 2⟩

TSTF-60 not adopted

(continued)

<CTS>

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
<3.6.6 Act 1> B. Required primary containment atmospheric monitoring system inoperable. (particulate) [3]	----------NOTE---------- LCO 3.0.4 is not applicable. ---------- [5] [2]	
	B.1 Analyze grab samples of primary containment atmosphere.	Once per 12 hours
	AND	
	B.2 Restore required primary containment atmospheric (particulate) monitoring system to OPERABLE status.	30 days [2]
C. Primary containment air cooler condensate flow rate monitoring system inoperable.	C.1 ----------NOTE---------- Not applicable when required primary containment atmospheric monitoring system is inoperable. ---------- Perform SR 3.4.6.1.	Once per 8 hours [4]

(continued)

⟨CTS⟩

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
~~D. Required primary containment atmospheric monitoring system inoperable. AND Primary containment air cooler condensate flow rate monitoring system inoperable.~~	~~NOTE LCO 3.0.4 is not applicable.~~ ~~D.1 Restore required primary containment atmospheric monitoring system to OPERABLE status. OR D.2 Restore primary containment air cooler condensate flow rate monitoring system to OPERABLE status.~~	~~30 days 30 days~~
⟨3.6.G Act 2⟩ C. Required Action and associated Completion Time of Condition A, B, [C, or D] not met. or	C. E.1 Be in MODE 3. AND C. E.2 Be in MODE 4.	12 hours 36 hours
⟨DOC A.4⟩ D. F. All required leakage detection systems inoperable.	D. F.1 Enter LCO 3.0.3.	Immediately

⟨CTS⟩
⟨Doc L.3⟩

Insert SR NOTE 6

SURVEILLANCE REQUIREMENTS

	SURVEILLANCE		FREQUENCY
⟨4.6.H.1⟩ 1	SR 3.4.5.1	Verify ~~Perform a CHANNEL CHECK of required~~ primary containment atmospheric monitoring system operating	12 hours 3
⟨Doc M.1⟩ 1	SR 3.4.5.2	Perform a CHANNEL FUNCTIONAL TEST of required leakage detection instrumentation.	31 days
⟨4.6.G.2⟩ 1	SR 3.4.5.3	Perform a CHANNEL CALIBRATION of required leakage detection instrumentation.	[18] 24 months 2

6

INSERT SR NOTE

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

DOC L.3

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.

JUSTIFICATION FOR DEVIATIONS FROM NUREG-1433, REVISION 1
ITS: 3.4.5 - RCS LEAKAGE DETECTION INSTRUMENTATION

1. BWR ISTS, NUREG-1433, Revision 1, Specification 3.4.5, "Reactor Coolant System Pressure Isolation Valve (PIV) Leakage," has not been incorporated in ITS. Subsequent ITS Specifications and Bases have been renumbered accordingly.

2. The brackets have been removed and the proper plant specific information/value has ben provided.

3. Changes have been made to reflect plant specific nomenclature and current licensing basis requirements.

4. The bracketed requirement/information has been deleted since it is not applicable to Quad Cities 1 and 2. The following requirements have been renumbered, where applicable, to reflect this deletion.

5. TSTF-60 moved the Notes in ISTS ACTIONS A and B to the beginning of the ACTIONS Table. This implies the Note is applicable to all the ACTIONS. However, the Note "LCO 3.0.4 in not applicable" is not applicable to proposed ISTS ACTIONS C and D, which require a plant shutdown. Therefore, to avoid confusion, TSTF-60 has not been incorporated.

6. 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 to be used unless the other channel is OPERABLE. This change has been previously approved at Georgia Power Company's Plant Hatch Units 1 and 2, in Amendments 185 and 125, respectively, and in the ITS conversion for Washington Public Power Supply System's WNP-2, Amendment 149.

⟨CTS⟩

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.6 RCS Specific Activity

1

⟨3.6.J⟩ LCO 3.4.6 The specific activity of the reactor coolant shall be limited to DOSE EQUIVALENT I-131 specific activity ≤ {0.2} 2 μCi/gm.

⟨Appl 3.6.J⟩ APPLICABILITY: MODE 1,
MODES 2 and 3 with any main steam line not isolated.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
⟨3.6.J Act 1⟩ A. Reactor coolant specific activity > {0.2} μCi/gm and ≤ 4.0 μCi/gm DOSE EQUIVALENT I-131. 2	----------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
⟨3.6.J Act 2⟩ B. Required Action and associated Completion Time of Condition A not met.	B.1 Determine DOSE EQUIVALENT I-131.	Once per 4 hours
	AND	
OR	B.2.1 Isolate all main steam lines.	12 hours
Reactor Coolant specific activity > {4.0} μCi/gm Dose EQUIVALENT I-131. 2	OR	

(continued)

⟨CTS⟩

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
⟨3.6.J Act 2⟩ B. (continued)	B.2.2.1 Be in MODE 3.	12 hours
	AND	
	B.2.2.2 Be in MODE 4.	36 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
⟨4.6.J⟩ SR 3.4.~~0~~.1 6 1	----------NOTE---------- Only required to be performed in MODE 1. ----------	
	Verify reactor coolant DOSE EQUIVALENT I-131 specific activity is ≤ ~~1~~0.2~~%~~ μCi/gm. 2	7 days

JUSTIFICATION FOR DEVIATIONS FROM NUREG-1433, REVISION 1
ITS: 3.4.6 - RCS SPECIFIC ACTIVITY

1. ISTS 3.4.7 has been renumbered as ITS 3.4.6 as a result of the deletion of ISTS 3.4.5, "Reactor Coolant System Pressure Isolation Valve (PIV) Leakage."

2. The brackets have been removed and the proper plant specific information/value has been provided.

3.4 REACTOR COOLANT SYSTEM (RCS)

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

LCO 3.4.8 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 removed from 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 the performance of Surveillances.

APPLICABILITY: MODE 3, with reactor steam dome pressure < 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(s) to OPERABLE status. AND	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.	1 hour
	AND	
	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.	Immediately
	AND	
	B.2 Verify reactor coolant circulation by an alternate method.	1 hour from discovery of no reactor coolant circulation AND Once per 12 hours thereafter
	AND	
	B.3 Monitor reactor coolant temperature and pressure.	Once per hour

<CTS>

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
<4.6.0> SR 3.4.8.1 1-7 ——NOTE—— Not required to be met until 2 hours after reactor steam dome pressure is ~~{~~ less than the RHR cut in permissive pressure~~}~~. 3 ——— Verify each ~~one~~ RHR shutdown cooling subsystem ~~or recirculation pump is operating~~. manual and power operated valve in the flow path, that is not locked, sealed or otherwise secured in position, is in the correct position or can be aligned to the correct position 2	12 hours

JUSTIFICATION FOR DEVIATIONS FROM NUREG-1433, REVISION 1
ITS: 3.4.7 - RHR SHUTDOWN COOLING SYSTEM — HOT SHUTDOWN

1. ISTS 3.4.8 is renumbered as ITS 3.4.7 as a result of the deletion of ISTS 3.4.5, "Reactor Coolant System Pressure Isolation Valve (PIV) Leakage."

2. The requirement in ISTS 3.4.8 (ITS 3.4.7) to maintain an RHR shutdown cooling subsystem or recirculation pump in operation has been deleted. This deviation from the Standard Technical Specifications was approved by the NRC in the SER for Amendments 162 (Unit 1) and 158 (Unit 2) from John F. Stang (NRC) to D.L. Farrar (ComEd), dated September 21, 1995. As a result, the LCO, and associated Notes, ACTIONS and Surveillances have been revised to reflect current allowances. (Refer to Discussion of Changes for further discussion). Since ISTS 3.4.8 Note 1 has been deleted, the changes approved in TSTF-153 are not shown.

3. 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~~ 8 Residual Heat Removal (RHR) Shutdown Cooling System—Cold Shutdown

1

LCO 3.4.~~9~~ 8

⟨3.6.P⟩

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. 2

⟨3.6.P footnote b⟩

TSTF-153 Changes not shown

⟨Doc L.1⟩

NOTES

1. Both RHR shutdown cooling subsystems and recirculation pumps may be removed from operation for up to 2 hours per 8 hour period. 2

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

Not required to be met during hydrostatic testing. 3

⟨Appl 3.6.P⟩

APPLICABILITY: MODE 4.

ACTIONS

⟨Doc A.4⟩

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

CONDITION	REQUIRED ACTION	COMPLETION TIME
⟨3.6.P Act 1⟩ ⟨3.6.P Act 2⟩ 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)

AND 2

<CTS>

— — — — — Note — — —
Only applicable if both RHR shutdown cooling subsystems are inoperable.
— — — — —

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
<3.6.P Act 1> <3.6.P Act 2> B. ~~No RHR shutdown cooling subsystem in operation.~~ ~~AND~~ ~~No recirculation pump in operation.~~	B.1 A.2 Verify reactor coolant circulating by an alternate method.	~~1 hour from discovery of no reactor coolant circulation~~ ~~AND~~ Once per 12 hours thereafter
	AND	
A.3	B.2 Monitor reactor coolant temperature.	Once per hour

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
<4.6.P> SR 3.4.8.1 Verify ~~one~~ each RHR shutdown cooling subsystem ~~or recirculation pump is operating~~.	12 hours

manual and power operated valve in the flow path that is not locked, sealed or otherwise secured in position, is in the correct position or can be aligned to the correct position

JUSTIFICATION FOR DEVIATIONS FROM NUREG-1433, REVISION 1
ITS: 3.4.8 - RHR SHUTDOWN COOLING SYSTEM — COLD SHUTDOWN

1. ISTS 3.4.9 is renumbered as ITS 3.4.8 as a result of the deletion of ISTS 3.4.5, "Reactor Coolant System Pressure Isolation Valve (PIV) Leakage."

2. The requirement in ISTS 3.4.9 (ITS 3.4.8) to maintain a RHR shutdown cooling subsystem or recirculation pump in operation has been deleted. This deviation from the Standard Technical Specifications was approved by the NRC in the SER for Amendments 162 (Unit 1) and 158 (Unit 2) from John F. Stang (NRC) to D.L. Farrar (ComEd), dated September 21, 1995. As a result, the LCO, LCO Note 1, ACTIONS and Surveillances have been revised to reflect current allowances (refer to Discussion of Changes for further discussion). Since ISTS 3.4.9 Note 1 has been deleted, the changes approved in TSTF-153 are not shown.

3. Note 1 has been added to ISTS 3.4.9 (ITS 3.4.8) in order allow the performance of the hydrostatic test with both RHR shutdown cooling subsystems inoperable. This allowance in ITS 3.4.8 is necessary since CTS 3.12.C (ISTS 3.10.1, "Inservice Leak and Hydrostatic Testing Operation") has been deleted in accordance with the Technical Specifications Change Request submitted to the NRC for approval per ComEd letter SVP-99-193, dated November 12, 1999. Since CTS 3.12.C (ISTS 3.10.1) allowed the suspension of the requirements in CTS 3.6.P (ISTS 3.4.9) to allow the performance of inservice leak or hydrostatic test, this allowance will be needed in ITS 3.4.8. 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.

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.10 RCS Pressure and Temperature (P/T) Limits

1 9

LCO 3.4.10 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. 2

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 MODES 1, 2, and 3. or	A.1 Restore parameter(s) to within limits.	30 minutes
	AND	
	A.2 Determine RCS is acceptable for continued operation.	72 hours
B. Required Action and associated Completion Time of Condition A not met.	B.1 Be in MODE 3.	12 hours
	AND	
	B.2 Be in MODE 4.	36 hours

3

(continued)

<CTS>

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
<3.6.K Act 1> <3.6.K Act 2> <DOC M.1> 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
1 9 SR 3.4.~~10~~.1 <3.6.K.1> <3.6.K.2> <4.6.K.1> <4.6.K.2> <3.6.K.3> ----------NOTE---------- Only required to be performed during RCS heatup and cooldown operations and RCS inservice leak and hydrostatic testing. ---------- Verify: a. RCS pressure, and b. RCS temperature ~~and RCS heatup and cooldown rates~~ are within the applicable limits specified in ~~the PTLR~~ Figures 3.4.9-1, 3.4.9-2, and 3.4.9-3;	30 minutes
9 1 SR 3.4.~~10~~.2 <4.6.K.2> <3.6.K.3> Verify RCS pressure and RCS temperature are within the applicable criticality limits specified in ~~the PTLR~~ Figure 3.4.9-3	Once within 15 minutes prior to control rod withdrawal for the purpose of achieving criticality

b. RCS heatup and cooldown rates are ≤ 100°F in any 1 hour period; and
c. RCS temperature change during inservice leak and hydrostatic testing is ≤ 20°F in any 1 hour period when the RCS pressure and RCS temperature are being maintained within the limits of Figure 3.4.9-1.

(continued)

2

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.4.10.3	NOTE Only required to be met in MODES 1, 2, 3, and 4 with reactor steam dome pressure ≥ 25 psig.	
	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.	Once within 15 minutes prior to each startup of a recirculation pump
SR 3.4.10.4	NOTE Only required to be met in MODES 1, 2, 3, and 4.	
	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.	Once within 15 minutes prior to each startup of a recirculation pump
SR 3.4.10.5	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.	30 minutes

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE	FREQUENCY
SR 3.4.10.6 (9) [1] ----------NOTE---------- Not required to be performed until 30 minutes after RCS temperature ≤ 80°F in MODE 4. (≤ 93) [5] ---------- Verify reactor vessel flange and head flange temperatures are within the limits specified in the PTLR. (≥83°F) [2]	30 minutes
SR 3.4.10.7 (9) [1] ----------NOTE---------- Not required to be performed until 12 hours after RCS temperature ≤ 100°F in MODE 4. (113) [5] ---------- Verify reactor vessel flange and head flange temperatures are within the limits specified in the PTLR. (≥83°F) [2]	12 hours

Insert Figures 3.4.9-1, 3.4.9-2, and 3.4.9-3 [2]

⟨CTS⟩

Insert Figure 3.4.9-1

⟨Fig. 3.6.K-1⟩

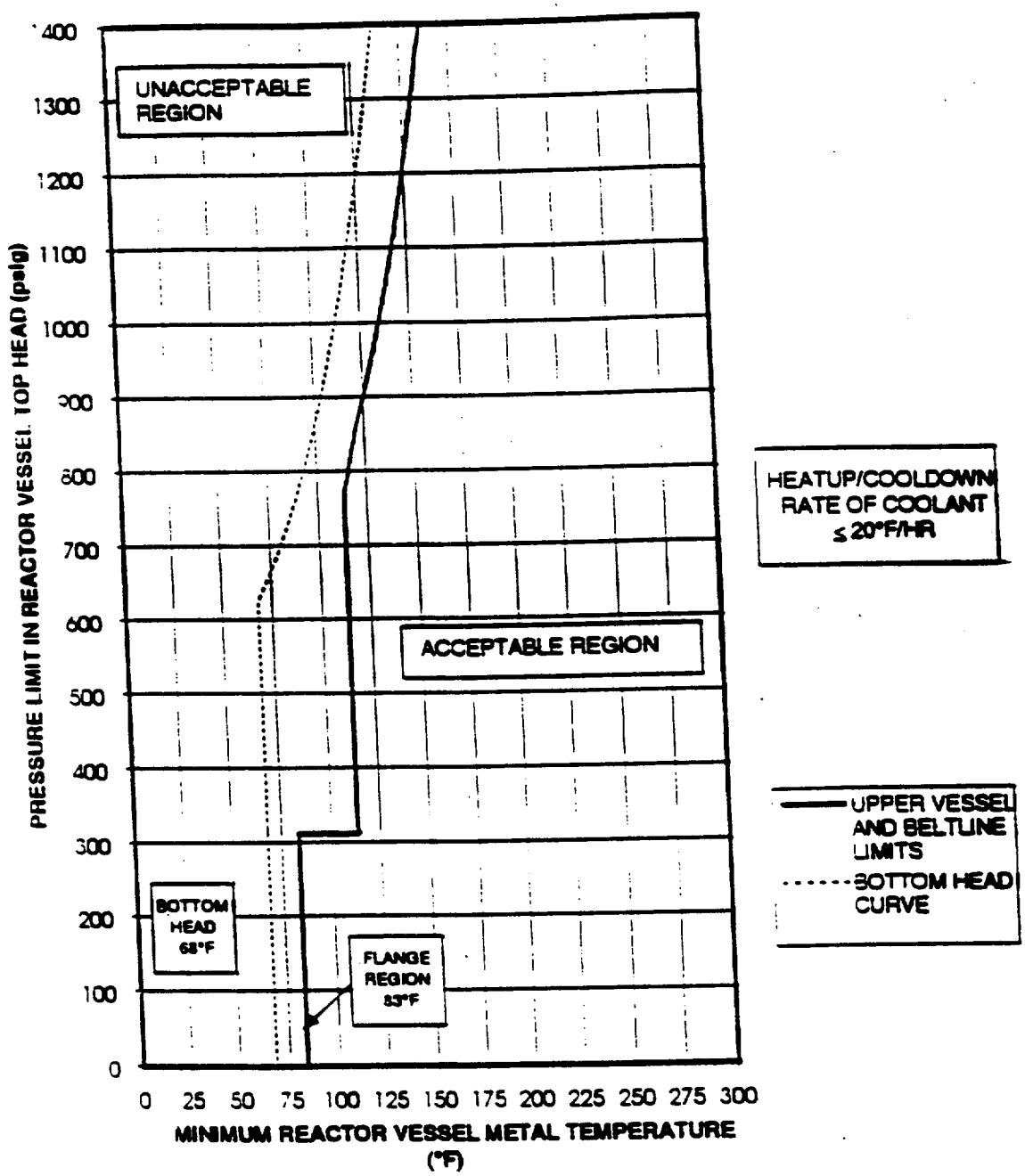


Figure 3.4.9-1 (Page 1 of 1)
Non-Nuclear Inservice Leak and Hydrostatic Testing Curve
(Valid to 32 EFPY)

Insert Page 3.4-26a

⟨CTS⟩

Insert Figure 3.4.9-2

⟨Fig 3.6.K-2⟩

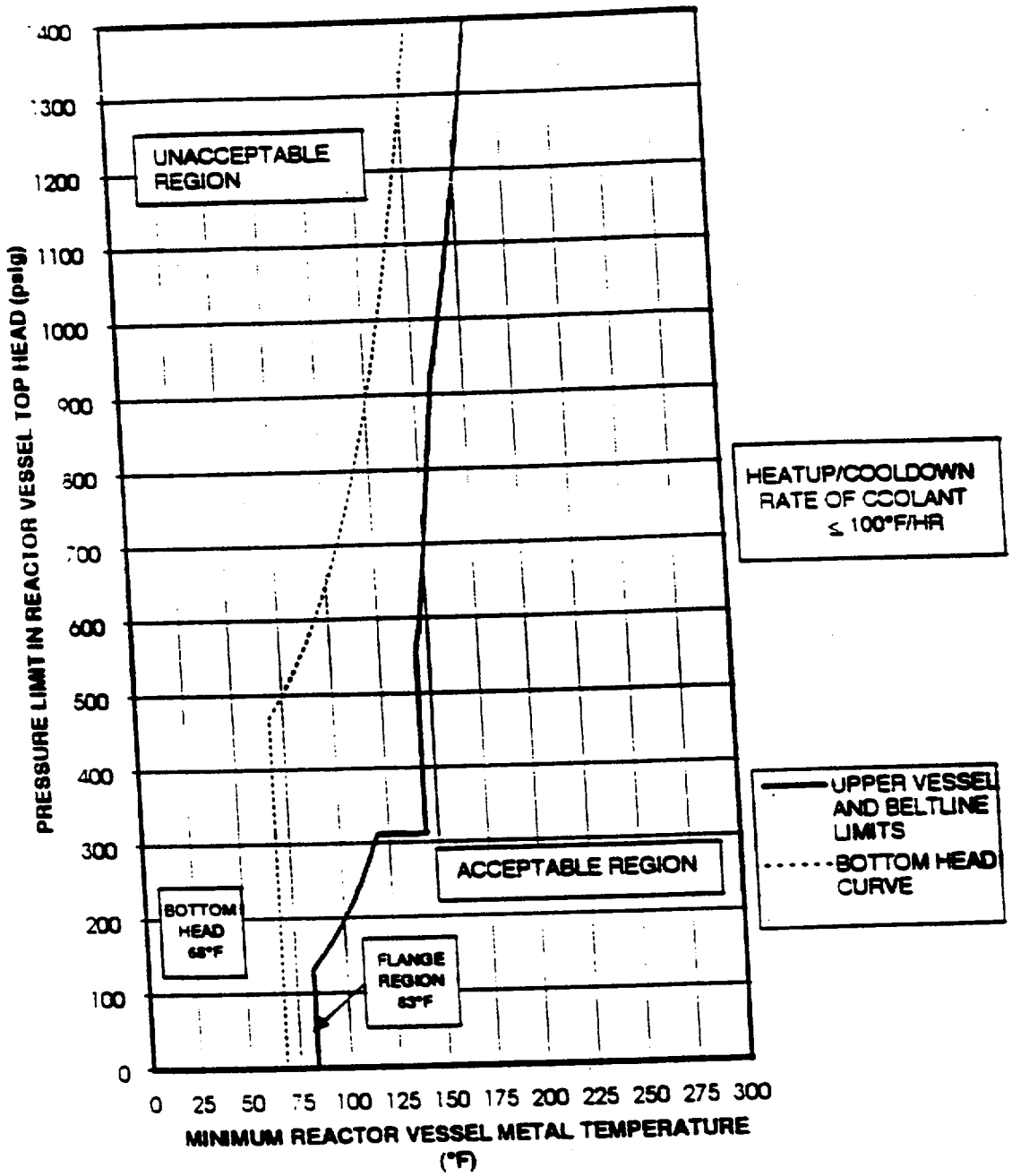


Figure 3.4.9-2 (Page 1 of 1)
Non-Nuclear Heatup/Cooldown Curve
(Valid to 32 EFPY)

<CTS>

Insert Figure 3.4.9-3

<Fig 3.6.K-3>

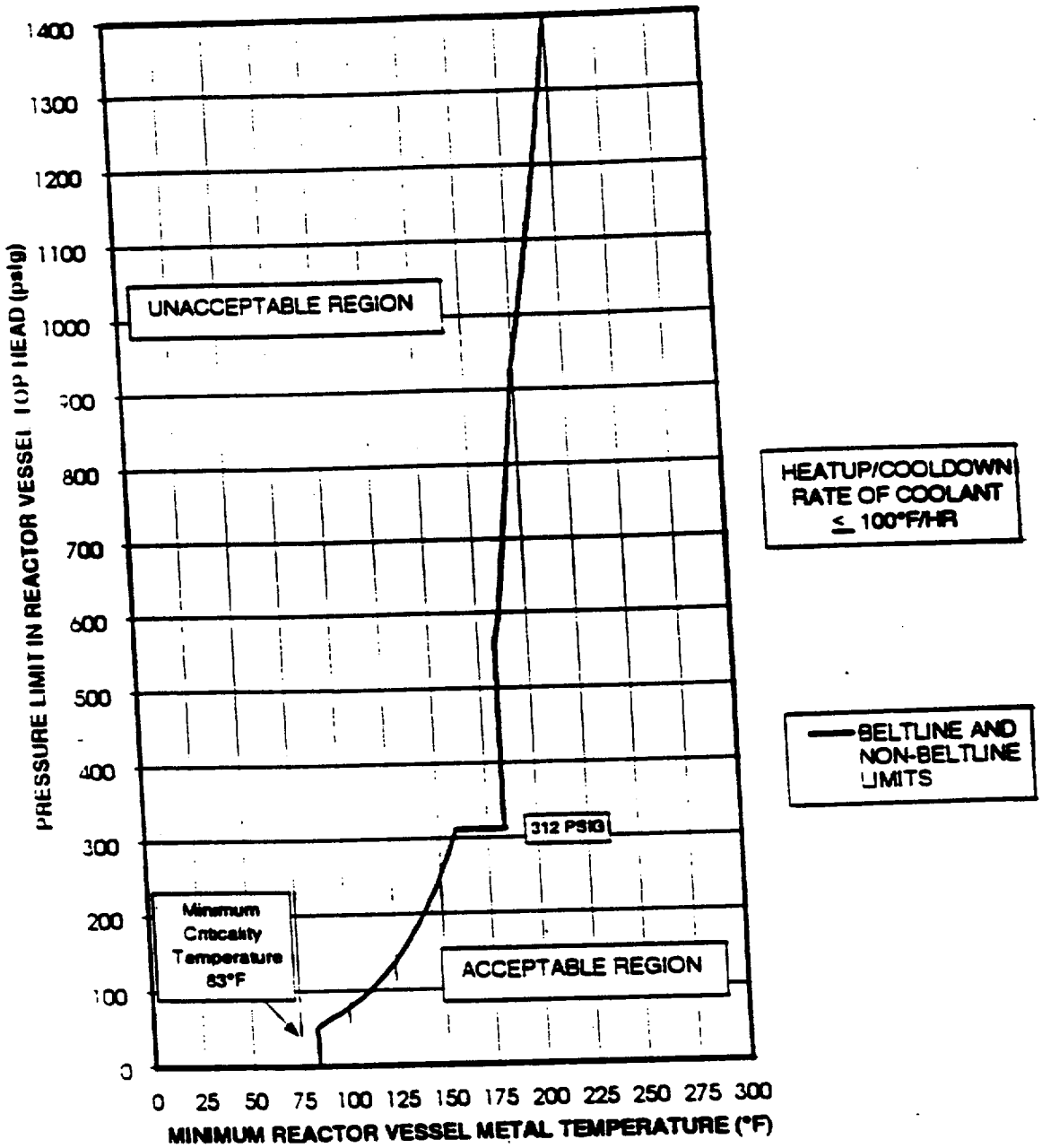


Figure 3.4.9-3 (Page 1 of 1)
Critical Operations Curve
(Valid to 32 EFPY)

Insert Page 3.4-26c

JUSTIFICATION FOR DEVIATIONS FROM NUREG-1433, REVISION 1
ITS: 3.4.9 - RCS PRESSURE AND TEMPERATURE (P/T) LIMITS

1. BWR ISTS, NUREG-1433, Revision 1, Specification 3.4.5, "Reactor Coolant System Pressure Isolation Valve (PIV) Leakage," has not been incorporated in ITS. Subsequent ITS Specifications and Bases have been renumbered accordingly.

2. 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, Quad Cities 1 and 2 do 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 3.4.9).

3. Editorial changes have been made to achieve consistency with the Writer's Guide.

4. The brackets have been removed and the information deleted since it does not apply.

5. The proper Quad Cities 1 and 2 plant specific value has been provided.

6. 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 Quad Cities 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.

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.10 Reactor Steam Dome Pressure

LCO 3.4.10 The reactor steam dome pressure shall be ≤ [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 3.4.10.1 Verify reactor steam dome pressure is ≤ [1020] psig.	12 hours

JUSTIFICATION FOR DEVIATIONS FROM NUREG-1433, REVISION 1
ITS: 3.4.10 - REACTOR STEAM DOME PRESSURE

1. ISTS 3.4.11 is renumbered as ITS 3.4.10 as a result of the deletion of ISTS 3.4.5, "Reactor Coolant System Pressure Isolation Valve (PIV) Leakage."

2. 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

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 one variable speed motor driven recirculation pump, a motor generator (MG) set to control pump speed and 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. 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. The subcooled water enters the bottom of the fuel channels and contacts the fuel cladding, where heat

(continued)

BASES

BACKGROUND (continued)

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 to 100% of RTP) without having to move control rods and disturb desirable flux patterns.

Each recirculation loop is manually started from the control room. The MG set provides regulation of individual recirculation loop drive flows. The flow in each loop is manually 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). 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. 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.

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

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). in 2

The transient analyses of Chapter 15 of the 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 requirements are modified. During single recirculation loop operation, modification to the Reactor Protection System (RPS) average power range monitor (APRM), ~~instrument setpoints~~ is also required to account for the different relationships between recirculation drive flow and reactor core flow. The APLHGR and MCPR ~~setpoints~~ for single loop operation are specified in the COLR. The APRM flow biased ~~simulated THERMAL POWER setpoint~~ is in LCO 3.3.1.1, "Reactor Protection System (RPS) Instrumentation."

Allowable Values 3 | limits | Neutron Flux-High | and the Rod Block Monitor 4 | Allowable Value 3 | 1 4 | The Rod Block Monitor - Upscale Allowable Value is in LCO 3.3.2.1, "Control Rod Block Instrumentation

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 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)"), ~~and~~ APRM Flow Biased ~~Simulated Thermal Power~~-High ~~setpoint~~ (LCO 3.3.1.1) ~~may~~ must be applied to allow continued operation consistent with the assumptions of Reference 3.

Alternatively, with 3 | Neutron Flux | 4 | and the Rod Block Monitor - Upscale Allowable Value (LCO 3.3.2.1) | 3 Allowable Value | must 3

(continued)

BASES (continued)

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.

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

4

Insert Actions

ACTIONS

A.1 B.1 and C.1

4

for reasons other than Condition A or B (e.g. one loop is "not in operation)

With the requirements of the LCO not met, the recirculation loops must be restored to operation with matched flows within 24 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.~~ 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.

for greater than 2 hours (i.e. Required Action B.1 has been taken)

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 the lower flow must be declared "not in operation," as required by Required Action B.1.

4

Alternatively, if the single loop requirements of the LCO are applied to the APLHGR and MCPR operating limits and RPS setpoints 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.

3

The 2 hour and 24 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.

4

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 pump speeds to re-establish forward flow or by tripping the pump.

(continued)

4

INSERT ACTIONS

A.1 and A.2

With no recirculation loops in operation, the probability of thermal-hydraulic oscillations is greatly increased. Therefore, action must be taken as soon as practicable to reduce power to assure stability concerns are addressed and place the unit in at least MODE 2 within 8 hours and 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 transients and minimal dependence on the recirculation loop coastdown characteristics. 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.

BASES

ACTIONS (continued)

~~B.1~~ D 4

With the

~~With no recirculation loops in operation or the~~ Required Action and associated Completion Time of Condition ~~B~~ C 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 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

APLHGRand 4

This SR ensures the recirculation loops 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.

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~~. The is 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 Surveillance 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.

not in operation

(continued)

BASES (continued)

REFERENCES

1. UFSAR, Section 6.3.3.2.2.4.
2. UFSAR, Section 5.5.1.4.
3. ~~Plant specific analysis for single loop operation.~~
4. VFSAR, Section 15.3.6

JUSTIFICATION FOR DEVIATIONS FROM NUREG-1433, REVISION 1
ITS BASES: 3.4.1 - RECIRCULATION LOOPS OPERATING

1. 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.

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.

6. Typographical/grammatical error corrected.

B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.2 Jet Pumps

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 these characteristics affect the Design Basis Accident (DBA) analyses. 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.

Each reactor coolant recirculation loop contains ten 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 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. 1 1

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 2 of the NRC Policy Statement. 3 10CFR 50.36(c)(2)(ii) 2 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 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 Completion Time of 12 hours is 3

(continued)

BASES

ACTIONS

A.1 (continued)

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 3.4.2.1

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). Each recirculation loop must satisfy one of the performance criteria provided. 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 base-lining new "established patterns", engineering judgement of the daily surveillance results is used to detect significant abnormalities which could indicate a jet pump failure.

The recirculation pump speed operating characteristics (pump flow ~~and loop flow~~ versus pump speed) are determined by the flow resistance from the loop suction through the jet pump nozzles. A change in the relationship indicates a plug, flow restriction, loss in pump hydraulic performance, leakage, or new flow path between the recirculation pump discharge and jet pump nozzle. For this criterion, the pump

(continued)

BASES

Flow from a jet pump may be used to simulate the flow in the other jet pump with the same riser. This allowance may be used for one or two jet pumps. This is necessary since one jet pump flow indication instrument line in Unit 1 has failed. An analysis has been performed which demonstrated the acceptability of this method (Ref. 4) [1]

SURVEILLANCE REQUIREMENTS

SR 3.4.2.1 (continued) [5]

flow ~~and loop flow~~ versus pump speed relationship must be verified.

Individual jet pumps in a recirculation loop normally 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 ~~(or 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. ~~This may be indicated by an increase in the relative flow for a jet pump that has experienced beam cracks.~~ [5] [6]

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.

The 24 hour Frequency has been shown by operating experience to be timely for detecting jet pump degradation and is consistent with the Surveillance 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 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. [7] until 24 hours after

Note 2 allows this SR not to be performed ~~when~~ THERMAL POWER ~~is~~ exceeds ≥ 25% of 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. [7]

The 24 hours is an acceptable time to establish conditions appropriate to perform this SR. [3]

(continued)

BASES (continued)

REFERENCES

1. FSAR, Section 6.3.

2. GE Service Information Letter No. 330, June 9, 1980.

3. NUREG/CR-3052, November 1984.

4. EAS 07-0289, Revision 1, "SAFETY EVALUATION TO JUSTIFY OPERATION WITH LOSS OF JET PUMP FLOW INDICATION FOR QUAD CITIES 1 AND 2," March 1989.

JUSTIFICATION FOR DEVIATIONS FROM NUREG-1433, REVISION 1
ITS BASES: 3.4.2 - 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, analysis description, or licensing basis description.

2. Typographical/grammatical error corrected.

3. Editorial change made for enhanced clarity or to be consistent with similar statements in other places in the Bases.

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. The following requirements have been renumbered, where applicable, to reflect the changes.

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.

All changes are [1] unless otherwise indicated

B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.3 Safety/Relief Valves (S/RVs)

Safety and Relief Valves

Each unit is designed with nine safety valves. One of these valves also functions in the relief mode. This valve is a dual function Target Rock safety/relief valve (S/RV). [2]

BASES

BACKGROUND

The ASME Boiler and Pressure Vessel Code requires the reactor pressure vessel be protected from overpressure during upset conditions by self-actuated safety valves. As part of the nuclear pressure relief system, the size and number of 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. 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 spring loaded pilot valve opens when steam pressure at the valve inlet overcomes the spring force holding the pilot valve closed. Opening the pilot valve allows a pressure differential to develop across the main valve piston and opens the main valve. This satisfies the Code requirement.

Each S/RV discharges steam through a discharge line to a point below the minimum water level in the suppression pool. The S/RVs that provide the relief mode are the low-low set (LLS) valves and the Automatic Depressurization System (ADS) valves. The LLS requirements are specified in LCO 3.6.1.6, "Low-Low Set (LLS) Valves," and the ADS requirements are specified in LCO 3.5.1, "ECCS—Operating."

APPLICABLE SAFETY ANALYSES

The overpressure protection system must accommodate the most severe pressurization 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. 1). For the purpose of the analyses, [1] 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

(continued)

2 Insert BKGD-1

the safety valve opens when the inlet steam pressure reaches the lift set pressure. At that point, the vertical upward force generated by the inlet pressure under the valve disc balances the downward force generated by the spring. Slight steam leakage develops across the valve disc-to-seat interface and is directed into the huddle chamber. Pressure builds up rapidly in the huddle chamber developing an additional vertical lifting force on the disc and disc holder. This additional force in conjunction with the expansive characteristic of steam causes the valve to "pop" open to almost full lift.

2 Insert BKGD-2

The S/RV is a dual function Target Rock valve that can actuate by either of two modes: the safety mode or the relief mode. In the safety mode (or spring mode of operation), the S/RV opens in the same manner as described above for the safety valves. In the relief mode (or power actuated mode of operation), automatic or manual switch actuation energizes a solenoid valve which pneumatically actuates a plunger located within the main valve body. Actuation of the plunger allows pressure to be vented from the top of the main valve piston. This allows reactor pressure to lift the main valve piston, which opens the main valve.

2 Insert BKGD-3

In addition to the safety valves and S/RV, each unit is designed with four relief valves which actuate in the relief mode to control RCS pressure during transient conditions to prevent the need for safety valve actuation (except S/RV) following such transients. The relief valves are also located on the main steam lines between the reactor vessel and the first isolation valve within the drywell. These valves are sized by assuming a turbine trip, a coincident scram and a failure of the turbine bypass system. For Unit 1, four of the relief valves are of the Electromatic type, which are opened by automatic or manual switch actuation of a solenoid. The switch energizes the solenoid to actuate a plunger, which contacts the pilot valve operating lever, thereby opening the pilot valve. When the pilot valve opens, pressure under the main valve disc is vented. This allows reactor pressure to overcome main valve spring pressure, which forces the main valve disc downward to open the main valve. For Unit 2, four of the relief valves are of the Target Rock power operated relief valve type. When the solenoid is energized, a magnetic force is developed which moves a plunger upward until it contacts the moveable core. This motion is transmitted through the pilot rod to fully open two pilot discs, allowing the control pressure above the main disc to vent through the second pilot seat to the downstream side of the valve. In addition, the motion of the pilot disc partially reduces the control pressure above the main disc. When the force of the control pressure acting on the top of the main disc falls below the force of the inlet pressure acting on the lower annular area, the main disc will move to the open position. In the open position,

2

Insert BKGD-3 (continued)

with the moveable core positioned close to the fixed core, the magnetic force is well in excess of the closing forces due to control pressure and return spring force. This ensures that the main disc will be held firmly in the open position. The main disc can be opened even with the valve inlet pressure equal to 0 psig.

BASES

APPLICABLE SAFETY ANALYSES (continued)

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.

Insert ASA

From an overpressure standpoint, the design basis events are bounded by the MSIV closure with flux scram event described above. Reference 2 discusses additional events that are expected to actuate the S/RVs.

safety and relief valves

Safety and Relief Valves

S/RVs satisfy Criterion 3 of the NRC Policy Statement.

10 CFR 50.36(c)(2)(ii)

LCO

The safety function of {[1]} S/RVs are required to be OPERABLE to satisfy the assumptions of the safety analysis (Refs. 1 and 2). 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 function).

safety valves

safety valve

The S/RV setpoints are established to ensure that the ASME Code limit on peak reactor pressure is satisfied. The ASME Code specifications require the lowest safety valve setpoint to be at or below vessel design pressure (1250 psig) and the highest safety valve to be set so that the total accumulated pressure does not exceed 110% of the design pressure for overpressurization conditions. The transient evaluations in the FSAR are based on these setpoints, but also include the additional uncertainties of ± 1% of the nominal setpoint drift to provide an added degree of conservatism.

Insert LCO

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.

safety and relief valves

APPLICABILITY

In MODES 1, 2, and 3, all S/RVs must be OPERABLE, since considerable energy may be in the reactor core and the limiting design basis transients are assumed to occur in these MODES. 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 core heat.

(continued)

2 Insert ASA

For other pressurization events, such as a turbine trip or generator load rejection with Main Turbine Bypass System failure (Refs. 2 and 3, respectively), the relief valves as well as the S/RV are assumed to function. [The opening of the relief valves during the pressurization event mitigates the increase in reactor vessel pressure, which affects the MINIMUM CRITICAL POWER RATIO (MCPR) during these events.] In these events, the operation of four of the five relief valves are required to mitigate the events.

2 Insert LCO

The relief valves, including the S/RV, 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.

BASES

s are

APPLICABILITY (continued)

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.

and MCPR

are

Safety and relief

ACTIONS

A.1

relief

relief valve (or S/RV)

With the safety function of one (or two) (required) S/RV(s) 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.

relief valves

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.

relief valve

relief valves

B.1 and B.2

safety valves

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 safety function of the inoperable required S/RVs cannot be restored to OPERABLE status within the associated Completion Time of Required Action A.1, or if the safety function of three or more (required) S/RVs is 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 and to MODE 4 within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

relief valves

relief

one

safety valves

the relief function of two or more relief valves are inoperable, or if

(continued)

BASES (continued)

SURVEILLANCE REQUIREMENTS

SR 3.4.3.1

This Surveillance requires that the [required] S/RVs will open at the pressures assumed in the safety analysis of Reference 1. The demonstration of the S/RV safe lift settings must be performed during shutdown, since this is a bench test, to be done 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 setpoints ± 1% for OPERABILITY; however, the valves are reset to ± 1% during the Surveillance to allow for drift.

The 18 month Frequency was selected because this Surveillance must be performed during shutdown conditions and is based on the time between refuelings.

safety valves, including the S/RV,
safety valve and S/RV
safety valve and SIRV
TSTF-298 changes not adopted

SR 3.4.3.2

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 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 S/RV diverts 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 [920] psig (the pressure recommended by the valve manufacturer). Adequate steam flow is represented by at least [2] turbine bypass valves open or total steam flow ≥ 10% in/hr. Plant startup is allowed prior to performing this test because valve OPERABILITY and the setpoints for overpressure protection are verified, per ASME Code 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

relief valve, including the
relief valve or the
300
Unit
(Ref. 5)

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.4.3.2 (continued)

conditions for testing the S/RV and provides a reasonable time to complete the SR. If fails to actuate due only to the failure of the solenoid but is capable of opening on overpressure, the safety function of the S/RV is considered OPERABLE.

The 24 month on a STAGGERED TEST BASIS Frequency ensures that each solenoid for each S/RV is tested. The 24 month Frequency was developed based on the relief valve tests required by the ASME Boiler and Pressure Vessel Code, Section XI (Ref. 5). Operating experience has shown that these components usually pass the Surveillance when performed at the 24 month Frequency. Therefore, the Frequency was concluded to be acceptable from a reliability standpoint.

REFERENCES

1. UFSAR, Section 5.2.2.2.
2. UFSAR, Chapter 15.
3. ASME, Boiler and Pressure Vessel Code, Section XI.

2. UFSAR, Section 15.2.3.1
3. UFSAR, Section 15.2.2.1

Insert SR 3.4.3.3

I Insert SR 3.4.3.3

SR 3.4.3.3

The relief valves, including the S/RV, are required to actuate automatically upon receipt of specific initiation signals. A system functional test is performed to verify that the mechanical portions (i.e., solenoids) of the relief valve operate as designed when initiated either by an actual or simulated automatic initiation signal. The LOGIC SYSTEM FUNCTIONAL TESTs in LCO 3.3.5.1, "Emergency Core Cooling System (ECCS) Instrumentation," and LCO 3.3.6.3, "Relief Valve Instrumentation," overlap this SR to provide complete testing of the 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 these components usually pass the Surveillance when performed at the 24 month Frequency. 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 3.4.3.2.

JUSTIFICATION FOR DEVIATIONS FROM NUREG-1433, REVISION 1
ITS BASES: 3.4.3 - SAFETY AND RELIEF VALVES

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, analysis description, or licensing basis description.

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 3.4.4 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). UFSAR, Sections 3.1.2.4 and 3.1.3.6

The safety significance of RCS LEAKAGE from the RCPB varies widely depending on the source, rate, and duration. Therefore, detection of LEAKAGE in the primary containment 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 that is detrimental to the safety of the facility or the public.

A limited amount of leakage inside primary containment is expected from auxiliary systems that cannot be made 100% leaktight. Leakage from these systems should be detected and isolated from the primary containment 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 that, 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. 2 and 3) shows that leakage rates of hundreds of gallons per minute will precede crack instability (Ref. 4). 1

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.

RCS operational LEAKAGE satisfies Criterion 2 of the NRC Policy Statement. 10 CFR 50.36 (c)(2)(ii) 1

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

The 5 gpm of unidentified LEAKAGE is allowed as a reasonable minimum detectable amount that the containment air monitoring, drywell sump level monitoring, and containment 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 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

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 service sensitive 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 piping is very susceptible to IGSCC.

identify the source of the unidentified leakage increase is not material

The 4 hour Completion Time is reasonable to properly reduce the LEAKAGE increase or verify the source before the reactor must be shut down without unduly jeopardizing plant safety.

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 safety systems.

SURVEILLANCE REQUIREMENTS

SR 3.4.4.1

The drywell floor drain sump flow integrator.

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 3.4.8, "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 5. In conjunction with alarms and other administrative controls, a 12 hour Frequency for this Surveillance is appropriate for identifying LEAKAGE and for tracking required trends (Ref. 4).

REFERENCES

1. 10 CFR 50, Appendix A, GDC 30.
2. GEAP-5620, April 1968.
3. NUREG-75/067, October 1975.
4. FSAR, Section 5.2.7.5.2.
5. Regulatory Guide 1.45.
6. Generic Letter 88-01, Supplement 1, February 1992.

UFSAR, Sections 3.1.2.4 and 3.1.3.6

"Failure Behavior in ASTM A106B Pipes Containing Axial Through-Wall Flaws,"

"Investigation and Evaluation of Cracking in Austenitic Stainless Steel Piping of Boiling Water Reactor Plants"

An alternate method which may be used to quantify LEAKAGE is calculating flow rates using sump pump run times.

JUSTIFICATION FOR DEVIATIONS FROM NUREG-1433, REVISION 1
ITS BASES: 3.4.4 - RCS OPERATIONAL LEAKAGE

1. 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.

2. The brackets have been removed and the proper plant specific information/value has been provided.

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. Typographical/grammatical error corrected.

B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.5 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). RCS PIVs are defined as any two normally closed valves in series within the reactor coolant pressure boundary (RCPB). 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 3.4.4, "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 event that 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 typically connected systems:

a. Residual Heat Removal (RHR) System;

b. Core Spray System;

(continued)

BASES

BACKGROUND (continued)

c. High Pressure Coolant Injection System; and

d. Reactor Core Isolation Cooling System.

The PIVs are listed in Reference 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 RCPB to detect PIV degradation that has the potential to cause a LOCA outside of containment. RCS PIV leakage satisfies Criterion 2 of the NRC Policy Statement.

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)

1

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 shutdown cooling flow path 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 manual, deactivated automatic, or check valve within 4 hours.

(continued)

BASES

ACTIONS

A.1 and A.2 (continued)

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 considers the time required to complete the 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 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 3.4.5.1

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 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 in series, the leakage

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.4.5.1 (continued)

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.

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 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 not required to be performed in MODE 3. 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, May 1980.
6. FSAR, Section [ ].
7. NEDC-31339, November 1986.

JUSTIFICATION FOR DEVIATIONS FROM NUREG-1433, REVISION 1
ISTS BASES: 3.4.5 - RCS PRESSURE ISOLATION VALVE (PIV) LEAKAGE

1. This Bases has been deleted since the associated Specification has been deleted.

1

B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4 6 5 RCS Leakage Detection Instrumentation

BASES

2

UFSAR, Sections 3.1.3.6 and 3.1.6.4

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 leakage rates. The Bases for LCO 3.4.4, "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 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.

2

in flow out of the drywell floor drain sump

provides a

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 primary containment floor drain sump has transmitters that supply level indications in the main control room.

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.

2

Leakage into the drywell floor drain sump is pumped through a piping header that penetrates the containment wall to the Liquid Radioactive Waste System

(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.

Insert BCKGD 1

a flow input to a

Insert BCKGD 2

A flow indicator (monitor) in the discharge line of the drywell floor drain sump pumps provides (integrator) flow indication in the control room. The pumps can also be started from the control room.

The primary containment air (atmospheric) monitoring systems continuously monitor the primary containment 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 primary containment 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).

satisfies the Regulatory Guide 1.45 (Ref. 2) recommended sensitivity of 1.0 E-9 μCi/cc radioactivity for air borne particulates

Condensate from four of the six primary containment coolers is routed to the primary containment floor drain sump and is monitored by a flow transmitter that provides indication and alarms in the control room. This primary containment 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.

The drywell floor drain sump monitoring system

A control room alarm provided by the primary containment atmospheric particulate radioactivity monitoring system, 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). Therefore, these actions provide adequate response before a significant break in the RCPB can occur.

(continued)

The primary containment atmospheric particulate radioactivity monitoring system provides indication of changes in leakage rates.

2 INSERT BKGD 1

Two drywell floor drain sump pumps take suction from the drywell floor drain sump and discharge to the Liquid Radioactive Waste System. When a high level is reached in the floor drain sump, a level switch actuates to start a floor drain sump pump when the pump discharge valves are open.

2 INSERT BKGD 2

The flow integrator is used to quantify the amount of sump input.

BASES

APPLICABLE SAFETY ANALYSES (continued)

RCS leakage detection instrumentation satisfies Criterion 1 of the NRC Policy Statement.

LCO

The drywell floor drain sump 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 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 3.4.4. This Applicability is consistent with that for LCO 3.4.4.

ACTIONS

A.1

With the drywell floor drain sump monitoring system inoperable, no other form of sampling can provide the equivalent information to quantify leakage. However, the primary containment atmospheric activity monitor [and the primary containment air cooler condensate flow rate monitor] will provide indication of changes in leakage.

With the drywell floor drain sump monitoring system inoperable, but with RCS unidentified and total LEAKAGE being determined every # hours (SR 3.4.4.1), 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 monitoring system is inoperable. This allowance is provided because other instrumentation is available to monitor RCS leakage.

(continued)

BASES

ACTIONS (continued)

B.1 and B.2

the system

With both gaseous and particulate primary containment atmospheric monitoring channels inoperable, grab samples of the primary containment atmosphere must be taken and analyzed to provide periodic leakage information. *Provided a sample is obtained and analyzed once every 12 hours, the plant may be operated for up to 30 days to allow restoration of at least one of the required monitors.* [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 5 3

The 12 hour interval provides periodic information that is adequate to detect LEAKAGE. The 30 day Completion Time for restoration recognizes that at least one other form of leakage detection is available.

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. 2

is methods are

C.1 1 3

With the required primary containment air cooler condensate flow rate monitoring system inoperable, SR 3.4.6.1 must be performed every 8 hours to provide periodic information of activity in the primary containment 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 primary containment atmospheric monitoring system is inoperable. Consistent with SR 3.0.1, Surveillances are not required to be performed on inoperable equipment.

(continued)

BASES

ACTIONS (continued)

1

D.1 and D.2

3

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

C E.1 and E.2

and 4

or 1

If any Required Action of Condition A, B, [E, 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 MODE 4 within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to perform the actions in an orderly manner and without challenging plant systems.

1

D 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.

(continued)

BASES (continued)

Insert SR

SURVEILLANCE REQUIREMENTS

SR 3.4.6.1

requires verification that

This SR is for the performance of a CHANNEL CHECK of the required primary containment 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.

ensures

being continuously monitored for particulate radioactivity

inoperation

primary containment atmosphere

SR 3.4.6.2

This SR is for 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.

SR 3.4.6.3

This SR is for the performance of a CHANNEL CALIBRATION of required leakage detection instrumentation channels. The calibration verifies the accuracy of the instrument string, including the instruments located inside containment. The Frequency of [18] months is a typical refueling cycle and considers channel reliability. Operating experience has proven this Frequency is acceptable.

24

REFERENCES

1. 10 CFR 50, Appendix A, GDC 30.
UFSAR, Sections 3.1.3.6 and 3.1.6.4
2. Regulatory Guide 1.45, May 1973.
3. FSAR, Section [5.2.7.2.1].
5.2.5.7
4. GEAP-5620, April 1968.
"Failure Behavior in ASTM A106B Pipes Containing Axial Through-Wall Flaws"
5. NUREG-75/067, October 1975.
"Investigation and Evaluation of Cracking in Austenitic Stainless Steel Piping of Boiling Water Reactor Plants,"
6. FSAR, Section [5.2.7.5.2].
5.2.5.5

I 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 (either the drywell floor drain sump monitoring system or the primary containment atmospheric particulate 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.

JUSTIFICATION FOR DEVIATIONS FROM NUREG-1433, REVISION 1
ITS BASES: 3.4.5 - RCS LEAKAGE DETECTION INSTRUMENTATION

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, analysis description, or licensing basis description.

3. The bracketed requirement/information has been deleted because it is not applicable to Quad Cities 1 and 2. The following requirements have been renumbered, where applicable, to reflect the changes.

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.

B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.7 RCS Specific Activity

BASES

BACKGROUND

During circulation, the reactor coolant acquires radioactive materials due to release of fission products from fuel leaks into the reactor coolant and activation of corrosion products in the reactor coolant. These radioactive materials in the reactor 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 that 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 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.

This MSLB release forms the basis for determining offsite and control room 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

The limits on the specific activity of the primary coolant also ensure the thyroid dose to control room operators, resulting from an MSLB outside containment during steady state operation will not exceed the limits of GDC 19 of 10CFR 50, Appendix A (Ref. 3)

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 values from a parametric evaluation of typical site locations. These limits are 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 the NRC Policy Statement.

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

(continued)

BASES

ACTIONS

A.1 and A.2 (continued)

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.

A Note to the Required Actions 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 ≤ 0.2 μCi/gm within 48 hours, or if at any time it is > 4.0 μCi/gm, it must be determined at least once 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 and GDC 19 of 10CFR 50, Appendix A (Ref. 3) during a postulated MSLB accident.

Alternatively, the plant can be placed in MODE 3 within 12 hours and in 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 placing the unit in MODES 3 and 4 are reasonable, based on operating

(continued)

BASES

ACTIONS

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

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

SURVEILLANCE REQUIREMENTS

SR 3.4.0.1 6 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. 5

2. FSAR, Section [15.1.10]. 6.4 4

4 2

3. 10 CFR 50, Appendix A, GDC 19. 2

JUSTIFICATION FOR DEVIATIONS FROM NUREG-1433, REVISION 1
ITS BASES: 3.4.6 - RCS SPECIFIC ACTIVITY

1. Changes have been made to reflect changes made to the Specification.

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. 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. 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 3.4.8 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.

The two redundant, manually controlled shutdown cooling subsystems of the RHR System provide decay heat removal. Each loop consists of two motor driven pumps, 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 RHR Service Water System (LCO 3.7.1, "Residual Heat Removal Service Water (RHRSW) System").

APPLICABLE SAFETY ANALYSES

Decay heat removal by operation of 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 subsystem 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.

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. The two subsystems have a common suction source and are allowed to have a common heat exchanger and

(continued)

BASES

LCO (continued)

common discharge piping. Thus, to meet the LCO, both pumps in one loop or one pump in each of the two loops must be OPERABLE. Since the piping and heat exchangers are passive components that are assumed not to fail, they are allowed to be common to both subsystems. 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. However, 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 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 the 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 steam dome pressure below the RHR cut in permissive pressure (i.e., the actual pressure at which the interlock resets) the RHR 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.

In MODES 1 and 2, and in MODE 3 with reactor steam dome 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

(continued)

BASES

APPLICABILITY (continued)

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 MODES 4 and 5 are discussed in LCO 3.4.8, "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(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.

A.3, A.4,

A.1, A.2, and A.

With one ~~required~~ RHR shutdown cooling subsystem the inoperable for decay heat removal, except as permitted by LCO Note 2, the inoperable subsystem must be restored to OPERABLE status

(continued)

BASES

ACTIONS

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

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.

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 Spent Fuel Pool Cooling System and the Reactor Water Cleanup System.

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 permitted by LCO Note 1, reactor coolant circulation by the RHR shutdown cooling subsystem or 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.

(continued)

I

Insert ACTIONS

Required Actions A.3 and A.4 are modified by Notes that clarify that these Required Actions are only applicable when both RHR shutdown cooling subsystems are inoperable since Condition A is applicable when one or two RHR shutdown cooling subsystems are inoperable.

BASES

ACTIONS

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

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.

move to page B 3.4-40 as indicated

During the period when the reactor coolant is being circulated by an alternate method (other than by ~~the~~ an ~~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.

2 (Required Action A4)

SURVEILLANCE REQUIREMENTS

SR 3.4.~~6~~.1 7 1

Insert SR 3.4.7.1

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.

1 verification that valves are aligned or can be aligned

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.

I

Insert SR 3.4.7.1

Verifying the correct alignment for manual and power operated valves in the two RHR shutdown cooling subsystems' flow paths provides assurance that the proper flow paths will exist for RHR 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 can be manually (from the control room or locally) aligned is allowed to be in a non-RHR shutdown cooling position provided the valve can be repositioned. 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.

JUSTIFICATION FOR DEVIATIONS FROM NUREG-1433, REVISION 1
ITS BASES: 3.4.7 - RHR SHUTDOWN COOLING SYSTEM — HOT SHUTDOWN

1. Changes have been made to reflect those changes made to the Specification.

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 (additions, deletions, and/or changes to the NUREG) to reflect the plant specific nomenclature, number, reference, system description, analysis description, or licensing basis description.

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. The brackets have been removed and the proper plant specific information/value has been provided.

6. Changes have been made to more closely match the LCO requirements.

7. The proper LCO number has been included.

B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.8 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 ≤ 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.

The two redundant, manually controlled shutdown cooling subsystems of the RHR System provide decay heat removal. Each loop consists of two motor driven pumps, 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 RHR Service Water System.

APPLICABLE SAFETY ANALYSES

Decay heat removal by operation of 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.

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, and the associated piping and valves. The two subsystems have a common suction source and are allowed to have a common heat exchanger and common discharge piping. Thus, to meet the LCO, both pumps

(continued)

BASES

LCO (continued)

in one loop (and two RHR service water pumps) or one RHR pump in each of the two loops must be OPERABLE. Since the piping and heat exchangers are passive components that are assumed not to fail, they are allowed to be common to both subsystems. In MODE 4, the RHR cross tie valve (2E11-F010) may be opened to allow pumps in one loop to discharge through the opposite recirculation loop to make a complete subsystem. Additionally, 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 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 or reduce the reactor coolant temperature as required. However, to ensure adequate core flow to allow for accurate average reactor coolant temperature monitoring, nearly continuous operation is required.

(and one RHR service water pump)

TSTF-153 changes not shown

Note 1 permits both RHR shutdown cooling subsystems 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 the 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.

shutdown cooling subsystem

Insert Note 1

shutdown cooling

APPLICABILITY

In MODE 4, the RHR Shutdown Cooling System may be operated in the shutdown cooling mode to remove decay heat to maintain coolant temperature below 200°F. Otherwise, recirculation pump is required to be in operation.

must be OPERABLE to ensure it can

In MODES 1 and 2, and in MODE 3 with reactor steam dome 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

(continued)

1

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)

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 3.4.8, "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 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.

A.1

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

(continued)

BASES

ACTIONS

A.1 (continued)

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 Spent Fuel Pool Cooling System and the Reactor Water Cleanup System.

Condensate/Feed and Main Steam Systems

in the decay heat removal mode (by itself or using feed and bleed in combination with Control Rod Drive System or Condensate/Feed System), and a combination of an ECCS pump and a safety/relief valve.

A.2 B.1 and A.3 B.2

With both RHR shutdown cooling subsystem inoperable, except as permitted by Note 1

With no RHR shutdown cooling subsystem and no recirculation pump in operation, except as 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 importance of 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.

This alternate method may be satisfied by placing a recirculation pump in operation.

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.

subsystems

one of the

Insert ACTIONS

(continued)

I

Insert ACTIONS

Required Actions A.2 and A.3 are modified by Notes that clarify that these Required Actions are only applicable when both RHR shutdown cooling subsystems are inoperable since Condition A is applicable when one or two RHR shutdown cooling subsystems are inoperable.

BASES (continued)

SURVEILLANCE REQUIREMENTS

SR 3.4.8.1 8 1

Insert SR 3.4.8.1 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.

I

Insert SR 3.4.8.1

Verifying the correct alignment for manual and power operated valves in the two RHR shutdown cooling subsystems' flow paths provides assurance that the proper flow paths will exist for RHR 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 can be manually (remote or local) aligned is allowed to be in a non-RHR shutdown cooling position provided the valve can be repositioned. 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.

JUSTIFICATION FOR DEVIATIONS FROM NUREG-1433, REVISION 1
ITS BASES: 3.4.8 - RHR SHUTDOWN COOLING SYSTEM — COLD SHUTDOWN

1. Changes have been made to reflect those changes made to the Specification.

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 (additions, deletions, and/or changes to the NUREG) to reflect the plant specific nomenclature, number, reference, system description, analysis description, or licensing basis description.

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.

6. The correct LCO number has been included.

1

B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.~~10~~ 9 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.

1

Specification

2

The ~~PTLR~~ contains P/T limit curves for heatup, criticality, cooldown, and inservice leak~~age~~ 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.~~ and also limits

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 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 Appendix H of 10 CFR 50 (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.

2 Insert BKGD-1

~~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.~~

2 P/T

2 non-critical

The criticality limits include the Reference 1 requirement that they be at least 40°F above the heatup curve or the cooldown curve and not lower than the minimum permissible temperature for the inservice leakage and hydrostatic testing. Insert BKGD-2 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. 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 1

(continued)

2

INSERT BKGD-1

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

INSERT BKGD-2

Reference 1 also allows boiling water reactors to operate with the core critical below the minimum permissible temperature allowed for the inservice hydrostatic pressure test (i.e., inservice leak and hydrostatic testing) when the water level is within the normal range for power operation and the pressure is less than 20% of the preservice system hydrostatic test pressure (for Quad Cities 1 and 2, this pressure is 312 psig). Under these conditions, the minimum temperature is 60°F above the RT_{NDT} of the closure flange regions which are stressed by the bolt preload (for Quad Cities 1 and 2, this temperature is 83°F).

BASES

APPLICABLE SAFETY ANALYSES (continued)

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 the NRC Policy Statement. 10CFR 50.36 (c)(2)(ii) 2

LCO

The elements of this LCO are: 1

a. RCS pressure, and temperature, and heatup or cooldown rates are within the limits specified in the PTLR, during RCS heatup, cooldown, and inservice leak and hydrostatic testing; (Figures 3.4.9-1, 3.4.9-2, and 3.4.9-3) (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 during recirculation pump startup, and during increases in THERMAL POWER or loop flow while operating at low THERMAL POWER or loop flow; (≤145°F) 2

c. The temperature difference between the reactor coolant in the respective recirculation loop and in the reactor vessel meets the limit of the PTLR during recirculation pump startup, and during increases in THERMAL POWER or loop flow while operating at low THERMAL POWER or loop flow; (15 ≤ 50°F) 2

d. RCS pressure and temperature are within the criticality limits specified in the PTLR, prior to achieving criticality; and (Figure 3.4.9-3)

e. The reactor vessel flange and the head flange temperatures are within the limits of the PTLR when tensioning the reactor vessel head bolting studs, (≥83°F)

and the RCS temperature change during inservice leak and hydrostatic testing is ≤ 20°F in any 1 hour period when the RCS temperature and pressure are being maintained within the limits of Figure 3.4.9-1.

in MODES 1, 2, 3, and 4 5

and 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.

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 leakage and hydrostatic testing P/T limit curves. Thus, the LCO for the 2

(continued)

BASES

LCO (continued)

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 MODES 1, 2, and 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 evaluation is required to determine if RCS operation can continue. The

(continued)

BASES

ACTIONS

A.1 and A.2 (continued)

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. 4

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 placed in 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 placing the plant in at least MODE 3 within 12 hours and in 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.

2 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 212 2 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 Code, 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 4

SURVEILLANCE REQUIREMENTS

SR 3.4.10.1 9 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 a reasonable time for assessment and correction of minor deviations. 1

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.

This SR has been modified with a Note that requires this Surveillance to be performed only during system heatup and cooldown operations and inservice leakage and hydrostatic testing. 2

(continued)

4 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 3.4.10.2

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.

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 3.4.10.3 and SR 3.4.10.4

Differential temperatures within the applicable P/T LR 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.

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 3.4.9.3 is to compare the temperatures on the bottom head drain line and the saturation temperature corresponding to reactor steam dome pressure.

An acceptable means of demonstrating compliance with the temperature differential requirement in SR 3.4.10.4 is to compare the temperatures of the operating recirculation loop and the idle loop.

SR 3.4.10.3 has been modified by a Note that requires the Surveillance to be performed 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.

and SR 3.4.9.4 have

The Note also states the SRs are only required to be met during a recirculation pump startup, since this is when stresses occur.

SR 3.4.10.5, SR 3.4.10.6, and SR 3.4.10.7

Limits on the reactor vessel flange and head flange temperatures are generally bounded by the other P/T limits

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.4.10.5, SR 3.4.10.6, and SR 3.4.10.7 (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.

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°F, 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°F, monitoring of the flange temperature is required every 12 hours to ensure the temperature is within the limits specified in the PTLR.

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.

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.
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. FSAR, Section [15.1.26]. 15.4.4.3

4

INSERT SR 3.4.9.5, SR 3.4.9.6, AND SR 3.4.9.7

SR 3.4.9.5 is modified by a Note that requires the Surveillance to be performed only when tensioning the reactor vessel head bolting studs. SR 3.4.9.6 is modified by a Note that requires the Surveillance to be initiated 30 minutes after RCS temperature $\leq$ 93°F in MODE 4. SR 3.4.9.7 is modified by a Note that requires the Surveillance to be initiated 12 hours after RCS temperature $\leq$ 113°F 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-1433, REVISION 1
ITS BASES: 3.4.9 - 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, analysis description, or licensing basis 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 Specification.

6. The brackets have been removed and the proper plant specific information/value has been provided.

B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.10 0 1 Reactor Steam Dome Pressure

BASES

BACKGROUND

The reactor steam dome pressure is an assumed initial condition of design basis accidents and transients and is also an assumed value in the determination of compliance with reactor pressure vessel overpressure protection criteria and is also. 2

APPLICABLE SAFETY ANALYSES

The reactor steam dome pressure of ≤ [1020] 1005 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 analyses of design basis accidents and transients used to determine the limits for fuel cladding integrity (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)"). 3 5 e 4 5

LCO 3.2.3, "LINEAR HEAT GENERATION RATE (LHGR)," and LCO 3.2.4, "Average Power Range Monitor (APRM) Gain and Setpoint" 5

Insert ASA 5

Reactor steam dome pressure satisfies the requirements of Criterion 2 of the NRC Policy Statement. 10 CFR 50.36(c)(2)(ii) 5

LCO

The specified reactor steam dome pressure limit of ≤ [1020] 1005 psig ensures the plant is operated within the assumptions of the transient analyses. Operation above the limit may result in a transient response more severe than analyzed. 3 5 reactor overpressure analyses

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)

5

INSERT ASA

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)

MODES, the reactor may be generating significant steam and the design basis accidents 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 occurring while pressure is greater than the limit is minimized. If the operator is unable to restore the reactor steam dome pressure to below the limit, then the reactor should be placed in MODE 3 to be operating within the assumptions of the transient analyses.

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 3.4.10.1

Verification that reactor steam dome pressure is ≤ ~~1020~~ psig ensures that the initial conditions of the design basis accidents 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.

(continued)

BASES (continued)

REFERENCES

1. FSAR, Section ~~{5.2.2.2.4}~~. 1 3
2. FSAR, ~~Section~~ Chapter ~~{15}~~. 3

4 5

JUSTIFICATION FOR DEVIATIONS FROM NUREG-1433, REVISION 1
ITS BASES: 3.4.10 - REACTOR STEAM DOME PRESSURE

1. Changes have been made to reflect those changes made to the Specification.

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. Typographical/grammatical error corrected.

5. 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.

6. Changes have been made to more closely match the LCO requirements.

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.

GENERIC NO SIGNIFICANT HAZARDS CONSIDERATION
ITS: SECTION 3.4 - REACTOR COOLANT SYSTEM

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 Quad Cities 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-1433, 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-1433, 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:
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 explicit requirement in CTS 3.6.A Action 1.e to electrically prohibit the idle recirculation pump from starting except to permit testing in preparation for returning to service is proposed to be deleted. While the inadvertent starting of a recirculation pump in an idle loop is assumed to be an initiator of an analyzed event, this change will not increase the probability of the event since multiple failures would be necessary to initiate the event. Plant operating practice and procedures are adequate to ensure that a recirculation pump is not inadvertently started. In addition, the requirements in CTS 3.6.D (proposed ITS 3.4.9, "RCS Pressure and Temperature (P/T) Limits") will provide assurance that thermal stresses from the startup of a recirculation pump in an idle loop will not exceed design allowances. 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 changes do not involve any design changes, plant modifications, or changes in plant operation. The system will continue to be operated and function in the same way as before the change. Therefore, the proposed changes do 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 does not involve a significant reduction in a margin of safety since this requirement is not necessary to minimize the consequences of any design basis accident. Plant operating practice and procedures are adequate to ensure the pumps are not inadvertently started. In addition, the requirements in CTS 3.6.D (ITS 3.4.9, "RCS Pressure and Temperature (P/T) Limits") will provide assurance that thermal stresses resulting from the startup of an idle recirculation pump will not exceed design allowances.

NO SIGNIFICANT HAZARDS CONSIDERATION
ITS: 3.4.1 - RECIRCULATION LOOPS OPERATING

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 allow continued operation with unmatched recirculation loop flows. While a recirculation loop flow increase and decrease are assumed to be initiators of an analyzed event, this change does not increase the probability of these events. In addition, the loop with lower flow is required to be considered as not in operation. This results in the necessary limit and setpoint changes to return the plant to conditions within the safety analysis. 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 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 conditions return the plant to conditions within the safety analysis within the same completion 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 - 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 jet pump flow indication is not assumed in the initiation of any analyzed event. The requirements for the jet pump flow indication do not need to be explicitly stated in the Technical Specifications. To perform the verifications required for the Surveillance Requirement of ITS SR 3.4.2.1, the jet pump flow indication must be OPERABLE. If the jet pump flow indication is inoperable, these verifications cannot be satisfied and the appropriate actions must be taken for jet pump inoperability in accordance with the ACTIONS of ITS 3.4.2. 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 jet pump flow indication requirements from Technical Specifications does not impact any margin of safety. The requirements for jet pump flow indication do not need to be explicitly stated in the Technical Specifications. To perform the verifications required for the Surveillance Requirement of ITS 3.4.2, the jet pump flow indication must be OPERABLE. If the jet pump flow indication is inoperable, these verifications cannot be satisfied and the appropriate actions must be taken for jet pump inoperability in accordance with the ACTIONS of ITS 3.4.2. As a result, the OPERABILITY of the jet pump flow indication will be maintained to satisfy the associated SR of ITS 3.4.2 without the need for explicit indication requirements in the Technical Specifications. Therefore, this change does not involve a significant reduction in a margin of safety.

NO SIGNIFICANT HAZARDS CONSIDERATION
ITS: 3.4.2 - 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 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 - SAFETY AND RELIEF VALVES

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 requirement to close an open relief valve or place the reactor mode switch in Shutdown is not assumed in the initiation of any analyzed event. The requirement of Action 1 of CTS 3.6.F was provided to ensure that, in the event of an open relief valve, which could not be closed, the reactor mode switch would be placed in the Shutdown position in anticipation of exceeding a suppression pool average temperature of 110°F. However, Required Action D.1 of ITS 3.6.2.1 will still require that the reactor mode switch be immediately placed in Shutdown if the suppression pool average temperature is ≥ 110°F. As such, the Required Actions of ITS 3.6.2.1 are adequate to ensure that the reactor mode switch will immediately be placed in the Shutdown position if the suppression pool average temperature exceeds 110°F. As a result, accident consequences are unaffected by the deletion of the requirement to place the reactor mode switch in the Shutdown position if an open relief valve is unable to be closed. 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?

   This change deletes the requirement to place the reactor mode switch in the Shutdown position if an open relief valve is unable to be closed. This requirement of Action 1 of CTS 3.6.F was provided to ensure that, in the event of an open relief valve which could not be closed, the reactor mode switch would be placed in the Shutdown position in anticipation of exceeding a suppression pool average temperature of 110°F. However, Required Action D.1 of ITS 3.6.2.1 will still require that the reactor mode switch be immediately placed in Shutdown if the suppression pool average temperature is ≥ 110°F. As such, the Required Actions of ITS 3.6.2.1 are adequate to ensure that the reactor mode switch will immediately be placed in the Shutdown position if the

L.1 CHANGE

3. (continued)

suppression pool average temperature exceeds 110°F. In addition, Emergency Operating Procedures address the appropriate actions to take in response to an open relief valve. As a result, continued assurance is provided that plant operation will be maintained with safety analysis assumptions. Therefore, this change does not involve a significant reduction in a margin of safety.

NO SIGNIFICANT HAZARDS CONSIDERATION
ITS: 3.4.4 - 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 4.6.H.2, the RCS Operational LEAKAGE verification, so that it is required to be performed every 12 hours instead of 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 will provide the same assurance as the current 8 hour, not to exceed 12 hour, frequency. Therefore, this change does not involve a significant reduction in the margin of safety.

NO SIGNIFICANT HAZARDS CONSIDERATION
ITS: 3.4.5 - 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?

   Mode changes are proposed to be allowed with the drywell floor drain sump monitoring system or the required primary containment atmospheric monitoring system 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.

NO SIGNIFICANT HAZARDS CONSIDERATION
ITS: 3.4.5 - RCS LEAKAGE DETECTION INSTRUMENTATION

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 does not involve any physical alteration of plant systems, structures or components, changes in parameters governing normal plant operation, or methods of operation. The proposed change extends the time the primary containment atmospheric particulate radioactivity sampling system is permitted to be inoperable from 24 hours to 30 days. The primary containment atmospheric particulate radioactivity sampling system is not assumed to be the initiator of any accident previously evaluated. Therefore, the probability of an accident previously evaluated is not significantly increased. Another form of leakage detection is still available during the extended interval. RCS unidentified and total leakage must be determined every 12 hours in accordance with SR 3.4.4.1. In addition, ITS 3.4.5 Required Action B.1 requires grab samples of the primary containment atmosphere to be analyzed once per 12 hours during the proposed 30 day period. This change will not alter assumptions relative to the mitigation of an accident or transient event since the primary containment atmospheric particulate radioactivity sampling system is not assumed to operate to mitigate the consequences of an accident. Therefore, allowing 30 days to comply with the LCO will not affect the consequences of an accident. The 30 days will allow time to restore the monitoring system to OPERABLE status and possibly avoid a shutdown. Shutting down the plant is a transient which puts thermal stress on components. 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 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 and Required Actions will ensure adequate information is maintained for 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 instrumentation 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. This change will minimize unnecessary entry/exit into the actions for OPERABLE instrumentation. Operational history indicates that the performance of the Surveillances associated with RCS Leakage Detection Instrumentation normally demonstrate that the instrumentation is in fact OPERABLE. CTS 4.0.C (and as modified in the Discussion of Changes for Section 3.0) allows entry into the actions to be delayed for missed Surveillances for up to a period of 24 hours, therefore this short time period of 6 hours for instrumentation tested within the required Surveillance Frequency should also be permitted. The RCS Leakage Detection Instrumentation is not credited in the mitigation of any accident previously evaluated. 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.

NO SIGNIFICANT HAZARDS CONSIDERATION
ITS: 3.4.6 - RCS SPECIFIC ACTIVITY

There were no plant specific less restrictive changes identified for this Specification.

NO SIGNIFICANT HAZARDS CONSIDERATION
ITS: 3.4.7 - 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 will allow one RHR shutdown cooling subsystem to be inoperable for 2 hours in order to perform Surveillances. Currently, the 2 hour allowance is not applicable. While the UFSAR evaluates the loss of all RHR shutdown cooling, the event is not an assumed accident. In addition, the change still requires the other RHR shutdown cooling subsystem to be OPERABLE when the 2 hour allowance is used. The remaining RHR shutdown cooling subsystem is capable of performing the decay heat removal function. 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 one RHR shutdown cooling subsystem to be inoperable for up to 2 hours in order to perform Surveillances. Currently, the 2 hour allowance is not applicable. The change does not affect the requirement for the other RHR shutdown cooling subsystem to be OPERABLE during the 2 hour period. The other RHR shutdown cooling subsystem is capable of performing the decay heat removal function. Therefore, the short period is considered acceptable, and 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 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 verify the system is OPERABLE after reaching the applicable conditions. Since the system cannot be verified to be OPERABLE 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.

NO SIGNIFICANT HAZARDS CONSIDERATION
ITS: 3.4.8 - RHR SHUTDOWN COOLING SYSTEM — COLD 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 will allow one RHR shutdown cooling subsystem to be inoperable for 2 hours in order to perform Surveillances. While the UFSAR evaluates the loss of all RHR shutdown cooling, the event is not an assumed accident. In addition, the change still requires the other RHR shutdown cooling subsystem to be OPERABLE when the 2 hour allowance is used. The remaining RHR shutdown cooling subsystem is capable of performing the decay heat removal function. 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 one required RHR shutdown cooling subsystem to be inoperable for up to 2 hours in order to perform Surveillances. The change does not affect the requirement for the other RHR shutdown cooling subsystem to be OPERABLE during the 2 hour period. The other RHR shutdown cooling subsystem is capable of performing the decay heat removal function. Therefore, the short period is considered acceptable, and 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 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 required RHR shutdown cooling subsystems to be inoperable in Cold Shutdown (MODE 4) for an indefinite period as long as an alternate method of decay heat removal is available for each inoperable RHR shutdown cooling subsystem, reactor coolant is circulating, and coolant temperature is being monitored. Since the normal method of decay heat removal is inoperable, the Completion Times associated with the Actions require frequent verification (e.g., once per 12 hours thereafter, or once per hour). RHR shutdown cooling is not credited in any event analyzed in the UFSAR. In addition, 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. This change is also consistent with allowances already approved at both Dresden and LaSalle and is consistent with NUREG-1433, Rev. 1. 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?

   This proposed change will allow both required RHR shutdown cooling subsystems to be inoperable in Cold Shutdown (MODE 4) for an indefinite period as long as an alternate method of decay heat removal is available for each inoperable RHR shutdown cooling subsystem, reactor cooling is circulating, and coolant temperature is being monitored. Since the normal method of decay heat removal is inoperable the Completion Times associated with the Actions require frequent verifications (e.g., once per 12 hours thereafter, or once per hour). In addition, 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. This change is also consistent with allowances already approved at both Dresden and LaSalle and is consistent with NUREG-1433, Rev. 1. Therefore, this change does not involve a significant reduction in a margin of safety.

NO SIGNIFICANT HAZARDS CONSIDERATION
ITS: 3.4.9 - RCS PRESSURE AND TEMPERATURE (P/T) LIMITS

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 relaxes the Completion Time requirement for an engineering evaluation to determine the acceptability of the Reactor Coolant System (RCS) for continued operation while in conditions other than MODES 1, 2, or 3. The proposed change imposes an operational limit by requiring the engineering evaluation to be completed demonstrating the acceptability of the RCS for operation prior to entering MODES 2 or 3 in lieu of specifying a Completion Time. While a failure to maintain RCS integrity is an accident initiator, this change will not increase the probability of the accident since continued plant operation is prohibited until RCS integrity is assured. Furthermore, the stresses on the RCS and potential for the propagation of RPV flaws in MODE 4 (Cold Shutdown) or 5 (Refueling) are significantly reduced from operating conditions (MODE 1, 2, or 3) due to the reduced pressures and temperatures involved. Therefore, the proposed change will not significantly 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 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 requires an engineering evaluation to establish that the RCS is acceptable for continued operation prior to allowing a transition to plant operating conditions (MODE 2 or 3). This change does not involve a significant reduction in a margin of safety since the proposed change prohibits a return to operating conditions until RCS integrity is assured.

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 will allow the control rods to be withdrawn without verifying within 15 minutes prior to the withdrawal the rate of change of the primary system coolant temperature is within limits. The verification is only required every 30 minutes during RCS heatup and cooldown operations (SR 3.4.9.1). The primary coolant temperature is not expected to change considerably until the reactor becomes critical, therefore, this Surveillance is not necessary. CTS 4.6.K.2.b, the requirement to verify the reactor vessel metal temperature and pressure to be within the Acceptable Region of the critical core operations curve (CTS Figure 3.6.K-3) once within 15 minutes prior to control rod withdrawal for the purpose of achieving criticality, is being retained in ITS SR 3.4.9.2. The proposed Frequencies of SR 3.4.9.1 and 3.4.9.2 are considered acceptable to ensure the RCS P/T limits are met during critical operations. Therefore, this change does not significantly increase the probability or 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 most surveillances only confirm that the limits are being met. This change will allow the control rods to be withdrawn without verifying within 15 minutes prior to the withdrawal the rate of change of the primary system coolant temperature is within limits. The verification is only required every 30 minutes during RCS heatup and cooldown operations (SR 3.4.9.1). The primary coolant temperature is not expected to change considerably until the reactor becomes critical, therefore this Surveillance is not necessary. CTS 4.6.K.2.b, the requirement to verify the reactor vessel metal temperature and pressure to be within the Acceptable Region of the critical core

L.2 CHANGE

3. (continued)

operations curve (CTS Figure 3.6.K-3) once within 15 minutes prior to control rod withdrawal for the purpose of achieving criticality, is being retained in ITS SR 3.4.9.2. The proposed Frequencies of SR 3.4.9.1 and 3.4.9.2 are considered acceptable to ensure the RCS P/T limits are met during critical operations. The proposed Surveillance Frequencies are consistent with the frequencies provided in the BWR ISTS, NUREG-1433, Rev. 1, which has been previously approved by the NRC.

NO SIGNIFICANT HAZARDS CONSIDERATION
ITS: 3.4.10 - REACTOR STEAM DOME PRESSURE

There were no plant specific less restrictive changes identified for this Specification.

NO SIGNIFICANT HAZARDS CONSIDERATION
CTS: 3/4.6.N - 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:

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.

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 four relief valves shall be OPERABLE.

APPLICABILITY: MODE 1,
MODES 2 and 3, except high pressure coolant injection (HPCI) and 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 Coolant Injection (LPCI) pump inoperable.	A.1 Restore LPCI pump to OPERABLE status.	30 days
B. One LPCI subsystem inoperable for reasons other than Condition A. OR One Core Spray subsystem inoperable.	B.1 Restore low pressure ECCS injection/spray subsystem to OPERABLE status.	7 days
C. One LPCI pump in each subsystem inoperable.	C.1 Restore one LPCI pump to OPERABLE status.	7 days

(continued)

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
D. Two LPCI subsystems inoperable for reasons other than Condition C.	D.1	Restore one LPCI subsystem to OPERABLE status.	72 hours
E. Required Action and associated Completion Time of Condition A, B, C, or D not met.	E.1	Be in MODE 3.	12 hours
	AND		
	E.2	Be in MODE 4.	36 hours
F. HPCI System inoperable.	F.1	Verify by administrative means RCIC System is OPERABLE.	Immediately
	AND		
	F.2	Restore HPCI System to OPERABLE status.	14 days
G. HPCI System inoperable.	G.1	Restore HPCI System to OPERABLE status.	72 hours
AND	OR		
One low pressure ECCS injection/spray subsystem is inoperable or Condition C entered.	G.2	Restore low pressure ECCS injection/spray subsystem(s) to OPERABLE status.	72 hours
H. One required ADS valve inoperable.	H.1	Restore ADS valve to OPERABLE status.	14 days

(continued)

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
I. Required Action and associated Completion Time of Condition F, G, or H not met. OR Two or more required ADS valves inoperable.	I.1 AND I.2	Be in MODE 3. Reduce reactor steam dome pressure to ≤ 150 psig.	12 hours 36 hours
J. Two or more low pressure ECCS injection/spray subsystems inoperable for reasons other than Condition C or D. OR HPCI System and one or more required ADS valves inoperable. OR One or more low pressure ECCS injection/spray subsystems inoperable and one or more required ADS valves inoperable.	J.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	----------NOTE---------- Low pressure coolant injection (LPCI) subsystems may be considered OPERABLE during alignment and operation for decay heat removal with reactor steam dome pressure less than the Residual Heat Removal (RHR) cut-in permissive pressure in MODE 3, if capable of being manually realigned and not otherwise inoperable. ---------- 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 correct breaker alignment to the LPCI swing bus.	31 days
SR 3.5.1.4	Verify each recirculation pump discharge valve cycles through one complete cycle of full travel or is de-energized in the closed position.	In accordance with the Inservice Testing Program

(continued)

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.5.1.5	Verify the following ECCS pumps develop the specified flow rate against a test line pressure corresponding to the specified reactor pressure. SYSTEM / FLOW RATE / NO. OF PUMPS / TEST LINE PRESSURE CORRESPONDING TO A REACTOR PRESSURE OF Core Spray: ≥ 4500 gpm, 1, ≥ 90 psig LPCI: ≥ 9000 gpm, 2, ≥ 20 psig	In accordance with the Inservice Testing Program
SR 3.5.1.6	----------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 ≤ 1005 and ≥ 920 psig, the HPCI pump can develop a flow rate ≥ 5000 gpm against a system head corresponding to reactor pressure.	In accordance with the Inservice Testing Program
SR 3.5.1.7	----------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 ≤ 180 psig, the HPCI pump can develop a flow rate ≥ 5000 gpm against a system head corresponding to reactor pressure.	24 months

(continued)

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.5.1.8	----------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 3.5.1.9	----------NOTE---------- Valve actuation may be excluded. ---------- Verify the ADS actuates on an actual or simulated automatic initiation signal.	24 months
SR 3.5.1.10	----------NOTE---------- Not required to be performed until 12 hours after reactor steam pressure and flow are adequate to perform the test. ---------- Verify each required ADS valve opens when manually actuated.	24 months
SR 3.5.1.11	Verify automatic transfer capability of the LPCI swing bus power supply from the normal source to the backup source.	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 low pressure ECCS injection/spray subsystems shall be OPERABLE.

APPLICABILITY: MODE 4,
MODE 5, except with the spent fuel storage pool gates removed and water level ≥ 23 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
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

(continued)

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
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 ECCS injection/spray subsystem, the: a. Suppression pool water level is ≥ 8.5 ft; or b. ----------NOTE---------- Only one required ECCS injection/spray subsystem may take credit for this option during OPDRVs. ---------- Contaminated condensate storage tank water level is ≥ 7.5 ft.	12 hours
SR 3.5.2.2	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 3.5.2.3	----------NOTE---------- One LPCI subsystem may be considered OPERABLE during alignment and operation for decay heat removal if capable of being manually realigned and not otherwise inoperable. ---------- 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)

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.5.2.4	Verify each required ECCS pump develops the specified flow rate against a test line pressure corresponding to the specified reactor pressure. SYSTEM — FLOW RATE — NO. OF PUMPS — TEST LINE PRESSURE CORRESPONDING TO A REACTOR PRESSURE OF CS — ≥ 4500 gpm — 1 — ≥ 90 psig LPCI — ≥ 4500 gpm — 1 — ≥ 20 psig	In accordance with the Inservice Testing Program
SR 3.5.2.5	----------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

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 Coolant Injection System is OPERABLE.	Immediately
	AND		
	A.2	Restore RCIC System to OPERABLE status.	14 days
B. Required Action and associated Completion Time not met.	B.1	Be in MODE 3.	12 hours
	AND		
	B.2	Reduce reactor steam dome pressure to ≤ 150 psig.	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 ≤ 1005 psig and ≥ 920 psig, the RCIC pump can develop a flow rate ≥ 400 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 ≤ 180 psig, the RCIC pump can develop a flow rate ≥ 400 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 consists of the High Pressure Coolant Injection (HPCI) System, the Core Spray (CS) System, the low pressure coolant injection (LPCI) mode of the Residual Heat Removal (RHR) System, and 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 contaminated condensate storage tank (CCST), it is capable of providing a source of water for the HPCI, LPCI and CS systems.

On receipt of an initiation signal, ECCS pumps automatically start; the system aligns and the pumps inject water, taken either from the CCST or 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 HPCI pump discharge pressure almost immediately exceeds that of the RCS, and the pump injects coolant into the vessel to cool the core. If the break is small, the HPCI System will maintain coolant inventory as well as vessel level while the RCS is still pressurized. If HPCI fails, it is backed up by ADS in combination with LPCI and CS. In this event, the ADS timed sequence would be allowed to time out and open the relief valves and safety/relief valve (S/RV) depressurizing the RCS, thus allowing the LPCI and CS to overcome RCS pressure and inject coolant into the vessel. If the break is large, RCS pressure initially drops rapidly and the LPCI and CS 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 RHR Service Water System. Depending on the location and size

(continued)

BASES

BACKGROUND (continued)

of the break, portions of 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.

The combined operation of 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 equipment.

The CS System (Ref. 1) is composed of two independent subsystems. Each subsystem consists of a motor driven pump, a spray sparger above the core, and piping and valves to transfer water from the suppression pool to the sparger. The CS System is designed to provide cooling to the reactor core when reactor pressure is low. Upon receipt of an initiation signal, the CS pumps in both subsystems are automatically started immediately when normal AC power is available and approximately 13 seconds after emergency power is available. When the RPV pressure drops sufficiently, CS System 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 CS System without spraying water in the RPV.

LPCI is an independent operating mode of the RHR System. There are two LPCI subsystems (Ref. 2), each consisting of two motor driven pumps and piping and valves to transfer water from the suppression pool to the RPV via the selected recirculation loop. The LPCI System is equipped with a loop select logic that determines which, if any, of the recirculation loops has been broken and selects the non-broken loop for injection. If neither loop is determined to be broken, then "B" recirculation loop is selected for injection. The LPCI System cross-tie valves must be open to support OPERABILITY of both LPCI subsystems. Similarly, the LPCI swing bus is required to be energized to support both LPCI subsystems. Therefore, with the LPCI cross-tie valves not full open, or the LPCI swing bus not energized, both LPCI subsystems must be considered inoperable. The LPCI subsystems are designed to provide core cooling at low RPV pressure. Upon receipt of an initiation signal, all four LPCI pumps are automatically started (B and D pumps immediately when normal AC power is available and in approximately 8 seconds after emergency AC power is available, and A and C pumps immediately when

(continued)

BASES

BACKGROUND (continued)

normal AC power is available and in approximately 3 seconds after emergency AC power is available). 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 selected recirculation loop. When the RPV pressure drops sufficiently, the LPCI flow to the RPV, via the selected recirculation loop, begins. The water then enters the reactor through the jet pumps. Full flow test lines are provided for each LPCI subsystem to route water from and to the suppression pool, to allow testing of the LPCI pumps without injecting water into the RPV. These test lines also provide suppression pool cooling capability, as described in LCO 3.6.2.3, "RHR Suppression Pool Cooling."

The HPCI System (Ref. 3) 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, where the coolant is distributed within the RPV through the feedwater sparger. Suction piping for the system is provided from the CCST and the suppression pool. Pump suction for HPCI is normally aligned to the CCST source to minimize injection of suppression pool water into the RPV. However, if the CCST water supply is low, or if the suppression pool level is high, an automatic transfer to the suppression pool water source ensures a water supply for continuous operation of the HPCI System. The steam supply to the HPCI turbine is piped from a main steam line upstream of the associated inboard main steam isolation valve.

The HPCI System is designed to provide core cooling for a wide range of reactor pressures (150 psig to 1120 psig). Upon receipt of an initiation signal, the HPCI turbine stop valve and turbine steam supply valve open simultaneously and the turbine accelerates to a specified speed. As the HPCI flow increases, the turbine control valve is automatically adjusted to maintain design flow. Exhaust steam from the HPCI turbine is discharged to the suppression pool. A full flow test line is provided to route water from and to the CCST to allow testing of the HPCI System during normal operation without injecting water into the RPV.

(continued)

BASES

BACKGROUND (continued)

The ECCS pumps are provided with minimum flow bypass lines, which discharge to the suppression pool. The valves in these lines automatically open or remain open 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, all ECCS pump discharge lines are filled with water. The LPCI and CS System discharge lines are kept full of water using a "keep fill" system (jockey pump system). The HPCI System is normally aligned to the CCST. The height of water in the CCST is sufficient to maintain the piping full of water up to the first closed isolation valve in the discharge piping. The relative height of the feedwater line connection for HPCI is such that the water in the feedwater lines keeps the remaining portion of the HPCI discharge line full of water. Therefore, HPCI does not require a "keep fill" system.

The ADS (Ref. 4) consists of 5 valves (4 relief valves and one S/RV). It is designed to provide depressurization of the RCS during a small break LOCA if HPCI 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 (CS and LPCI), so that these subsystems can provide coolant inventory makeup. The S/RV used for automatic depressurization is equipped with one air accumulator and associated inlet check valve. The accumulator provides the pneumatic power to actuate the valve. However, the S/RV is not credited in the safety analysis since qualification of the accumulator for this valve to perform the ADS function has not been demonstrated (Ref. 5).

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 6 and 7. The required analyses and assumptions are defined in Reference 8. The results of these analyses are also 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 $\leq$ 2200°F;

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

b. Maximum cladding oxidation is ≤ 0.17 times the total cladding thickness before oxidation;

c. Maximum hydrogen generation from a 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 8. For GE fuel, the recirculation suction line break with 125 VDC battery failure is considered the most limiting break/failure combination. In addition, to the 125 VDC battery failure, HPCI was assumed to fail to be consistent with generic analyses performed for the BWR 3/4 design generic analysis. For Siemens fuel, the recirculation suction line break with a failure of the entire LPCI System is considered the most limiting break/failure combination. For a small break LOCA, HPCI failure is the most severe failure. In the analysis of events requiring ADS operation, it is assumed that only three of the five ADS valves operate. Therefore, four ADS valves are required to be OPERABLE to meet single failure criteria. The remaining OPERABLE ECCS subsystems provide the capability to adequately cool the core and prevent excessive fuel damage.

The ECCS satisfy Criterion 3 of 10 CFR 50.36(c)(2)(ii).

LCO

Each ECCS injection/spray subsystem and four ADS relief valves are required to be OPERABLE. The S/RV can not be used to satisfy the ADS requirement. The ECCS injection/spray subsystems are defined as the two CS subsystems, the two LPCI subsystems, and one HPCI System. The low pressure ECCS injection/spray subsystems are defined as the two CS subsystems and the two LPCI subsystems.

With less than the required number of ECCS subsystems OPERABLE, the potential exists that during a limiting design basis LOCA concurrent with the worst case single failure,

(continued)

BASES

LCO (continued)

the limits specified in Reference 10 could be exceeded. All ECCS subsystems must therefore be OPERABLE to satisfy the single failure criterion required by Reference 10.

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 LPCI pump is not operating or when the system is realigned from or to the RHR shutdown cooling mode. 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, when reactor steam dome pressure is $\leq$ 150 psig, ADS and HPCI are not required to be OPERABLE because the low pressure ECCS subsystems can provide sufficient flow 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 LPCI pump is inoperable, the inoperable pump must be restored to OPERABLE status within 30 days. In this Condition, the remaining OPERABLE pumps provide adequate core cooling during a LOCA. However, overall ECCS reliability is reduced, because a single failure in one of the remaining OPERABLE LPCI subsystems, concurrent with a LOCA, may result in the LPCI subsystems not being able to perform their intended safety function. The 30 day Completion Time is based on a reliability study cited in Reference 11 that evaluated the impact on ECCS availability, assuming 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 allowable repair times (i.e., Completion Times).

(continued)

BASES

ACTIONS (continued)

B.1

If a LPCI subsystem is inoperable for reasons other than Condition A, or a CS subsystem is inoperable, the inoperable low pressure ECCS injection/spray 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. 11) that evaluated the impact on ECCS availability, assuming 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).

C.1

If one LPCI pump in each subsystem is inoperable, one LPCI pump must be restored to OPERABLE status within 7 days. In this Condition, the remaining OPERABLE ECCS subsystems provide adequate core cooling during a LOCA. However, overall ECCS reliability is reduced because a single failure in one of the remaining OPERABLE ECCS 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. 11) that evaluated the impact on ECCS availability, assuming 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).

D.1

If two LPCI subsystems are inoperable for reasons other than Condition C, one inoperable subsystem must be restored to OPERABLE status within 72 hours. In this Condition, the

(continued)

BASES

ACTIONS

D.1 (continued)

remaining OPERABLE CS subsystems provide adequate core cooling during a LOCA. However, overall ECCS reliability is reduced, because a single failure in one of the remaining CS subsystems, concurrent with a LOCA, may result in ECCS not being able to perform its intended safety function. The 72 hour Completion Time is based on a reliability study cited in Reference 11 that evaluated the impact on ECCS availability, assuming 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 allowable repair times (i.e., Completion Times).

E.1 and E.2

If any Required Action and associated Completion Time of Condition A, B, C, or D is 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 and F.2

If the HPCI System is inoperable and the RCIC System is verified to be OPERABLE, the HPCI 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 ADS. Also, the RCIC System will automatically provide makeup water at most reactor operating pressures. Verification of RCIC OPERABILITY is therefore required immediately when HPCI is inoperable. This may be performed as an administrative check by examining logs or other information to determine if RCIC is out of service for maintenance or other reasons. It does not mean to perform the Surveillances needed to demonstrate the OPERABILITY of

(continued)

BASES

ACTIONS

F.1 and F.2 (continued)

the RCIC System. If the OPERABILITY of the RCIC System cannot be verified, however, Condition I must be immediately entered. In the event of component failures concurrent with a design basis LOCA, there is a potential, depending on the specific failures, that the minimum required ECCS equipment will not be available. A 14 day Completion Time is based on a reliability study cited in Reference 11 and has been found to be acceptable through operating experience.

G.1 and G.2

If any one low pressure ECCS injection/spray subsystem, or one LPCI pump in both LPCI subsystems, is inoperable in addition to an inoperable HPCI System, the inoperable low pressure ECCS injection/spray subsystem(s) or the HPCI System must be restored to OPERABLE status within 72 hours. In this Condition, adequate core cooling is ensured by the OPERABILITY of the ADS and the remaining low pressure ECCS subsystems. However, the overall ECCS reliability is significantly reduced 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 both a high pressure system (HPCI) and a low pressure subsystem(s) are inoperable, a more restrictive Completion Time of 72 hours is required to restore either the HPCI System or the low pressure ECCS injection/spray subsystem(s) to OPERABLE status. This Completion Time is based on a reliability study cited in Reference 11 and has been found to be acceptable through operating experience.

H.1

The LCO requires four ADS valves to be OPERABLE in order to provide the ADS function. Reference 12 contains the results of an analysis that evaluated the effect of two ADS valves being out of service. Per this analysis, operation of only three 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,

(continued)

BASES

ACTIONS

H.1 (continued)

operation is only allowed for a limited time. The 14 day Completion Time is based on a reliability study cited in Reference 11 and has been found to be acceptable through operating experience.

I.1 and I.2

If any Required Action and associated Completion Time of Condition F, G, or H 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 ≤ 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.

J.1

When multiple ECCS subsystems are inoperable, as stated in Condition J, the plant is in a condition outside of the accident analyses. 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 HPCI System, CS System, and LPCI subsystems full of water ensures that the ECCS will perform properly, injecting its full capacity into the RCS upon demand. This will also prevent a water hammer following an ECCS initiation signal. One acceptable method of ensuring that the lines are full is to vent at the high points. The 31 day Frequency is based on the gradual nature of void buildup in the ECCS piping, the procedural controls governing system operation, and operating experience.

(continued)

BASES

SURVEILLANCE REQUIREMENTS (continued)

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 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 HPCI 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 once 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 only affect a single subsystem. This Frequency has been shown to be acceptable through operating experience.

In MODE 3 with reactor steam dome pressure less than the actual RHR cut-in permissive pressure, the RHR System may be required to operate in the shutdown cooling mode to remove decay heat and sensible heat from the reactor. Therefore, this SR is modified by a Note that allows LPCI subsystems to be considered OPERABLE during alignment and operation for decay heat removal, 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 being realigned from or to the RHR shutdown cooling mode. At the low pressures and decay heat loads associated with operation in MODE 3 with reactor steam dome pressure less than the RHR cut-in permissive pressure, a reduced complement of low pressure ECCS subsystems should provide the required core cooling, thereby allowing operation of RHR shutdown cooling, when necessary.

(continued)

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.5.1.3

Verification every 31 days of the correct breaker alignment to the LPCI swing bus demonstrates that the AC electrical power is available to ensure proper operation of the associated LPCI injection valves and the recirculation pump discharge valves. The 31 day Frequency has been found acceptable based on engineering judgment and operating experience.

SR 3.5.1.4

Cycling the recirculation pump discharge valves through one complete cycle of full travel demonstrates that the valves are mechanically OPERABLE and will close when required. Upon initiation of an automatic LPCI subsystem injection signal, these valves are required to be closed to ensure full LPCI subsystem flow injection in the reactor via the recirculation jet pumps. De-energizing the valve in the closed position will also ensure the proper flow path for the LPCI subsystem. Acceptable methods of de-energizing the valve include de-energizing breaker control power, racking out the breaker or removing the breaker.

The Frequency of this SR is in accordance with the Inservice Testing Program. If any recirculation pump discharge valve is inoperable and in the open position, both LPCI subsystems must be declared inoperable.

SR 3.5.1.5, SR 3.5.1.6, and SR 3.5.1.7

The performance requirements of the low pressure 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 low pressure ECCS pump flow rates ensure that adequate core cooling is provided to satisfy the acceptance criteria of Reference 10. The pump flow rates are verified against a test line pressure or system head equivalent to

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.5.1.5, SR 3.5.1.6, and SR 3.5.1.7 (continued)

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 a LOCA. These values have been established analytically.

The flow tests for the HPCI System are performed at two different pressure ranges such that system capability to provide rated flow against a system head corresponding to reactor pressure is tested at both 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 HPCI System diverts steam flow. Therefore, sufficient time is allowed after adequate pressure and flow are achieved to perform these tests. Reactor steam pressure must be $\geq$ 920 psig to perform SR 3.5.1.6 and $\geq$ 150 psig to perform SR 3.5.1.7. Adequate steam flow is represented by at least 2 turbine bypass valves open, or total steam flow $\geq 10^6$ lb/hr. Reactor startup is allowed prior to performing the low pressure Surveillance test because the reactor pressure is low and the time allowed to satisfactorily perform the Surveillance test 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 HPCI is inoperable.

Therefore, SR 3.5.1.6 and SR 3.5.1.7 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 performing the flow test after the required pressure and flow are reached is sufficient to achieve stable conditions for testing and provides reasonable time to complete the SRs.

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.5.1.5, SR 3.5.1.6, and SR 3.5.1.7 (continued)

The Frequency for SR 3.5.1.5 and SR 3.5.1.6 is in accordance with the Inservice Testing Program requirements. The 24 month Frequency for SR 3.5.1.7 is based on the need to perform the Surveillance under the conditions that apply 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 3.5.1.8

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 HPCI, CS, 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 SR also ensures that the HPCI System will automatically restart on an RPV low-low water level signal received subsequent to an RPV high water level trip and that the HPCI suction is automatically transferred from the CCST to the suppression pool on high suppression pool water level or low CCST water level. The LOGIC SYSTEM FUNCTIONAL TEST performed in LCO 3.3.5.1 overlaps this Surveillance to provide complete testing of the assumed safety 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/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.

(continued)

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.5.1.9

The ADS designated valves 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 3.5.1.10 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 the 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 3.5.1.10.

SR 3.5.1.10

A manual actuation of each ADS valve is performed to verify that the valve and solenoid are functioning properly and that no blockage exists in the valve discharge lines. This is demonstrated by the response of the turbine control or bypass valve or by a change in the measured 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 SR. Adequate pressure at which this SR is to be performed is 300 psig (the pressure

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.5.1.10 (continued)

recommended by the valve manufacturer). Adequate steam flow is represented by at least 2 turbine bypass valves open. Reactor startup is allowed prior to performing this SR 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 and provides adequate time to complete the Surveillance. SR 3.5.1.9 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 Frequency of 24 months is based on the need to perform the 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 3.5.1.11

The LPCI System injection valves and recirculation pump discharge valves are powered from the LPCI swing bus, which must be energized after a single failure, including loss of power from the normal source to the swing bus. Therefore, the automatic transfer capability from the normal power source to the backup power source must be verified to ensure the automatic capability to detect loss of normal power and initiate an automatic transfer to the swing bus backup power source. Verification of this capability every 24 months ensures that AC electrical power is available for proper operation of the associated LPCI injection valves and recirculation pump valves. The swing bus automatic transfer scheme must be OPERABLE for both LPCI subsystems to be OPERABLE. The Frequency of 24 months is based on the need

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BASES

SURVEILLANCE REQUIREMENTS

SR 3.5.1.11 (continued)

to perform the Surveillance under the conditions that apply during a startup from a plant outage. Operating experience has shown that the 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.

REFERENCES

1. UFSAR, Section 6.3.2.1.
2. UFSAR, Section 6.3.2.2.
3. UFSAR, Section 6.3.2.3.
4. UFSAR, Section 6.3.2.4.
5. Letter from J.A. Zwolinski (NRC) to D.L. Farrar (Commonwealth Edison Company), "Resolution of NUREG-0737 Item II.K.3.28, Verify Qualification of Accumulators on Automatic Depressurization Valves," dated June 16, 1986.
6. UFSAR, Section 15.6.4.
7. UFSAR, Section 15.6.5.
8. 10 CFR 50, Appendix K.
9. UFSAR, Section 6.3.3.
10. 10 CFR 50.46.
11. Memorandum from R.L. Baer (NRC) to V. Stello, Jr. (NRC), "Recommended Interim Revisions to LCOs for ECCS Components," December 1, 1975.
12. UFSAR, Section 6.3.3.1.4.

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 Core Spray (CS) System and the 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 judgement, 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. The low pressure ECCS subsystems satisfy Criterion 3 of 10 CFR 50.36(c)(2)(ii).
LCO	Two low pressure ECCS injection/spray subsystems are required to be OPERABLE. The low pressure ECCS injection/spray subsystems consist of two CS subsystems and two LPCI subsystems. Each CS subsystem consists of one motor driven pump, piping, and valves to transfer water from the suppression pool or contaminated condensate storage tank (CCST) to the reactor pressure vessel (RPV). Each LPCI subsystem consists of one motor driven pump, piping, and valves to transfer water from the suppression pool or the CCST to the RPV. A single LPCI pump is required per subsystem because of similar injection capacity in relation to a CS subsystem. One LPCI subsystem may be considered OPERABLE during alignment and operation for decay heat removal, if capable of being manually realigned (remote or local) to the LPCI

(continued)

BASES

LCO (continued)

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 mode. 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 low pressure 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 ≥ 23 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 CS System and the LPCI subsystems can provide core cooling without any depressurization of the primary system.

The High Pressure Coolant Injection System is not required to be OPERABLE during MODES 4 and 5 since the low pressure ECCS injection/spray subsystems can provide sufficient flow to the vessel.

ACTIONS

A.1 and B.1

If any one required low pressure ECCS injection/spray subsystem is inoperable, the inoperable subsystem must be restored to OPERABLE status in 4 hours. In this Condition, the remaining OPERABLE subsystem can provide sufficient vessel flooding capability to recover from an inadvertent vessel draindown. However, overall system reliability is

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BASES

ACTIONS

A.1 and B.1 (continued)

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 low pressure ECCS injection/spray subsystem to OPERABLE status is based on engineering judgment that considered the remaining available subsystem and the low probability of a vessel draindown event.

With the inoperable subsystem not restored to OPERABLE status in the required Completion Time, action must be immediately initiated 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

With both of the required ECCS injection/spray subsystems inoperable, all coolant inventory makeup capability may be unavailable. Therefore, actions must immediately be initiated to suspend OPDRVs 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 ECCS injection/spray subsystem must also be restored to OPERABLE status within 4 hours. The 4 hour Completion Time to restore at least one low pressure 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 required low pressure 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

(continued)

BASES

ACTIONS

C.1, C.2, D.1, D.2, and D.3 (continued)

available in each associated penetration flow path not isolated that is assumed to be isolated to mitigate radioactivity 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 may 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 is indicated). OPERABILITY may be verified by an administrative check, or by examining logs or other information, to determine whether the components are out of service for maintenance or other reasons. It is not necessary to perform the Surveillances needed to demonstrate the OPERABILITY of the components. If, however, any required component is inoperable, then it must be restored to OPERABLE status. In this case, the Surveillance 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

The minimum water level of 8.5 feet above the bottom of the suppression chamber required for the suppression pool is periodically verified to ensure that the suppression pool will provide adequate net positive suction head (NPSH) for the CS System and LPCI subsystem 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 CCST.

When suppression pool level is < 8.5 feet, the CS and LPCI subsystems are considered OPERABLE only if they can take suction from the CCST, and the CCST water level is sufficient to provide the required NPSH for the CS pump and LPCI pump. Therefore, a verification that either the suppression pool water level is ≥ 8.5 feet or that required low pressure ECCS injection/spray subsystems are aligned to take suction from the CCST and the CCST contains ≥ 110,000 gallons of water, equivalent to 7.5 ft, ensures

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BASES

SURVEILLANCE REQUIREMENTS

SR 3.5.2.1 (continued)

that the required low pressure ECCS injection/spray subsystems can supply at least 50,000 gallons of makeup water to the RPV. The water volume below a CCST level of 4 ft (approximately 60,000 gallons) is not considered available to the low pressure pumps since vortexing may occur. However, as noted, only one required low pressure ECCS injection/spray subsystem may take credit for the CCST option during OPDRVs. During OPDRVs, the volume in the CCST may not provide adequate makeup if the RPV were completely drained. Therefore, only one low pressure ECCS injection/spray subsystem is allowed to use the CCST. This ensures the other required ECCS subsystem has adequate makeup volume.

The 12 hour Frequency of these SRs was developed considering operating experience related to suppression pool water level and CCST water level variations and instrument drift during the applicable MODES. Furthermore, the 12 hour Frequency is considered adequate in view of other indications available in the control room, including alarms, to alert the operator to an abnormal suppression pool or CCST water level condition.

SR 3.5.2.2, SR 3.5.2.4, and SR 3.5.2.5

The Bases provided for SR 3.5.1.1, SR 3.5.1.5, and SR 3.5.1.8 are applicable to SR 3.5.2.2, SR 3.5.2.4, and SR 3.5.2.5, respectively.

SR 3.5.2.3

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

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BASES

SURVEILLANCE REQUIREMENTS

SR 3.5.2.3 (continued)

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 be required to operate in the shutdown cooling mode to remove decay heat and sensible heat from the reactor. Therefore, this SR is modified by 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 (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 being realigned from or to the RHR shutdown cooling mode. Because of the 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. This will ensure adequate core cooling if an inadvertent RPV draindown should occur.

REFERENCES

1. UFSAR, Section 6.3.3.1.2.1.

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 makeup water and maintain the RPV water level above the top of the core. Under these conditions, the High Pressure Coolant Injection (HPCI) and RCIC systems perform similar functions. The RCIC System design requirements ensure that the criteria of Reference 1 are satisfied.

The RCIC System (Ref. 1) 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, where the coolant is distributed within the RPV through the feedwater sparger. Suction piping is provided from the contaminated condensate storage tank (CCST) and the suppression pool. Pump suction is normally aligned to the CCST to minimize injection of suppression pool water into the RPV. However, if the CCST 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 RCIC System. The steam supply to the turbine is piped from a main steam line upstream of the associated inboard main steam line isolation valve.

The RCIC System is designed to provide makeup water for a wide range of reactor pressures 150 psig to 1120 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 CCST 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 piping is kept full of water. The RCIC System is normally aligned to the CCST. The height of water ($\geq$ 12 ft) in the CCST is sufficient to maintain the piping full of water up to the first isolation valve in the discharge piping. The relative height of the feedwater line connection for RCIC is such that the water in the feedwater lines keeps the remaining portion of the RCIC discharge line full of water. Therefore, RCIC does not require a "keep fill" system.
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 makeup water such that actuation of any of the low pressure ECCS subsystems is not required in the event of RPV isolation accompanied by a loss of feedwater flow. The RCIC System has sufficient capacity for maintaining RPV inventory during an isolation event.
APPLICABILITY	The RCIC System is required to be OPERABLE during MODE 1, and MODES 2 and 3 with reactor steam dome pressure > 150 psig, since RCIC is the primary non-ECCS water source for makeup when the reactor is isolated and pressurized. In MODES 2 and 3 with reactor steam dome pressure $\leq$ 150 psig, and in MODES 4 and 5, RCIC is not required to be OPERABLE since the low pressure ECCS injection/spray subsystems can provide sufficient flow to the RPV.

(continued)

BASES (continued)

ACTIONS

A.1 and A.2

If the RCIC System is inoperable during MODE 1, or MODE 2 or 3 with reactor steam dome pressure > 150 psig, and the HPCI 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 the overall plant capability to provide makeup inventory at high reactor pressure since the HPCI System is the only high pressure system assumed to function during a loss of coolant accident (LOCA). OPERABILITY of HPCI 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 HPCI is out of service for maintenance or other reasons. It does not mean it is necessary to perform the Surveillances needed to demonstrate the OPERABILITY of the HPCI System. If the OPERABILITY of the HPCI System cannot be immediately verified, however, Condition B must be entered. For transients and certain abnormal events with no LOCA, RCIC (as opposed to HPCI) is the preferred source of makeup coolant because of its relatively small capacity, which allows easier control of the 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. 2) that evaluated the impact on ECCS availability, assuming 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 similar functions of HPCI and RCIC, the AOTs (i.e., Completion Times) determined for HPCI 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 HPCI System is simultaneously inoperable, the plant must be brought to a condition in which the LCO does not apply. To achieve this

(continued)

BASES

ACTIONS

B.1 and B.2 (continued)

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.

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 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. For the RCIC System, this SR also includes the steam flow path for the turbine and the flow controller position.

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.5.3.2 (continued)

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 position would affect only the RCIC System. This Frequency has been shown to be acceptable through operating experience.

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 system head 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 $\geq$ 920 psig to perform SR 3.5.3.3 and $\geq$ 150 psig to perform SR 3.5.3.4. Adequate steam flow is represented by at least 1 turbine bypass valve open. Reactor startup is allowed prior to performing the low pressure Surveillance because the reactor pressure is low and the time allowed 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 Surveillance 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.

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.5.3.3 and SR 3.5.3.4 (continued)

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 the Surveillance under conditions that apply 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 3.5.3.5

The RCIC System is required to actuate automatically in order to verify its design function satisfactorily. This Surveillance verifies that, with a required system initiation signal (actual or simulated), the automatic initiation logic of the RCIC System 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 the RCIC System will automatically restart on an RPV low-low water level signal received subsequent to an RPV high water level trip and that the suction is automatically transferred from the CCST to the suppression pool on a CCST low water level signal. 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.

(continued)

BASES (continued)

REFERENCES

1. UFSAR, Section 5.4.6.

2. Memorandum from R.L. Baer (NRC) to V. Stello, Jr. (NRC), "Recommended Interim Revisions to LCOs for ECCS Components," December 1, 1975.

A.1

ITS 3.5.1

3.5 - LIMITING CONDITIONS FOR OPERATION

A. Emergency Core Cooling System - Operating

LCO 3.5.1 The emergency core cooling systems (ECCS) shall be OPERABLE with:

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

   a. One OPERABLE CS pump, and

   b. An OPERABLE flow path capable of taking suction from the suppression chamber and transferring the water through the spray sparger to the reactor vessel. LA.1

M.1 2. The low pressure coolant injection (LPCI) subsystem comprised of[a]: S

   a. Four OPERABLE LPCI pumps, and

   b. An OPERABLE flow path capable of taking suction from the suppression chamber and transferring the water to the reactor vessel. LA.1

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

   a. One OPERABLE HPCI pump, and

   b. An OPERABLE flow path capable of taking suction from the suppression chamber and transferring the water to the reactor vessel. LA.1

4.5 - SURVEILLANCE REQUIREMENTS

A. Emergency Core Cooling System - Operating

The ECCS shall be demonstrated OPERABLE by:

1. At least once per 31 days:

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

      1) Verifying that the system piping from the pump discharge valve to the system isolation valve is filled with water. SR 3.5.1.1

      2) Verifying that each valve, manual, power operated or automatic, in the flow path that is not locked, sealed, or otherwise secured in position, is in its correct[a] position. SR 3.5.1.2 LA.2

   b. For the HPCI system, verifying that the HPCI pump flow controller is in the correct position. LA.3

add proposed SR 3.5.1.3, SR 3.5.1.4, and SR 3.5.1.11 M.2

SR 3.5.1.2 Note

a The LPCI subsystem may be considered OPERABLE during alignment and operation for decay heat removal when below the actual RHR cut in permissive pressure in MODE 3, if capable of being manually realigned (remote or local) to the LPCI mode and not otherwise inoperable. LA.4

a Except that an automatic valve capable of automatic return to its ECCS position when an ECCS signal is present may be in position for another mode of operation. LA.2

3.5 - LIMITING CONDITIONS FOR OPERATION

LCO 3.5.1

4. The automatic depressurization system (ADS) with at least 5 OPERABLE ADS valves. 4 L.1

APPLICABILITY:

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

ACTION:

1. For the core spray system:

ACTION B

a. With one CS subsystem inoperable, provided that the LPCI subsystem is OPERABLE, restore the inoperable CS subsystem to OPERABLE status within 7 days, or

ACTION E

be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours. M.3

Enter LCO 3.0.3

ACTION J

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

2. For the LPCI subsystem:

ACTION A

a. With one LPCI pump inoperable[f], provided that both CS subsystems are OPERABLE, restore the inoperable LPCI pump to OPERABLE status within 30 days, A.2

4.5 - SURVEILLANCE REQUIREMENTS

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

SR 3.5.1.5

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

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

SR 3.5.1.6

c. The HPCI pump develops a flow of at least 5000 gpm against a system head corresponding to reactor vessel pressure, when steam is being supplied to the turbine between 920 and 1005 psig[b]. A.3

3. At least once per 18 months: 24 L.D.1

SR 3.5.1.8

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

Applicability

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

f The provisions of Specification 3.9.A, Actions 4.a or 6.b are applicable to the LPCI subsystem such that with an inoperable diesel generator, for the remaining OPERABLE diesel generator, both LPCI pumps (and their associated flow path) associated with that OPERABLE diesel generator, shall be OPERABLE. Otherwise, enter Specification 3.5.A Action 2.c. A.2

SR 3.5.1.6 Note

c The provisions of Specification 4.0.D are not applicable provided the surveillance is performed within 12 hours after reactor steam pressure is adequate to perform the test. and flow A.3

3.5 - LIMITING CONDITIONS FOR OPERATION

ACTION E — or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

M.1 A.2

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

ACTION B
ACTION C
ACTION E
A.4
M.1
add proposed ACTION D

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

ACTION J
Enter LCO 3.0.3
A.4
M.3

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

ACTION F
ACTION I

add proposed ACTION G
L.3

4.5 - SURVEILLANCE REQUIREMENTS

b. For the HPCI system, verifying that:

SR 3.5.1.7

1) The system develops a flow of ≥5000 gpm against a system head corresponding to reactor vessel pressure, when steam is being supplied to the turbine between 150 and 180 psig[a].

A.3
LA.3

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

LA.3

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

L.2

d. Deleted.

---

[a] The provisions of Specification 3.9.A, Actions 4.a or 6.b are applicable to the LPCI subsystem such that with an inoperable diesel generator, for the remaining OPERABLE diesel generator, both LPCI pumps (and their associated flow path) associated with that OPERABLE diesel generator, shall be OPERABLE. Otherwise, enter Specification 3.5.A, Action 2.c.

A.2

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

A.4

SR 3.5.1.7 Note

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

A.3

3.5 - LIMITING CONDITIONS FOR OPERATION

4. For the ADS:

ACTION H

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

ACTION I

ACTION I

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

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

L.2

add proposed ACTION J — A.5 — M.3

4.5 - SURVEILLANCE REQUIREMENTS

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

24 — LD.1

SR 3.5.1.9

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

L.1

SR 3.5.1.10

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

A.3

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

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

LA.3

SR 3.5.1.11 Note

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

and flow — A.3

3.5 - LIMITING CONDITIONS FOR OPERATION

4.5 - SURVEILLANCE REQUIREMENTS

6. Deleted.

7. ~~In the event an ECCS system is actuated and injects water into the Reactor Coolant System, a Special Report shall be prepared and submitted to the Commission pursuant to Specification 6.9.B within 90 days describing the circumstances of the actuation and the total accumulated actuation cycles to date. The current value of the usage factor for each affected safety injection nozzle shall be provided in this Special Report whenever its value exceeds 0.70.~~ L.4

DISCUSSION OF CHANGES
ITS: 3.5.1 - ECCS — OPERATING

ADMINISTRATIVE

A.1 In the conversion of the Quad Cities 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-1433, Rev. 1 (i.e., the Improved Standard Technical Specifications (ISTS)).

A.2 CTS 3.5.A Actions 2.a and 2.b footnote (f), which provides a cross reference to CTS 3.9.A has been deleted. The format of the proposed Technical Specifications does not include providing "cross references." Proposed ITS 3.8.1 Required Action B.2 adequately prescribes the necessary actions when redundant required feature(s) are inoperable. Therefore, the existing reference in CTS 3.5.A Actions 2.a and 2.b footnote (f) to CTS 3.9.A serves no functional purpose, and its removal is administrative.

A.3 CTS 4.5.A.2.c and CTS 4.5.A.3.b.l) footnote (c) allow the HPCI flow tests to be performed within 12 hours after adequate reactor steam pressure is available. In addition, CTS 4.5.A.4.b footnote (c) allows the ADS valve actuation test to be deferred until 12 hours after adequate reactor steam 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 Note to SRs 3.5.1.6, 3.5.1.7, and 3.5.1.10) to allow deferral until adequate flow is also available. Therefore, this change is considered administrative.

A.4 The CTS 3.5.A Actions 2.b and 2.c, footnote (d) provides an allowance that when two or more RHR shutdown cooling 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 3.6.O and the proposed ACTIONS of ITS 3.4.7 and 3.4.8. 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 CTS 3.5.A Actions 1, 2, 3, and 4 provide Actions for each specific ECCS (CS, LPCI, HPCI and ADS). ITS 3.5.1 ACTION J provides direction for various interrelationships between ECCS subsystems and ADS. The ACTION requires entry into LCO 3.0.3 for various combinations of inoperable components which are consistent with the present ACTIONS for the same combinations, except as identified in Discussion of Changes M.1 and L.3. Therefore, the statements in CTS 3.5.A Actions 1, 2, 3, and 4 that require the other ECCS equipment to be OPERABLE ("provided that..") are unnecessary and have been deleted.

TECHNICAL CHANGES - MORE RESTRICTIVE

M.1 CTS 3.5.A.2 requires the low pressure coolant injection (LPCI) subsystem to be OPERABLE and comprised of four OPERABLE LPCI pumps and an OPERABLE flow path capable of taking suction from the suppression chamber and transferring the water to the reactor vessel. ITS 3.5.1 will require each ECCS injection subsystem to be OPERABLE. The Bases describes the OPERABILITY requirements for LPCI. There are two LPCI subsystems, each consisting of two motor driven pumps, piping and valves capable of transferring water from the suppression pool to the RPV via the "selected" recirculation loop. Since the CTS only requires that LPCI be able to transfer water to the reactor vessel this change is considered more restrictive on plant operation, however necessary to ensure assumptions of the design basis accidents can be satisfied. In addition, the allowance in CTS 3.5.A Action 2.b which allows the entire LPCI System to be inoperable for 7 days has been modified to allow only one LPCI subsystem to be inoperable (first part of ITS 3.5.1, Condition B) or one LPCI pump in each LPCI subsystem (second part of ITS 3.5.1 Condition C) to be inoperable. A new Action has also been added (ITS 3.5.1 Action D) which allows the entire LPCI System to be inoperable (i.e., both LPCI subsystems inoperable), however the Completion Time associated with this ACTION has been reduced to 72 hours. These changes are acceptable since with one LPCI subsystem inoperable or one LPCI pump in each subsystem inoperable (e.g., 2 pumps inoperable) and the failure of another ECCS subsystem (i.e., another LPCI pump or CS subsystem), the ECCS continues to be able to perform its intended safety function. However, with the entire LPCI System inoperable (i.e., all four pumps or any injection pathway inoperable), the overall ECCS reliability is reduced because a single failure in one of the remaining OPERABLE subsystems (e.g., CS) concurrent with a design basis LOCA will result in the ECCS not being able to perform its intended function. These changes represent additional restrictions on plant operation necessary to maintain overall ECCS reliability.

TECHNICAL CHANGES - MORE RESTRICTIVE (continued)

M.2 Three new Surveillances have been added to the Technical Specifications.

ITS SR 3.5.1.3 will require the verification of correct breaker alignment to the LPCI swing bus every 31 days. Each unit includes only one swing bus therefore this Surveillance will help ensure the required components are in their correct or designed position.

ITS SR 3.5.1.4 will require the verification that each recirculation pump discharge valve cycles through one complete cycle of full travel or is de-energized in the closed position. This will ensure that each valve is capable of closure or is closed as required by the accident analysis. The Frequency is in accordance with the Inservice Testing Program. This Frequency is considered acceptable due to the demonstrated reliability of the valves.

In addition, ITS SR 3.5.1.11 will require verification of the automatic transfer capability of the LPCI swing bus power supply from its normal power source to its backup power source. This will help ensure the ECCS meets its design bases as described in the UFSAR. The proposed Frequency of 24 months is consistent with the operating fuel cycle.

These Surveillances represent additional restrictions on plant operation necessary to help ensure the OPERABILITY of the LPCI subsystems is maintained.

M.3 CTS 3.5.A Action 1.b requires a normal plant shutdown with both CS subsystems inoperable and CTS 3.5.A Action 2.c requires a normal plant shutdown with the LPCI subsystem and one or both CS subsystems inoperable. These same conditions in the ITS will require entry into LCO 3.0.3. While operations in CTS 3.5.A Action 2.c may not necessarily be outside the plant design bases (i.e., both CS subsystems inoperable or one LPCI subsystem inoperable and one or both CS subsystems inoperable), these inoperabilities will require entry into ITS LCO 3.0.3. With HPCI System and one or more required ADS valves inoperable, the CTS will require entry in Specification 3.0.C since the plant is outside its design basis and no condition exist for this condition in CTS 3.5.A. CTS 3.5.A Action 4.a requires a normal plant shutdown with one or more required ADS valves inoperable and one or more low pressure ECCS subsystems inoperable. These same conditions in the proposed ITS will require entry into LCO 3.0.3. Operation in CTS 3.5.A Action 4.a may not necessarily be outside the plant design bases since CTS requires five ADS valves to be OPERABLE (see Discussion of Change L.1). Proposed ITS 3.5.1 requires four ADS valves to be OPERABLE and will require entry into ITS LCO 3.0.3 (proposed ACTION J) since the plant will be outside of the analyzed conditions.

TECHNICAL CHANGES - MORE RESTRICTIVE

M.3 (cont'd) This change represents an additional restriction on plant operation necessary to achieve consistency with other Specifications and BWR ISTS, NUREG-1433, Revision 1.

TECHNICAL CHANGES - LESS RESTRICTIVE

"Generic"

LA.1 The details of CTS 3.5.A relating to ECCS subsystem OPERABILITY (number of pumps and flow path 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.

LA.2 The description in CTS 4.5.A.1.a.2) footnote (a) of what "correct position" means for an automatic valve is proposed to be relocated to the Bases. This detail is not necessary to ensure the automatic valves are in their proper position. The requirement of proposed SR 3.5.1.2 is adequate to ensure the automatic valves are in their proper position and the ECCS subsystems are maintained OPERABLE. As such, this 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.

LA.3 The details of CTS 4.5.A.1.b, 4.5.A.3.b.1), 4.5.A.3.b.2), and 4.5.A.4.b relating to methods for performing Surveillances (i.e., the minimum pressure to perform the low pressure HPCI flow test, verifying the HPCI System pump flow controller is in the correct position, verifying the HPCI suction is automatically transferred from the condensate storage tank to the suppression pool on the proper signals, 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.

TECHNICAL CHANGES - LESS RESTRICTIVE (continued)

LA.4 The description in CTS 3.5.A footnote (e), that one LPCI subsystem may be considered OPERABLE if capable of being manually realigned via "remote or local" methods is proposed to be relocated to the Bases. The detail is not necessary since proposed SR 3.5.1.2 will require the LPCI system valves aligned for decay heat removal to have the capability to be manually realigned. As such, this 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.

LD.1 The Frequencies for performing CTS 4.5.A.3.a, 4.5.A.3.b.1), 4.5.A.3.b.2), 4.5.A.4.a, and 4.5.A.4.b (proposed SRs 3.5.1.8, 3.5.1.7, 3.5.1.9, and 3.5.1.10) have been extended from 18 months to 24 months. The ECCS system functional tests, CTS 4.5.A.3.a (proposed SR 3.5.1.8), ensure that a system initiation signal (actual or simulated) to the automatic initiation logic of HPCI, CS, and LPCI will cause the 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 HPCI flow test, CTS 4.5.A.3.b.1) (proposed SR 3.5.1.7), ensures that the HPCI System can perform its design function by developing the appropriate system flow. The HPCI automatic suction transfer test, CTS 4.5.A.3.b.2 (proposed SR 3.5.1.8 as discussed in Discussion of Change LA.3) ensures the HPCI suction is automatically transferred from the contaminated condensate storage tank to the suppression pool. The ADS System functional test, CTS 4.5.A.4.a (proposed SR 3.5.1.9), ensures the mechanical portions of the ADS function (i.e., solenoids) operate as designed when initiated either by an actual or simulated initiation signal. The ADS valve test, CTS 4.5.A.4.b (proposed SR 3.5.1.10), ensures the valve actuator 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.B 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.B 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

TECHNICAL CHANGES - LESS RESTRICTIVE

LD.1 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 and SR 3.5.1.6) 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 functions. 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 relief valves associated with the 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 a backup to the HPCI System. If HPCI were to fail, ADS must activate to lower reactor pressure so that the low pressure ECCS spray/injection systems may operate. Furthermore, as stated in the NRC Safety Evaluation Report (dated August 2, 1993) relating to extension of the Peach Bottom Atomic Power Station, Unit Numbers 2 and 3 surveillance intervals from 18 to 24 months:

> "Industry reliability studies for boiling water reactors (BWRs) prepared by the BWR Owners Group (NEDC-30936P) show that the overall safety systems' reliabilities are not dominated by the reliabilities of the logic system, but by that of the mechanical components, (e.g., pumps and valves), which are consequently tested on a more frequent basis. Since the probability of a relay or contact failure is small relative to the probability of mechanical component failure, increasing the logic system functional test interval represents no significant change in the overall safety system unavailability."

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 number of ADS valves required to be OPERABLE in CTS 3.5.A.4 is proposed to be reduced from five to four. CTS 3.5.A Action 4.a, which allows one of the five ADS valves to be inoperable for a period of time prior to requiring a shutdown, CTS 3.5.A Action 4.b, which requires a shutdown when two or more ADS valves are inoperable, and CTS 4.5.A.4.b, which requires each ADS valve to be opened, have also been revised to reflect this change. This change is based on the analysis summarized in UFSAR, Section 6.3.3.1.4. This analysis demonstrates adequate core cooling is provided during a small break LOCA and a simultaneous battery failure (i.e., battery failure and resulting HPCI System failure) with two of the five ADS valves out-of-service. This change reflects the credit provided through the use of NRC approved methods for calculating more realistic (yet conservative) peak cladding temperatures during accident situations.

L.2 The CHANNEL CALIBRATION of the ECCS discharge line "keep filled" alarm instrumentation in CTS 4.5.A.3.c does not necessarily relate directly to the OPERABILITY of the ECCS subsystems. The BWR ISTS, NUREG-1433, 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. This instrumentation provides an alarm when the discharge pressure is low. Failure of the alarm does not result in the ECCS subsystem being incapable of performing its intended function. The requirement to verify, for each ECCS injection/spray subsystem, the piping is filled with water from the pump discharge valve to the injection valve (proposed SR 3.5.1.1) will ensure the associated ECCS subsystem is OPERABLE. Therefore, this instrumentation, along with the supporting ACTIONS (CTS 3.5.A Action 5) and Surveillances, are proposed to be deleted.

L.3 Proposed ACTION G is being added to ITS 3.5.1 for the condition of HPCI inoperable coincident with one low pressure coolant injection subsystem (or one LPCI pump in each subsystem) inoperable. The current Technical Specifications require entry into Specification 3.0.C (ITS LCO 3.0.3) for this condition, implying that the plant is outside design basis. The analyses summarized in UFSAR Section 6.3.3 demonstrate that adequate core cooling is provided by the OPERABLE HPCI or ADS System and the remaining OPERABLE low pressure

TECHNICAL CHANGES - LESS RESTRICTIVE

L.3 (cont'd) injection/spray systems. However, the redundancy has been reduced such that another single failure may not maintain the ability to provide adequate core cooling. Proposed ACTION G requires a restrictive Completion Time of 72 hours since both a high pressure (HPCI) and a low pressure subsystem (or one LPCI pump in each subsystem) are inoperable. This Completion Time is based on a reliability study (Memorandum from R. L. Baer (NRC) to V. Stello, Jr. (NRC), "Recommended Interim Revisions to LCOs for ECCS Components," December 1, 1975) and has been found to be acceptable through operating experience.

L.4 The CTS 3.5.A Action 7 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 Quad Cities 1 and 2 will be controlled by plant procedures. Regulations provide sufficient control of these provisions for their removal from Technical Specifications.

RELOCATED SPECIFICATIONS

None

A.1

SR 3.5.2.2
SR 3.5.2.3
SR 3.5.2.4
SR 3.5.2.5

LD.1

3.5 - LIMITING CONDITIONS FOR OPERATION

B. Emergency Core Cooling System - Shutdown (low pressure ECCS injection/spray subsystems)

At least two of the following four subsystems/loops shall be OPERABLE[a]: LA.1

1. One or both core spray (CS) subsystems with:

   a. An OPERABLE flow path capable of taking suction from at least one of the following water sources and transferring the water through the spray sparger to the reactor vessel:

      1) From the suppression chamber, or

      2) When the suppression chamber water level is less than the limit or is drained, from the condensate storage tank containing at least 140,000 available gallons of water. (50,000) L.5

SR 3.5.2.1.6

2. One or both low pressure coolant injection (LPCI) subsystem loops with a subsystem loop comprised of: LA.1

   a. At least one OPERABLE LPCI pump, and

   b. An OPERABLE flow path capable of taking suction from at least one of the following water sources and transferring the water to the reactor vessel:

      1) From the suppression chamber, or

4.5 - SURVEILLANCE REQUIREMENTS

B. Emergency Core Cooling System - Shutdown

The required ECCS shall be demonstrated OPERABLE per Surveillance Requirement 4.5.A, except: A.2

1. The LPCI subsystems cross-tie valves may be closed. LA.1

2. Each LPCI pump develops the required flow when tested pursuant to Specification 4.0.E. M.1

SR 3.5.2.4

add proposed flow rate and head conditions for one pump

SR 3.5.2.3 Note

a One LPCI subsystem may be aligned for decay heat removal and considered OPERABLE for the ECCS function, if it can be manually realigned (remote or local) to the LPCI mode and is not otherwise inoperable. LA.2

A.1

ITS 3.5.2

LA.1

3.5 - LIMITING CONDITIONS FOR OPERATION

4.5 - SURVEILLANCE REQUIREMENTS

SR 3.5.2.1.b

2) When the suppression chamber water level is less than the limit or is drained, from the condensate storage tank containing at least ~~140,000~~ 50,000 available gallons of water. L.5

APPLICABILITY:

OPERATIONAL MODE(s) 4 and 5[b].

ACTION:

ACTION A

1. With one of the above required subsystems/loops inoperable, restore at least two subsystems/loops to OPERABLE status within 4 hours or

ACTION B

suspend all operations with a potential for draining the reactor vessel.

ACTION C

2. With both of the above required subsystems/loops inoperable, suspend CORE ALTERATION(s) and all operations with a potential for draining the reactor vessel. Restore at least one subsystem/loop to OPERABLE status within 4 hours or establish L.1

ACTION D

SECONDARY CONTAINMENT INTEGRITY within the next 8 hours. A.3

A.4

Applicability b 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 Specification 3.10.G and 3.10.H. A.5

A.1

3.5 - LIMITING CONDITIONS FOR OPERATION

C. Suppression Chamber

A.6 moved to ITS 3.6.2.2

The suppression chamber shall be OPERABLE:

1. In OPERATIONAL MODE(s) 1, 2, and 3 with a contained water volume equivalent to a water level of ≥ 14' 1" above the bottom of the suppression chamber.

2. In OPERATIONAL MODE(s) 4 and 5[a] with a contained volume equivalent to a water level of ≥8.5' above the bottom of the suppression chamber, except that the suppression chamber level may be less than the limit provided that:

   a. No operations are performed that have a potential for draining the reactor vessel,

   b. The reactor mode switch is locked in the Shutdown or Refuel position,

   c. The condensate storage tank contains ≥140,000 available gallons of water, and

   d. The ECCS systems are OPERABLE per Specification 3.5.B.

LA.1

L.2

L.3

50,000

L.5

4.5 - SURVEILLANCE REQUIREMENTS

C. Suppression Chamber

The suppression chamber shall be determined OPERABLE by verifying:

1. For OPERATIONAL MODE(s) 1, 2 and 3, at least once per 24 hours, the water level to be ≥ 14' 1".

2. For OPERATIONAL MODE(s) 4 or 5[a], at least once per 12 hours:

   SR 3.5.2.1.a a. The water level to be ≥8.5', or

   SR 3.5.2.1.b b. Verify the alternate conditions of Specification 3.5.C.2, or the conditions of footnote (a), to be satisfied.

L.4

A.7

SR 3.5.2.1
SR 3.5.2.2.a

SR 3.5.2.1.b

add proposed SR 3.5.2.1.b Note

LCO 3.5.2

APPLICABILITY:

A.6 moved to ITS 3.6.2.2

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

A.5

Applicability

a 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 Specification 3.10.G and 3.10.H.

M.2

3.5 - LIMITING CONDITIONS FOR OPERATION

4.5 - SURVEILLANCE REQUIREMENTS

ACTION:

1. In OPERATIONAL MODE(s) 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.

A.6 moved to 3.6.2.2

L.2 add proposed Required Action A.1.

ACTION C

2. In OPERATIONAL MODE(s) 4 or 5[a] with the suppression chamber water level less than the above limit or drained and the above required conditions not satisfied, suspend CORE ALTERATION(s) and all operations that have a potential for draining the reactor vessel and lock the reactor mode switch in the Shutdown position. Establish SECONDARY CONTAINMENT INTEGRITY within 8 hours.

L.1 L.3 A.4 A.3

ACTION C and D

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a 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, water in the cavity is flooded, and the water level is maintained within the limits of Specification 3.10.G and 3.10.H.

A.5

Applicability.

M.2

DISCUSSION OF CHANGES
ITS: 3.5.2 - ECCS — SHUTDOWN

ADMINISTRATIVE

A.1 In the conversion of the Quad Cities 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-1433, Rev. 1 (i.e., the Improved Standard Technical Specifications (ISTS)).

A.2 CTS 4.5.B requires the required ECCS to be demonstrated OPERABLE per CTS 4.5.A. Under the new format of BWR ISTS, NUREG-1433, Revision 1, the individual Surveillance Requirements of CTS 4.5.B 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.A for low pressure ECCS are also presented in the Surveillance Requirements for this Specification. 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 and Discussion of Change M.1 for ITS 3.5.2.

A.3 The CTS 3.5.B Action 2 and CTS 3.5.C Action 2 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-1433, Rev. 1. Because this is an enhanced presentation of existing intent, the proposed change is considered administrative.

ADMINISTRATIVE (continued)

A.4 This proposed change replaces the use of the defined term SECONDARY CONTAINMENT INTEGRITY in CTS 3.5.3 Action 2 and CTS 3.5.C Action 2 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 ISTS, NUREG-1433, Rev. 1, and is considered administrative only.

A.5 The statement in CTS 3.5.B footnote (b) and CTS 3.5.C footnote (a), 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 within the limits of Specifications 3.10.G and 3.10.H. The spent fuel pool gates can be removed and the water level maintained within the limits of CTS 3.10.G and 3.10.H only if the head is also removed and the cavity flooded, since CTS 3.10.G (ITS 3.9.6 and 3.9.7) is applicable only during handling of fuel assemblies or control rods within the reactor pressure vessel. Therefore, these additional words have been deleted as an administrative change.

A.6 The CTS 3.5.C.1 and associated Applicability, Action 1, and CTS 4.5.C.1 requirements are being moved to ITS 3.6.2.2 in accordance with the format of the BWR ISTS, NUREG-1433, Revision 1. Any technical changes to this requirement will be addressed in the Discussion of Changes for ITS: 3.6.2.2.

A.7 CTS 4.5.C.2.b requires periodic verification that the specified conditions of CTS 3/4.5.C.2 Applicability footnote (a) 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 ISTS, NUREG-1433, 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, the CTS 4.5.C.2.b requirement for footnote (a) to be satisfied 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.

TECHNICAL CHANGES - MORE RESTRICTIVE

M.1 CTS 4.5.B requires that each LPCI pump develop the required flow when tested pursuant to Specification 4.0.E. ITS SR 3.5.2.4 also requires the surveillance to be performed, however, explicit values of flow (4500 gpm) and system head corresponding to reactor pressure (20 psig) are specified. Since explicit values are provided this change is considered more restrictive.

M.2 The allowance in CTS 3/4.5.C.2 footnote (a) and CTS 3.5.C Action 2 footnote (a) to not require the suppression pool to be OPERABLE during cavity flooding have 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 relating to system OPERABILITY in CTS 3/4.5.B (what constitutes an OPERABLE ECCS subsystem), and CTS 3.5.C.2 (reference for suppression chamber level) are proposed to be relocated to the Bases. ITS 3.5.2 will continue to require two ECCS subsystems to be OPERABLE and suppression chamber level to be maintained. 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 description in CTS 3.5.B footnote (a), that the LPCI may be considered OPERABLE if capable of being manually realigned via "remote or local" methods is proposed to be relocated to the Bases. The detail is not necessary since proposed SR 3.5.2.3 will require that the LPCI system valves aligned for decay heat removal have the capability to be manually realigned. As such, this 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 CTS 4.5.B requires ECCS to be demonstrated OPERABLE per Surveillance Requirement 4.5.A. This includes the actual or simulated automatic initiation test associated with the CS and LPCI subsystems which is currently required to be performed at an 18 month Frequency specified in CTS 4.5.A.3.a (Although HPCI is included in CTS 4.5.A.3.a it is not applicable to MODES 4 and 5 since it is not required to be OPERABLE). The Frequency for performing CTS 4.5.A.3.a during shutdown (proposed SR 3.5.2.5) has been extended from 18 months to 24 months. The ECCS system functional tests, CTS 4.5.A.3.a (proposed SR 3.5.2.5) ensure that a system initiation signal (actual or simulated) to the automatic initiation logic of CS and LPCI will cause the 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 proposed change will allow this Surveillance to extend its 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.B 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.B 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. During MODE 4 and 5 operations, two low pressure ECCS injection/spray subsystems are required to be OPERABLE. Based on engineering judgement only one low pressure ECCS injection/spray subsystem is necessary to maintain adequate reactor vessel water level in the event of an inadvertent vessel draindown. Therefore, by requiring two ECCS injection/spray subsystems to be OPERABLE adequate redundancy is provided. Each of the ECCS injection/spray systems are tested every three months according to the ASME Section XI inservice testing program (proposed SR 3.5.2.4) 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, and 3.5.2.3 are also performed more frequently to ensure the ECCS subsystems are available to perform their required functions. Furthermore, as stated in the NRC Safety Evaluation Report (dated August 2, 1993) relating to extension of the Peach Bottom Atomic Power Station, Unit Numbers 2 and 3 surveillance intervals from 18 to 24 months:

TECHNICAL CHANGES - LESS RESTRICTIVE

LD.1 (cont'd)

"Industry reliability studies for boiling water reactors (BWRs) prepared by the BWR Owners Group (NEDC-30936P) show that the overall safety systems' reliabilities are not dominated by the reliabilities of the logic system, but by that of the mechanical components, (e.g., pumps and valves), which are consequently tested on a more frequent basis. Since the probability of a relay or contact failure is small relative to the probability of mechanical component failure, increasing the logic system functional test interval represents no significant change in the overall safety system unavailability."

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.B Action 2 to suspend CORE ALTERATIONS when both ECCS subsystems are inoperable and the requirement of CTS 3.5.C Action 2 to suspend CORE ALTERATIONS when the suppression pool 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, 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

One of the provisions of CTS 3.5.C.2 that allows the suppression pool to be drained is that no operations are performed that have a potential for draining the reactor vessel (OPDRVs). CTS 3.5.C Action 2 requires suspension of OPDRVs if the suppression pool water level is not within limits or if the suppression pool is drained and the requirements of CTS 3.5.C.2 are not met. However, for the requirements of CTS 3.5.C.2 to be met OPDRVs must be suspended. Therefore, CTS 3.5.C.2 does not allow OPRDVs when the source of water is the condensate storage tank (known as the Contaminated Condensate Storage Tank (CCST) in

TECHNICAL CHANGES - LESS RESTRICTIVE

L.2 (cont'd) the ITS). CTS 3.5.B Action 1 allows OPDRVS to be performed for up to 4 hours with one required ECCS subsystem inoperable. In CTS 3.5.B no restrictions apply regarding water sources and the available source may be from either the suppression pool or the CCST. ITS 3.5.2 relaxes the limitation in CTS 3.5.C if the water source is only available from the CCST and OPDRVs are in progress. If OPDRVs are in progress only one ECCS subsystem is allowed to credit the CCST as indicated in proposed Note to SR 3.5.2.1.b, therefore, one ECCS subsystem must be declared inoperable. This is necessary since the available volume is limited. This will therefore limit the time that OPDRVs can be performed, since an ECCS subsystem must be declared inoperable and ITS 3.5.2 Required Action A.1 only provides 4 hours to restore the inoperable ECCS subsystem to OPERABLE status prior to suspending OPDRVs. Therefore, when credit is being taken for the CCST and the suppression pool level is not within limits operations must be in accordance with ITS 3.5.2 ACTIONS A and B, where the Required Action of Condition B precludes OPDRVs (note that Condition B applies 4 hours after Condition A, i.e., one ECCS subsystem inoperable, is entered). This change is considered acceptable given the remaining OPERABLE ECCS subsystem and the low probability of a reactor vessel drain down event during this time period.

L.3 The CTS 3.5.C.2.b requirement to "lock" the reactor mode switch in Shutdown or Refuel when the suppression pool is not within the required limit or is drained, and the CTS 3.5.C Action 2 requirement to "lock" the reactor mode switch in shutdown when the suppression pool water level is not within the required limit or is drained and the CTS 3.5.C.2 requirements not met, are 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 Refuel and Shutdown 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 or Refuel is proposed to be deleted from Technical Specifications.

L.4 CTS 4.5.C.2.b, the verification that the requirements in CTS 3.5.C.2 are satisfied every 12 hours when the suppression chamber water level limit is not met, has been modified to only require the Surveillances to be verified at the current specified frequencies not at this 12 hour frequency. CTS 3.5.C.2

TECHNICAL CHANGES - LESS RESTRICTIVE

L.4 (cont'd) specifies that no operations are performed that have a potential for draining the reactor vessel (revised as discussed in Discussion of Change L.2), the reactor mode switch is locked in the shutdown or Refuel position (deleted as discussed in Discussion of Change L.3), the condensate storage tank contains a specified volume of water (the 12 hour Frequency will be retained as indicated in proposed SR 3.5.2.1.b), and the ECCS are OPERABLE per Specification 3.5.B.

In the ITS, the requirements of 3/4.5.C and 3/4.5.B are incorporated in one Specification (ITS 3.5.2) and only the normal Surveillance Frequencies are proposed. This change is based on the fact that it is overly conservative to assume that systems or components are inoperable when a surveillance has not been performed. The opposite is in fact the case, the vast majority of surveillances demonstrate that systems or components in fact are operable. Therefore, even with low suppression pool level, the normal frequencies (e.g., LPCI testing in accordance with the Inservice Testing Frequency) are considered sufficient to ensure the OPERABILITY of the systems and that the parameters are within limits.

L.5 The condensate storage tank (known as the contaminated condensate storage tank in the ITS) water level requirement in CTS 3.5.B.1.a.2), CTS 3.5.B.2.b.2) and CTS 3.5.C.2.c for MODE 4 and 5 is proposed to be decreased from 140,000 available gallons to 50,000 available gallons (7.5 ft from the bottom of the tank) in ITS SR 3.5.2.1.b. The new water level is based on ensuring adequate net positive suction head (NPSH) and vortex prevention for all of the ECCS pumps, and provides 50,000 gallons of water for a recirculation/makeup volume. These three considerations (NPSH, vortexing, and recirculation/makeup volume) are described in the Bases as the reason for the level requirement. The proposed water level requirement will ensure there is a sufficient volume of water available for more than ten minutes with one ECCS pump operating at the required flow rate. This will provide time for the operators to obtain additional water supply for the contaminated condensate storage tank or obtain an alternate makeup source.

RELOCATED SPECIFICATIONS

None

3.5 - LIMITING CONDITIONS FOR OPERATION

D. Reactor Core Isolation Cooling System

LCO 3.5.3 The reactor core isolation cooling (RCIC) system shall be OPERABLE with an OPERABLE flow path capable of automatically taking suction from the suppression chamber and transferring the water to the reactor pressure vessel. LA.1

APPLICABILITY:

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

ACTION:

ACTION A — With the RCIC system inoperable, operation may continue provided the HPCI system is OPERABLE; restore the RCIC system to OPERABLE status within 14 days or be in

ACTION B — at least HOT SHUTDOWN within the next 12 hours and reduce reactor steam dome pressure to ≤150 psig within the following 24 hours.

4.5 - SURVEILLANCE REQUIREMENTS

D. Reactor Core Isolation Cooling System

The RCIC system shall be demonstrated OPERABLE:

1. At least once per 31 days by:

SR 3.5.3.1 a. Verifying that the system piping from the pump discharge valve to the system isolation valve is filled with water.

SR 3.5.3.2 b. Verifying that each valve, manual, power operated or automatic in the flow path that is not locked, sealed or otherwise secured in position, is in its correct position.

c. Verifying that the pump flow controller is in the correct position. LA.2

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

SR 3.5.3.3

3. At least once per 18 months by: 24 LD.1

SR 3.5.3.5 a. Verifying the RCIC system actuates on an actual or simulated automatic initiation signal. Actual injection of coolant into the reactor vessel may be excluded.

SR 3.5.3.5 NOTE

SR 3.5.3.3 NOTE

a The provisions of Specification 4.0.D are not applicable provided the surveillance is performed within 12 hours after reactor steam pressure and flow is adequate to perform the test. A.2

3.5 - LIMITING CONDITIONS FOR OPERATION

4.5 - SURVEILLANCE REQUIREMENTS

SR 3.5.3.4

b. Verifying that the system will develop a flow of ≥400 gpm against a system head corresponding to reactor vessel pressure, when steam is supplied to the turbine at a pressure between 150 and 180 psig[a]. A2 LA.3

c. ~~Verifying that the suction for the RCIC system is automatically transferred from the condensate storage tank to the suppression pool on a condensate storage tank water level - low signal and on a suppression pool water level - high signal.~~ LA.2

SR 3.5.3.4 NOTE

a The provisions of Specification 4.0.D are not applicable provided the surveillance is performed within 12 hours after reactor steam pressure and flow is adequate to perform the test. A.2

DISCUSSION OF CHANGES
ITS: 3.5.3 - RCIC SYSTEM

ADMINISTRATIVE

A.1 In the conversion of the Quad Cities 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-1433, Rev. 1 (i.e., the Improved Standard Technical Specifications (ISTS)).

A.2 CTS 4.5.D.2 footnote (a) and CTS 4.5.D.3.b footnote (a) 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.

TECHNICAL CHANGES - MORE RESTRICTIVE

None

TECHNICAL CHANGES - LESS RESTRICTIVE

"Generic"

LA.1 The details of CTS 3.5.D 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 4.5.D.1.c and 4.5.D.3.c relating to methods for performing Surveillances (i.e., verifying that the RCIC pump controller is in the correct position and verifying the RCIC suction is automatically transferred from the condensate storage tank, known as the contaminated condensate storage tank in the ITS, to the suppression pool on the proper signal) 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

TECHNICAL CHANGES - LESS RESTRICTIVE

LA.2 (cont'd) 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 4.5.D.3.b requires verifying RCIC System flow when steam pressure is between 150 and 180 psig. The minimum pressure for the test (150 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 (a) (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.

LD.1 The Frequencies for performing CTS 4.5.D.3.a, 4.5.D.3.b and 4.5.D.3.c (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 system 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.B 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.B 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

TECHNICAL CHANGES - LESS RESTRICTIVE

LD.1 (cont'd) 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 HPCI, and Technical Specifications do not permit HPCI 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. Furthermore, as stated in the NRC Safety Evaluation Report (dated August 2, 1993) relating to extension of the Peach Bottom Atomic Power Station, Unit Numbers 2 and 3 surveillance intervals from 18 to 24 months:

> "Industry reliability studies for boiling water reactors (BWRs) prepared by the BWR Owners Group (NEDC-30936P) show that the overall safety systems' reliabilities are not dominated by the reliabilities of the logic system, but by that of the mechanical components, (e.g., pumps and valves), which are consequently tested on a more frequent basis. Since the probability of a relay or contact failure is small relative to the probability of mechanical component failure, increasing the logic system functional test interval represents no significant change in the overall safety system unavailability."

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 4.5.D.3.a, 4.5.D.3.b and 4.5.D.3.c 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.

"Specific"

None

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 3/4.5-3) have been completely replaced by revised Bases that reflect the format and applicable content of the Quad Cities 1 and 2 ITS Section 3.5, consistent with the BWR ISTS, NUREG-1433, Rev. 1. The revised Bases are as shown in the Quad Cities 1 and 2 ITS Bases.

All Changes are [2] unless otherwise indicated

⟨CTS⟩ 3.5 EMERGENCY CORE COOLING SYSTEMS (ECCS) AND REACTOR CORE ISOLATION COOLING (RCIC) SYSTEM

3.5.1 ECCS—Operating

⟨3.5.A⟩ LCO 3.5.1 Each ECCS injection/spray subsystem and the Automatic Depressurization System (ADS) function of ~~{seven}~~ four safety/relief valves shall be OPERABLE. [1]

⟨Appl 3.5.A⟩ APPLICABILITY: MODE 1,
MODES 2 and 3, except high pressure coolant injection (HPCI) and ADS valves are not required to be OPERABLE with reactor steam dome pressure ≤ ~~{150}~~ psig. [1]

Insert ACTIONS A, B, C and D

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
⟨3.5.A Act 1.a⟩ ⟨3.5.A Act 2.a⟩ ⟨3.5.A Act 2.b⟩ E ~~B~~. Required Action and associated Completion Time of Condition A not met. (B, C or D)	~~B~~.1 Be in MODE 3. (E) AND ~~B~~.2 Be in MODE 4. (E)	12 hours 36 hours
⟨3.5.A Act 3⟩ F ~~C~~. HPCI System inoperable.	~~C~~.1 Verify by administrative means RCIC System is OPERABLE. (F) AND ~~C~~.2 Restore HPCI System to OPERABLE status. (F)	~~1 hour~~ Immediately [11] 14 days

(continued)

<CTS>

2 INSERT ACTIONS A, B, C, & D

<3.5.A Action 2.a>	A. One Low Pressure Coolant Injection (LPCI) pump inoperable.	A.1	Restore LPCI pump to OPERABLE status.	30 days
<3.5.A Action 1.a>	B. One LPCI subsystem inoperable for reasons other than Condition A. OR One Core Spray subsystem inoperable.	B.1	Restore low pressure ECCS injection/spray subsystem to OPERABLE status.	7 days
<3.5.A Act 2.b>	C. One LPCI pump in each subsystem inoperable.	C.1	Restore one LPCI pump to OPERABLE status.	7 days
<3.5.A Act 2.b> <DOCM.1>	D. Two LPCI subsystems inoperable for reasons other than Condition C.	D.1	Restore one LPCI subsystem to OPERABLE status.	72 hours

All changes are [2] unless otherwise indicated

⟨CTS⟩

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
TSTF-318 G D. HPCI System inoperable. C [14] AND or Condition A entered One low pressure ECCS injection/spray subsystem is inoperable. ⟨DOC L.3⟩ [3]	D.1 Restore HPCI System to OPERABLE status. G OR D.2 Restore low pressure ECCS injection/spray subsystem to OPERABLE status. (5) TSTF-318	72 hours 72 hours
⟨3.5.A Act 4.a⟩ H E. One required ADS valve inoperable.	H E.1 Restore ADS valve to OPERABLE status.	14 days
[2] TSTF-318 change not adopted 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. [3]	72 hours 72 hours [13]
⟨3.5.A Act 3⟩ I G. Two or more required ADS valves inoperable. ⟨3.5.A Act 4.a⟩ ⟨3.5.A Act 4.b⟩ [4] OR Required Action and associated Completion Time of Condition D, F, G, or H not met. G H	G.1 Be in MODE 3. AND I G.2 Reduce reactor steam dome pressure to ≤ ~~150~~ psig. [1]	12 hours 36 hours

(continued)

All changes are [2] unless otherwise indicated

⟨CTS⟩

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
⟨3.5.A Act 1.b⟩ (J) Two or more low pressure ECCS injection/spray subsystems inoperable. (for reasons other than Condition C or D) OR HPCI System and one or more required ADS valves inoperable. [3]	(J) .1 Enter LCO 3.0.3.	Immediately

OR
One or more low pressure ECCS injection/spray subsystems inoperable and one or more required ADS valves inoperable [13]

⟨CTS⟩ SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.5.1.1 ⟨4.5.A.1.a.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 ⟨4.5.A.1.a.2⟩	----------NOTE---------- Low pressure coolant injection (LPCI) subsystems may be considered OPERABLE during alignment and operation for decay heat removal with reactor steam dome pressure less than the Residual Heat Removal (RHR) cut-in permissive pressure in MODE 3, if capable of being manually realigned and not otherwise inoperable. ----------	1
	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 [air supply header] pressure is ≥ [90] psig.	31 days 5
SR 3.5.1.4	Verify the [RHR] System cross tie valve[s] [is] closed and power is removed from the valve operator[s].	31 days 6
⟨DOC M.2⟩ SR 3.5.1.5	Verify each LPCI inverter output voltage is ≥ [570] V and ≤ [630] V while supplying the respective bus. to the LPCI swing	31 days 1

correct breaker alignment

(continued)

⟨CTS⟩ SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE	FREQUENCY
SR 3.5.1.4 ⟨DOC M.2⟩ (4) [6] ~~NOTE~~ ~~Not required to be performed if performed within the previous 31 days.~~ Verify each recirculation pump discharge valve ~~{and bypass valve}~~ cycles through one complete cycle of full travel {or is de-energized in the closed position}.	[12] [6] In accordance with Inservice Testing Program ~~Once each startup prior to exceeding 25% RTP~~
SR 3.5.1.5 ⟨4.5.A.2.a⟩ ⟨4.5.A.2.b⟩ (5) [6] Verify the following ECCS pumps develop the specified flow rate {against a ~~system head~~ test line pressure corresponding to the specified reactor pressure}. [1] SYSTEM / FLOW RATE (TEST LINE PRESSURE) / NO. OF PUMPS / {~~SYSTEM HEAD~~ CORRESPONDING TO A REACTOR PRESSURE OF} Core Spray: ≥ {~~4250~~ 4500} gpm, {1}, ≥ {~~113~~ 90} psig LPCI: ≥ {~~17,000~~ 9000} gpm, {2}, ≥ {20} psig	[In accordance with the Inservice Testing Program ~~or 92 days~~] [1] [1]
SR 3.5.1.6 ⟨4.5.A.2.c⟩ (6) [6] ~~NOTE~~ ~~Not required to be performed until 12 hours after reactor steam pressure and flow are adequate to perform the test.~~ [1] Verify, with {reactor pressure} ≤ {~~1020~~ 1005} and ≥ {920} psig, the HPCI pump can develop a flow rate ≥ {~~4250~~ 5000} gpm {against a system head corresponding to reactor pressure}. [1]	In accordance with the Inservice Testing Program [8] ~~92 days~~

(continued)

<CTS> SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
<4.5.A.3.b> SR 3.5.1.~~9~~ 7 6	----------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 HPCI pump can develop a flow rate ≥ ~~[4250]~~ 5000 gpm {against a system head corresponding to reactor pressure}.	~~18~~ 24 months 180 9 1
<4.5.A.3.a> SR 3.5.1.~~10~~ 8 6	----------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.A.4.a> SR 3.5.1.~~11~~ 9 6	----------NOTE---------- Valve actuation may be excluded. ---------- Verify the ADS actuates on an actual or simulated automatic initiation signal.	~~[18]~~ 24 months 1
<4.5.A.4.b> SR 3.5.1.~~12~~ 10 6	----------NOTE---------- Not required to be performed until 12 hours after reactor steam pressure and flow are adequate to perform the test. ---------- Verify each required ADS valve opens when manually actuated.	~~18~~ 24 months ~~[on a STAGGERED TEST BASIS for each valve solenoid]~~ 3 9 6

<DOC M,2>. Insert SR 3.5.1.11 10

<CTS>

10

INSERT SR 3.5.1.12

<DOC M.2>

SR 3.5.1.11	Verify automatic transfer capability of the LPCI swing bus power supply from the normal source to the backup source.	24 months

JUSTIFICATION FOR DEVIATIONS FROM NUREG-1433, 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. Three new ACTIONS (ITS 3.5.1 ACTIONS A, C, and D) are added to BWR ISTS, NUREG-1433, Revision 1 Specification 3.5.1 to allow one LPCI pump to be inoperable for 30 days, one LPCI pump in each subsystem to be inoperable for 7 days, and two LPCI subsystems inoperable for 72 hours. In addition, ISTS 3.5.1 Condition A (ITS 3.5.1 Condition B) has been modified to reflect the inclusion of ACTION A. These Conditions are provided in the current Technical Specifications. Due to these additions, subsequent Conditions and Required Actions have been modified and renumbered as required. ITS 3.5.1 ACTION C is similar to the change in ISTS 3.5.1 ACTION A from TSTF-318, Rev. 0. The other changes of TSTF-318, Rev. 0, are not incorporated in ITS 3.5.1 since they are not supported by current analyses.

3. The word "required" has been added consistent with its use throughout the ITS (only four of the five installed ADS valves are required).

4. Change made to be consistent with the Writer's Guide.

5. ISTS SR 3.5.1.3 has been deleted to reflect the plant design. The pneumatic operated Target Rock valve is not credited for the ADS function. Only three of the four relief valves are credited. Therefore, the requirement to verify the ADS supply header pressure required for pneumatic operation is not required. Subsequent SRs have been renumbered as required.

6. The brackets have been removed and the information deleted since it does not apply. Subsequent SRs have been renumbered as required.

7. LPCI injection and recirculation pump discharge valves are supplied by the LPCI swing bus. Proper breaker alignment is necessary to help ensure OPERABILITY of these valves. Therefore, ISTS SR 3.5.1.5 (ITS SR 3.5.1.3), the LPCI inverter surveillance, has been revised to reflect the appropriate requirement for the Quad Cities 1 and 2 design.

8. The Frequency of ISTS SR 3.5.1.8 (ITS SR 3.5.1.6), the HPCI high pressure flow test Surveillance Frequency, has been changed from 92 days to "In accordance with the Inservice Testing Program" consistent with current Technical Specifications.

9. The Frequency of ISTS SR 3.5.1.9 (ITS SR 3.5.1.7), the HPCI flow test at low pressure, and ISTS SR 3.5.1.12 (ITS SR 3.5.1.10) has been extended from 18 to 24 months. See the Discussion of Changes for ITS 3.5.1 for further justification of this change.

10. ITS SR 3.5.1.11 has been added to require the verification of the automatic transfer capability of the LPCI swing bus power supply from the normal source to the backup source every 24 months. This added requirement is necessary to help ensure the safety analysis assumptions are satisfied.

11. The time allowed to complete ITS 3.5.1, Required Action F.1 has been changed from 1 hour to Immediately. Due to the mechanics of how Completion Times work, the 1 hour allowance can probably never be used. For example, if HPCI is inoperable, ITS 3.5.1, Condition F is entered, and the 1 hour verification of Required Action F.1 is performed. If RCIC is not inoperable at this time, the Required Action is met. However, since the Completion Time starts upon entry into this Condition, if RCIC becomes inoperable greater than 1 hour later, the 1 hour time in the HPCI ACTION has already expired. Thus a unit shutdown would be required immediately upon discovery of RCIC being inoperable, even though the RCIC Required Action (ITS 3.5.3, Required Action A.1) appears to allow 1 hour to verify HPCI OPERABILITY. To avoid this confusion, the original time allowed by BWR ISTS, NUREG-1433, Revision 0, and the current Quad Cities 1 and 2 Technical Specifications has been used. This change is also consistent with TSTF-301, Rev.0.

12. The Frequency of ISTS 3.5.1.6 (ITS SR 3.5.1.4), the recirculation pump discharge valve stroke test Surveillance Frequency, has been changed from "Once each startup prior to exceeding 25% RTP" to "In accordance with the Inservice Testing Program" consistent with the current licensing basis testing requirements. In addition, the Note for ISTS SR 3.5.1.6 (ITS SR 3.5.1.4) has been deleted consistent with the current licensing basis testing requirements.

13. ISTS 3.5.1 ACTION F has been deleted and the condition included in proposed ACTION J to reflect the current licensing basis. Subsequent ACTIONS have been renumbered as required.

14. ISTS 3.5.1 ACTION D (ITS 3.5.1 ACTION G) has been revised to be consistent with TSTF-318. However, since ITS 3.5.1 ACTIONS A, B, C, and D have been added as described in comment 2, reference to Condition A has been changed to Condition C.

⟨CTS⟩

3.5 EMERGENCY CORE COOLING SYSTEMS (ECCS) AND REACTOR CORE ISOLATION COOLING (RCIC) SYSTEM

3.5.2 ECCS—Shutdown

⟨3.5.B⟩
⟨3.5.C.2.d⟩

LCO 3.5.2 Two low pressure ECCS injection/spray subsystems shall be OPERABLE.

⟨Appl. 3.5.B⟩
⟨Appl 3.5.C⟩

APPLICABILITY: MODE 4,
MODE 5, except with the spent fuel storage pool gates removed and water level ≥ [23 ft] over the top of the reactor pressure vessel flange.] 1

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
⟨3.5.B Act 1⟩ A. One required ECCS injection/spray subsystem inoperable.	A.1 Restore required ECCS injection/spray subsystem to OPERABLE status.	4 hours
3.5.B Act 1 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.B Act 2 3.5.C Act 2 C. Two required ECCS injection/spray subsystems inoperable.	C.1 Initiate action to suspend OPDRVs.	Immediately
	AND	
	C.2 Restore one ECCS injection/spray subsystem to OPERABLE status.	4 hours

(continued)

⟨CTS⟩

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
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

⟨3.5.B Act 2⟩

3.5.C Act 2

1

1

1

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.5.2.1	Verify, for each required low pressure coolant injection (LPCI) subsystem, the suppression pool water level is ≥ [12 ft 2 inches].	12 hours

2

(continued)

<CTS>

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
<4.5.C.2> <3.5.C.2> <3.5.C.2.c> <3.5.B.1.a.2> <3.5.B.2.b.2> SR 3.5.2.~~2~~ 1	Verify, for each required ~~core spray (CS)~~ ECCS injection/spray subsystem, the: a. Suppression pool water level is ≥ ~~[12 ft 2 inches]~~ 8.5 ft; or b. ----------NOTE---------- Only one required CS subsystem may take credit for this option during OPDRVs. ---------- Contaminated Condensate storage tank water level is ≥ ~~[12 ft]~~ ≥7.5 feet.	12 hours 2 1
<4.5.B> SR 3.5.2.~~3~~ 2	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.B> SR 3.5.2.~~4~~ 3	----------NOTE---------- One LPCI subsystem may be considered OPERABLE during alignment and operation for decay heat removal if capable of being manually realigned and not otherwise inoperable. ---------- 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.B⟩				
⟨4.5.B.2⟩ SR 3.5.2.~~5~~ 4 [2]	Verify each required ECCS pump develops the specified flow rate ~~against a system head corresponding to the specified reactor pressure~~. [1] test line pressure			In accordance with the Inservice Testing Program ~~or 92 days~~ [1]
	SYSTEM	FLOW RATE	NO. OF PUMPS	~~SYSTEM HEAD~~ CORRESPONDING TO A REACTOR PRESSURE OF ~~:~~ [1]
	CS	≥ ~~[4250]~~ 4500 gpm	~~{1}~~	≥ ~~{113}~~ 96 psig
	LPCI	≥ ~~[7700]~~ 4500 gpm	~~{1}~~	≥ ~~{20}~~ psig
⟨4.5.B⟩ SR 3.5.2.~~6~~ 5 [2]	----------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]

JUSTIFICATION FOR DEVIATIONS FROM NUREG-1433, REVISION 1
ITS: 3.5.2 - ECCS — SHUTDOWN

1. The brackets have been removed and the proper plant specific information/value has been provided.

2. The requirements for suppression pool water level and contaminated condensate storage tank levels are applicable to both CS and LPCI subsystems. Therefore, ISTS SR 3.5.2.1 is deleted and the requirement to verify suppression pool water level for the LPCI subsystem is addressed in ISTS 3.5.2.2 (ITS 3.5.2.1). Subsequent SRs are renumbered, as required.

3. Changes have been made (additions, deletions, and/or changes to NUREG) to reflect the plant specific nomenclature, number, reference, system description, analysis description, or licensing basis description.

4. 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

<3.5.D>

LCO 3.5.3 The RCIC System shall be OPERABLE.

APPLICABILITY: MODE 1,
MODES 2 and 3 with reactor steam dome pressure > ~~150~~ psig. [1]

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. RCIC System inoperable.	A.1 Verify by administrative means High Pressure Coolant Injection System is OPERABLE.	~~1 hour~~ Immediately [2] JSTF-301
	AND	
	A.2 Restore RCIC System to OPERABLE status.	14 days
B. Required Action and associated Completion Time not met.	B.1 Be in MODE 3.	12 hours
	AND	
	B.2 Reduce reactor steam dome pressure to ≤ ~~150~~ psig. [1]	36 hours

<3.5.D Act>

<3.5.D Act>

⟨CTS⟩ SURVEILLANCE REQUIREMENTS

	SURVEILLANCE		FREQUENCY
⟨4.5.D.1.a⟩	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
⟨4.5.D.1.b⟩	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
⟨4.5.D.2⟩	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]~~ 1005 psig and ≥ ~~[920]~~ psig, the RCIC pump can develop a flow rate ≥ ~~[400]~~ gpm ~~[~~against a system head corresponding to reactor pressure~~]~~.	92 days [1]
⟨4.5.D.3.b⟩	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~~]~~ ≤ ~~[185]~~ 180 psig, the RCIC pump can develop a flow rate ≥ ~~[400]~~ gpm ~~[~~against a system head corresponding to reactor pressure~~]~~.	~~[18]~~ 24 months [1]

(continued)

<CTS>

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.5.3.5 <4.5.D.3>	----------NOTE---------- Vessel injection may be excluded. ---------- Verify the RCIC System actuates on an actual or simulated automatic initiation signal.	24 [~~[18]~~] months [1]

JUSTIFICATION FOR DEVIATIONS FROM NUREG-1433, REVISION 1
ITS: 3.5.3 - RCIC SYSTEM

1. The brackets have been removed and the proper plant specific information/value has been provided.

2. The time allowed to complete ITS 3.5.3, Required Action A.1 has been changed from 1 hour to Immediately. Due to the mechanics of how Completion Times work, the 1 hour allowance can probably never be used. For example, if RCIC is inoperable, ITS 3.5.3, Condition A is entered, and the 1 hour verification of Required Action A.1 is performed. If HPCI is not inoperable at this time, the Required Action is met. However, since the Completion Time starts upon entry into this Condition, if HPCI becomes inoperable greater than 1 hour later, the 1 hour time in the RCIC ACTION has already expired. Thus a unit shutdown would be required immediately upon discovery of HPCI being inoperable, even though the HPCI Required Action (ITS 3.5.1, Required Action F.1) appears to allow 1 hour to verify RCIC OPERABILITY. To avoid this confusion, the original time allowed by BWR ISTS, NUREG-1433, Revision 0 and the current Quad Cities 1 and 2 Technical Specifications has been used. This change is also consistent with TSTF-301, Rev. 0.

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 consists of the High Pressure Coolant Injection (HPCI) System, the Core Spray (CS) System, the low pressure coolant injection (LPCI) mode of the Residual Heat Removal (RHR) System, and 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 contaminated condensate storage tank (CST), it is capable of providing a source of water for the HPCI, LPCI and CS systems.

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. Although the system is initiated, ADS action is delayed, allowing the operator to interrupt the timed sequence if the system is not needed. The HPCI pump discharge pressure almost immediately exceeds that of the RCS, and the pump injects coolant into the vessel to cool the core. If the break is small, the HPCI System will maintain coolant inventory as well as vessel level while the RCS is still pressurized. If HPCI fails, it is backed up by ADS in combination with LPCI and CS. In this event, the ADS timed sequence would be allowed to time out and open the selected safety and safety/relief valve relief valves (S/RVs) depressurizing the RCS, thus allowing the LPCI and CS to overcome RCS pressure and inject coolant into the vessel. If the break is large, RCS pressure initially drops rapidly and the LPCI and CS 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 RHR Service Water System. Depending on the location and size of

(continued)

BASES

BACKGROUND (continued)

the break, portions of 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. ~~Although no credit is taken in the safety analysis for the RCIC System, it performs a similar function as HPCI, 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]

The combined operation of [3]

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 equipment.

The CS System (Ref. 1) is composed of two independent subsystems. Each subsystem consists of a motor driven pump, a spray sparger above the core, and piping and valves to transfer water from the suppression pool to the sparger. The CS System is designed to provide cooling to the reactor core when reactor pressure is low. Upon receipt of an initiation signal, the CS pumps in both subsystems are automatically started immediately when normal AC power is available and in approximately 13 seconds after emergency AC power is available. When the RPV pressure drops sufficiently, CS System 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 CS System without spraying water in the RPV.

LPCI is an independent operating mode of the RHR System. There are two LPCI subsystems (Ref. 2), each consisting of two motor driven pumps and piping and valves to transfer water from the suppression pool to the RPV via the selected ~~corresponding~~ recirculation loop. Insert BKGD-1 ~~The two LPCI subsystems can be interconnected via the RHR System cross tie valve; however, the cross tie valve is maintained closed with its power removed to prevent loss of both LPCI subsystems during a LOCA.~~ The LPCI subsystems are designed to provide core cooling at low RPV pressure. Upon receipt of an initiation signal, all four LPCI pumps are automatically started (B and C pumps immediately when normal AC power is available, and A and D pumps ~~approximately 10 seconds after~~ and in approximately 8 seconds after emergency AC power is available AC power is available). 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 selected recirculation loop. When the RPV pressure drops sufficiently, the LPCI flow to the RPV, via the

Immediately when normal AC power is available and in approximately 3 seconds after emergency

(continued)

1 INSERT BKGD-1

The LPCI System is equipped with a loop select logic that determines which, if any, of the recirculation loops has been broken and selects the non-broken loop for injection. If neither loop is determined to be broken, then "B" recirculation loop is selected for injection. The LPCI System cross-tie valves must be open to support OPERABILITY of both LPCI subsystems. Similarly, the LPCI swing bus is required to be energized to support both LPCI subsystems. Therefore, with the LPCI cross-tie valves not full open, or the LPCI swing bus not energized, both LPCI subsystems must be considered inoperable.

All changes are [1] unless otherwise indicated

BASES

selected each subsystem

BACKGROUND (continued)

~~corresponding~~ recirculation loop, begins. The water then enters the reactor through the jet pumps. Full flow test lines are provided for ~~the four~~ LPCI ~~pumps~~ to route water from the suppression pool, to allow testing of the LPCI pumps without injecting water into the RPV. These test lines also provide suppression pool cooling capability, as described in LCO 3.6.2.3, "RHR Suppression Pool Cooling."

and to

The HPCI System (Ref. 3) 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, where the coolant is distributed within the RPV through the feedwater sparger. Suction piping for the system is provided from the CST and the suppression pool. Pump suction for HPCI 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 if the suppression pool level is high, an automatic transfer to the suppression pool water source ensures a water supply for continuous operation of the HPCI System. The steam supply to the HPCI turbine is piped from a main steam line upstream of the associated inboard main steam isolation valve.

The HPCI System is designed to provide core cooling for a wide range of reactor pressures (~~162 psid to 1135 psid, vessel to pump suction~~). Upon receipt of an initiation signal, the HPCI turbine stop valve and turbine ~~control~~ valve open simultaneously and the turbine accelerates to a specified speed. As the HPCI flow increases, the turbine ~~governor~~ valve is automatically adjusted to maintain design flow. Exhaust steam from the HPCI turbine is discharged to the suppression pool. A full flow test line is provided to route water from and to the CST to allow testing of the HPCI System during normal operation without injecting water into the RPV.

150 psig to 1120 psig

steam supply

control

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. To ensure rapid delivery of water to the RPV and to minimize water hammer effects, all ECCS pump discharge lines are filled with water. The LPCI and CS System discharge lines are kept full of water using a "keep fill" system (jockey pump system). The HPCI System is normally aligned to the

or remain open

(continued)

All changes are [1] unless otherwise indicated

BASES

BACKGROUND (continued)

CST. The height of water in the CST is sufficient to maintain the piping full of water up to the first closed isolation valve. The relative height of the feedwater line connection for HPCI is such that the water in the feedwater lines keeps the remaining portion of the HPCI discharge line full of water. Therefore, HPCI does not require a "keep fill" system.

in the discharge piping

The ADS (Ref. 4) consists of 5 valves (4 relief valves and one S/RV) ~~7 of the 11 S/RVs~~. It is designed to provide depressurization of the RCS during a small break LOCA if HPCI 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 (CS and LPCI), so that these subsystems can provide coolant inventory makeup. Each of ~~the~~ S/RV~~s~~ used for automatic depressurization is equipped with one air accumulator and associated inlet check valves. The accumulator provides the pneumatic power to actuate the valve~~s~~. However, the S/RV is not credited in the safety analysis since qualification of the accumulator for this valve to perform the ADS function has not been demonstrated (Ref. 5).

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 Reference 8. The results of these analyses are also 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 a 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

(continued)

All changes are [1] unless otherwise indicated

BASES

APPLICABLE SAFETY ANALYSES (continued)

e. Adequate long term cooling capability is maintained.

The limiting single failures are discussed in Reference ~~11~~ 8. For a large discharge pipe break LOCA, failure of the LPCI valve on the unbroken recirculation loop is considered the most severe failure. For a small break LOCA, HPCI 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 provide the capability to adequately cool the core and prevent excessive fuel damage.

For GE fuel, the recirculation suction line break with 125 VDC battery failure

Insert ASA-1

Insert ASA-2

The ECCS satisfy Criterion 3 of the NRC Policy Statement 10 CFR 50.36(c)(2)(ii).

LCO

Each ECCS injection/spray subsystem and ~~seven~~ four ADS valves are required to be OPERABLE. The ECCS injection/spray subsystems are defined as the two CS subsystems, the two LPCI subsystems, and one HPCI System. The low pressure ECCS injection/spray subsystems are defined as the two CS subsystems and the two LPCI subsystems.

The S/RV can not be used to satisfy the ADS requirement.

With less than the required number of ECCS subsystems OPERABLE, the potential exists that during a limiting design basis LOCA concurrent with the worst case single failure, the limits specified in Reference 10 could be exceeded. All ECCS subsystems must therefore be OPERABLE to satisfy the single failure criterion required by Reference 10.

[3]

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. 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]

Alignment and operation for decay heat removal includes when the required LPCI pump is not operating or when the system is realigned from or to the RHR shutdown cooling mode.

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, when reactor steam dome pressure

(continued)

1 INSERT ASA-1

limiting break/failure combination. In addition, to the 125 VDC battery failure, HPCI was assumed to fail to be consistent with generic analyses performed for the BWR 3/4 design generic analysis. For Siemens fuel, the recirculation suction line break with a failure of the entire LPCI System is considered the most limiting break/failure combination.

1 INSERT ASA-2

In the analysis of events requiring ADS operation, it is assumed that only three of the five ADS valves operate. Therefore, four ADS valves are required to be OPERABLE to meet single failure criteria.

All changes are [4] unless otherwise indicated

BASES

APPLICABILITY (continued) is ≤ 150 psig, ADS and HPCI are not required to be OPERABLE because the low pressure ECCS subsystems can provide sufficient flow below this pressure. ECCS requirements for MODES 4 and 5 are specified in LCO 3.5.2, "ECCS—Shutdown."

ACTIONS

Insert Action A (B)

~~A.1~~ a LPCI subsystem is inoperable for reasons other than Condition A, or a CS.

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. 10) that evaluated the impact on ECCS availability, assuming 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). (1) [1]

low pressure ECCS injection/spray

Insert Actions C and D (E)

~~B.1 and B.2~~ any Required Action and associated Completion Time of Condition A, B, C or D is not met [3]

If ~~the inoperable low pressure ECCS subsystem cannot be restored to OPERABLE status 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 from full power conditions in an orderly manner and without challenging plant systems.

(F) ~~C.1 and C.2~~

If the HPCI System is inoperable and the RCIC System is verified to be OPERABLE, the HPCI 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

(continued)

4 INSERT ACTION A

A.1

If any one LPCI pump is inoperable, the inoperable pump must be restored to OPERABLE status within 30 days. In this Condition, the remaining OPERABLE pumps provide adequate core cooling during a LOCA. However, overall ECCS reliability is reduced, because a single failure in one of the remaining OPERABLE LPCI subsystems, concurrent with a LOCA, may result in the LPCI subsystems not being able to perform their intended safety function. The 30 day Completion Time is based on a reliability study cited in Reference 11 that evaluated the impact on ECCS availability, assuming 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 allowable repair times (i.e., Completion Times).

4 INSERT ACTIONS C and D

C.1

If one LPCI pump in each subsystem is inoperable, one LPCI pump must be restored to OPERABLE status within 7 days. In this Condition, the remaining OPERABLE ECCS subsystems provide adequate core cooling during a LOCA. However, overall ECCS reliability is reduced because a single failure in one of the remaining OPERABLE ECCS 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. 11) that evaluated the impact on ECCS availability, assuming 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).

D.1

If two LPCI subsystems are inoperable for reasons other than Condition C, one inoperable subsystem must be restored to OPERABLE status within 72 hours. In this Condition, the remaining OPERABLE CS subsystems provide adequate core cooling during a LOCA. However, overall ECCS reliability is reduced, because a single failure in one of the remaining CS subsystems, concurrent with a LOCA, may result in ECCS not being able to perform its intended safety function. The 72 hour Completion Time is based on a reliability study cited in Reference 11 that evaluated the impact on ECCS availability, assuming 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 allowable repair times (i.e., Completion Times).

BASES

ACTIONS

~~D.1 and D.2~~ (continued)

conjunction with ADS. Also, the RCIC System will automatically provide makeup water at most reactor operating pressures. Verification of RCIC OPERABILITY ~~within 1 hour~~ is therefore required when HPCI is inoperable. This may be performed as an administrative check by examining logs or other information to determine if RCIC is out of service for maintenance or other reasons. It does not mean to perform the Surveillances needed to demonstrate the OPERABILITY of the RCIC System. If the OPERABILITY of the RCIC System cannot be verified, however, Condition ~~D~~ must be immediately 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 a reliability study cited in Reference 1~~2~~ and has been found to be acceptable through operating experience.

~~E.1 and E.2~~

If any one low pressure ECCS injection/spray subsystem or one LPCI pump in both LPCI subsystems, is inoperable in addition to an inoperable HPCI System, the inoperable low pressure ECCS injection/spray subsystem or the HPCI System must be restored to OPERABLE status within 72 hours. In this Condition, adequate core cooling is ensured by the OPERABILITY of the ADS and the remaining low pressure ECCS subsystems. However, the overall ECCS reliability is significantly reduced 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 both a high pressure system (HPCI) and a low pressure subsystem are inoperable, a more restrictive Completion Time of 72 hours is required to restore either the HPCI System or the low pressure ECCS injection/spray subsystem to OPERABLE status. This Completion Time is based on a reliability study cited in Reference 1~~3~~ and has been found to be acceptable through operating experience.

~~F.1~~

The LCO requires ~~seven~~ four ADS valves to be OPERABLE in order to provide the ADS function. Reference 1~~4~~ contains the results

(continued)

BASES

ACTIONS

D.1 (continued)

of an analysis that evaluated the effect of one ADS valve being out of service. Per this analysis, operation of only six 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 cited in Reference 12 and has been found to be acceptable through operating experience.

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 HPCI and the remaining low pressure ECCS injection/spray subsystem. However, overall ECCS reliability is reduced because a single active component failure concurrent with a design basis LOCA could result in the minimum required ECCS equipment not being available. Since both a high pressure system (ADS) and a low pressure subsystem are inoperable, a more restrictive Completion Time of 72 hours is required to restore either the low pressure ECCS subsystem or the ADS valve to OPERABLE status. This Completion Time is based on a reliability study cited in Reference 12 and has been found to be acceptable through operating experience.

TSTF-318 changes not shown

G.1 and G.2

If any Required Action and associated Completion Time of Condition C, D, E, or F is 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.

(continued)

BASES

ACTIONS (continued)

J J 4

~~D~~.1

When multiple ECCS subsystems are inoperable, as stated in Condition ~~D~~, the plant is in a condition outside of the accident analyses. 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 HPCI System, CS System, and LPCI subsystems full of water ensures that the ECCS will perform properly, injecting its full capacity into the RCS upon demand. This will also prevent a water hammer following an ECCS initiation signal. One acceptable method of ensuring that the lines are full is to vent at the high points. The 31 day Frequency is based on the gradual nature of void buildup in the ECCS piping, the procedural controls governing system operation, and operating experience.

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 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 HPCI 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

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.5.1.2 (continued)

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 position would only affect a single subsystem. This Frequency has been shown to be acceptable through operating experience.

3

Insert SR 3.5.1.2 A

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.

3

Insert SR3.5.1.2B

3

SR 3.5.1.3

Verification every 31 days that ADS air supply header pressure is ≥ [90] psig ensures adequate air pressure for reliable ADS operation. The accumulator on each ADS valve provides pneumatic pressure for valve actuation. The design 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. 11). 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 ≥ [90] psig is provided by the ADS instrument air supply. The 31 day Frequency takes into consideration administrative controls over operation of the air system and alarms for low air pressure.

4

SR 3.5.1.4

Verification every 31 days that the RHR System cross tie valve is closed and power to its operator is disconnected ensures that each LPCI subsystem remains independent and a failure of the flow path in one subsystem will not affect the flow path of the other LPCI subsystem. Acceptable methods of removing power to the operator include de-energizing breaker control power or racking out or

4

(continued)

3 INSERT SR 3.5.1.2A

In MODE 3 with reactor steam dome pressure less than the actual RHR cut-in permissive pressure, the RHR System may be required to operate in the shutdown cooling mode to remove decay heat and sensible heat from the reactor. Therefore,

3 INSERT SR 3.5.1.2B

Alignment and operation for decay heat removal includes when the required RHR pump is not operating or when the system is being realigned from or to the RHR shutdown cooling mode. At the low pressures and decay heat loads associated with operation in MODE 3 with reactor steam dome pressure less than the RHR cut-in permissive pressure, a reduced complement of low pressure ECCS subsystems should provide the required core cooling, thereby allowing operation of RHR shutdown cooling, when necessary.

BASES

SURVEILLANCE REQUIREMENTS

SR 3.5.1.4 (continued)

removing the breaker. If the RHR System cross tie valve is open or power has not been removed from the valve operator, both LPCI subsystems must be considered inoperable. The 31 day Frequency has been found acceptable, considering that these valves are under strict administrative controls that will ensure the valves continue to remain closed with either control or motive power removed.

4

SR 3.5.1.6 3

of the Correct breaker alignment to the LPCI swing

4

Verification every 31 days that each LPCI inverter output has a voltage of ≥ [570] V and ≤ [630] V while supplying its respective bus demonstrates that the AC electrical power is available to ensure proper operation of the associated LPCI inboard injection and minimum flow valves and the recirculation pump discharge valve. Each inverter must be OPERABLE for the associated LPCI subsystem to be OPERABLE. The 31 day Frequency has been found acceptable based on engineering judgment and operating experience.

1 5 1

SR 3.5.1.7 4

5

Cycling the recirculation pump discharge (and bypass) valves through one complete cycle of full travel demonstrates that the valves are mechanically OPERABLE and will close when required. Upon initiation of an automatic LPCI subsystem injection signal, these valves are required to be closed to ensure full LPCI subsystem flow injection in the reactor via the recirculation jet pumps. De-energizing the valve in the closed position will also ensure the proper flow path for the LPCI subsystem. Acceptable methods of de-energizing the valve include de-energizing breaker control power, racking out the breaker or removing the breaker.

The specified Frequency is once during reactor startup before THERMAL POWER is > 25% RTP. However, this SR is modified by a Note that states the Surveillance is only required to be performed if the last performance was more than 31 days ago. Therefore, implementation of this Note requires this test to be performed during reactor startup before exceeding 25% RTP. Verification during reactor

4

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.5.1.~~7~~ (continued)

startup prior to reaching > 25% RTP is an exception to the normal Inservice Testing Program generic valve cycling Frequency of 92 days, but is considered acceptable due to the demonstrated reliability of these valves. If ~~the~~ valve is inoperable and in the open position, ~~the associated~~ LPCI subsystem must be declared inoperable.

SR 3.5.1.~~8~~, SR 3.5.1.~~9~~, and SR 3.5.1.~~10~~

The performance requirements of the low pressure 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 low pressure ECCS pump flow rates ensure that adequate core cooling is provided to satisfy the acceptance criteria of Reference 10. The pump flow rates are verified against a system head 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 a LOCA. These values ~~may be~~ established ~~during preoperational testing~~.

The flow tests for the HPCI System are performed at two different pressure ranges such that system capability to provide rated flow is tested at both the higher and lower operating ranges of the system. Additionally, adequate steam flow must be passing through the main turbine or turbine bypass valves to continue to control reactor pressure when the HPCI System diverts steam flow. Reactor steam pressure must be ≥ {920} psig to perform SR 3.5.1.~~8~~ and ≥ {150} psig to perform SR 3.5.1.~~9~~. Adequate steam flow is represented by {at least 1.75 turbine bypass valves open, or total steam flow ≥ 10^6 lb/hr}. Therefore, sufficient time is allowed after adequate pressure and flow are achieved to perform these tests. Reactor startup is allowed prior to performing the low pressure Surveillance test because the reactor pressure is low and the time allowed to satisfactorily perform the Surveillance test is short. The reactor pressure is allowed to be increased to normal

(continued)

3 INSERT SR 3.5.1.6

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.

All changes are [4] unless otherwise indicated

BASES

SURVEILLANCE REQUIREMENTS

SR 3.5.1.~~7~~ (5), SR 3.5.1.~~8~~ (6), and SR 3.5.1.~~9~~ (7) (continued)

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 HPCI is inoperable.

Therefore, SR 3.5.1.~~8~~ (6) and SR 3.5.1.~~9~~ (7) 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.

[3] Insert SR 3.5.1.7

The Frequency for SR 3.5.1.~~7~~ (5) and SR 3.5.1.~~8~~ (6) is in accordance with the Inservice Testing Program requirements. The ~~18~~ (24) month Frequency for SR 3.5.1.~~9~~ (7) is based on the need to perform the Surveillance under the conditions that apply ~~just prior to or~~ [3] during a 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 3.5.1.~~10~~ (8)

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 HPCI, CS, 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 low- positions. This SR also ensures that the HPCI 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 HPCI suction is automatically transferred from the CST to the suppression pool on high suppression pool water level or low CCST water level. [3] The LOGIC SYSTEM FUNCTIONAL TEST performed in LCO 3.3.5.1 overlaps this Surveillance to provide complete testing of the assumed safety function. (c) [1]

[1] The 18 month Frequency is based on the need to perform the Surveillance under the conditions that apply during a plant outage and the potential for an unplanned transient if the Surveillance ~~were~~ can be performed with the reactor at power. (3)

While this

(continued)

3 INSERT SR 3.5.1.7

The 12 hours allowed for performing the flow test after the required pressure and flow are reached is sufficient to achieve stable conditions for testing and provides reasonable time to complete the SRs.

All changes are [4] unless otherwise indicated

BASES

SURVEILLANCE REQUIREMENTS

SR 3.5.1.~~10~~ 8 (continued)

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.

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.~~11~~ 9

The ADS designated ~~S/RVs~~ valves 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.~~12~~ 10 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 ~~18~~ 24 month Frequency is based on the need to perform the 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.

This SR is modified by a Note that excludes valve actuation. [3] [1] This prevents an/RPV pressure/blowdown. Since the valves are individually tested in accordance with SR 3.5.1.10

SR 3.5.1.~~12~~ 10

A manual actuation of each ADS valve is performed to verify that the valve and solenoid are functioning properly and

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.5.1.~~12~~ 10 (continued)

that no blockage exists in the ~~SRV~~ valve discharge lines. This is demonstrated by the response of the turbine control or bypass valve or by a change in the measured 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 SR. Adequate pressure at which this SR is to be performed is ~~{920 psig}~~ 300 (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/h}~~ 2. Reactor startup is allowed prior to performing this SR 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 and provides adequate time to complete the Surveillance. SR 3.5.1.~~11~~ 9 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 Frequency of ~~18~~ 24 months on a ~~STAGGERED TEST BASIS~~ ensures that both solenoids for each ADS valve are alternately tested. The Frequency is based on the need to perform the 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~~ 24 month Frequency, which is based on the refueling cycle. Therefore, the Frequency was concluded to be acceptable from a reliability standpoint.

Insert SR 3.5.1.11

REFERENCES

1. FSAR, Section ~~{6.3.2.2.3}~~ 6.
2. FSAR, Section ~~{6.3.2.2.8}~~ 6.

(continued)

4 INSERT SR 3.5.1.11

The LPCI System injection valves and recirculation pump discharge valves are powered from the LPCI swing bus, which must be energized after a single failure, including loss of power from the normal source to the swing bus. Therefore, the automatic transfer capability from the normal power source to the backup power source must be verified to ensure the automatic capability to detect loss of normal power and initiate an automatic transfer to the swing bus backup power source. Verification of this capability every 24 months ensures that AC electrical power is available for proper operation of the associated LPCI injection valves and recirculation pump valves. The swing bus automatic transfer scheme must be OPERABLE for both LPCI subsystems to be OPERABLE. The Frequency of 24 months is based on the need to perform the 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.

All changes are [1] unless otherwise indicated

BASES

REFERENCES (continued)

3. FSAR, Section ~~{6.3.2.2}~~.
4. FSAR, Section ~~{6.3.2.2}~~.
5. ~~FSAR, Section [15.2.8].~~
6. FSAR, Section ~~{15.6.4}~~.
7. FSAR, Section ~~{15.6.5}~~.
8. 10 CFR 50, Appendix K.
9. FSAR, Section ~~{6.3.3}~~.
10. 10 CFR 50.46.
11. ~~FSAR, Section [7.3.1.2.2].~~
12. Memorandum from R.L. Baer (NRC) to V. Stello, Jr. (NRC), "Recommended Interim Revisions to LCOs for ECCS Components," December 1, 1975.
13. FSAR, Section ~~{6.3.3.3}~~.

5. Letter from J.A. Zwolinski (NRC) to D.C. Farrar (Commonwealth Edison Company), "Resolution of NUREG-0737 Item II.K.3.28, Verify Qualification of Accumulators on Automatic Depressurization Valves", dated June 16, 1986.

JUSTIFICATION FOR DEVIATIONS FROM NUREG-1433, 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, analysis description, or licensing basis 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 deleted.

6. The brackets have been removed and the proper plant specific information/value has been provided.

All changes are [1] unless otherwise noted

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 Core Spray (CS) System and the 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 judgement, 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.

The low pressure ECCS subsystems satisfy Criterion 3 of ~~the NRC Policy Statement~~ 10 CFR 50.36 (c)(2)(ii).

LCO

Two low pressure ECCS injection/spray subsystems are required to be OPERABLE. The low pressure ECCS injection/spray subsystems consist of two CS subsystems and two LPCI subsystems. Each CS subsystem consists of one motor driven pump, piping, and valves to transfer water from the suppression pool or contaminated condensate storage tank (CST) or the CCST to the reactor pressure vessel (RPV). Each LPCI subsystem consists of one motor driven pump, piping, and valves to transfer water from the suppression pool to the RPV. ~~Only~~ A single LPCI pump is required per subsystem because of ~~the larger~~ similar injection capacity in relation to a CS subsystem. ~~In MODES 4 and 5, the RHR System cross tie valve is not required to be closed.~~

(continued)

BASES

Capable of being

LCO (continued)

during alignment and operation for decay heat removal

Insert LCO

One LPCI subsystem may be aligned for decay heat removal and considered OPERABLE for the ECCS function, if it can be manually realigned (remote or local) to the LPCI mode and is not otherwise inoperable. 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.

2

APPLICABILITY

OPERABILITY of the low pressure 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 ≥ 23 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 CS System and the LPCI subsystems can provide core cooling without any depressurization of the primary system.

The High Pressure Coolant Injection System is not required to be OPERABLE during MODES 4 and 5 since the low pressure ECCS injection/spray subsystems can provide sufficient flow to the vessel.

ACTIONS

A.1 and B.1

If any one required low pressure ECCS injection/spray subsystem is inoperable, the inoperable subsystem must be restored to OPERABLE status in 4 hours. In this Condition, the remaining OPERABLE subsystem can provide sufficient vessel flooding capability to recover from an inadvertent vessel draindown. However, overall system reliability is

(continued)

2 INSERT LCO

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.

BASES

ACTIONS

A.1 and B.1 (continued)

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 low pressure ECCS injection/spray subsystem to OPERABLE status is based on engineering judgment that considered the remaining available subsystem and the low probability of a vessel draindown event.

With the inoperable subsystem not restored to OPERABLE status in the required Completion Time, action must be immediately initiated 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

With both of the required ECCS injection/spray subsystems inoperable, all coolant inventory makeup capability may be unavailable. Therefore, actions must immediately be initiated to suspend OPDRVs 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. <move from next page>

secondary containment

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.

If at least one low pressure 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 (i.e., one isolation valve and associated instrumentation are OPERABLE or other acceptable administrative controls to assure isolation capability) is available in each associated penetration flow path not isolated that is assumed to be isolated to mitigate radioactivity releases. OPERABILITY may be verified by an administrative check, or by examining logs or other information, to determine whether the components are out of service for maintenance or other reasons. It is not

2

(continued)

All changes are [1] unless otherwise indicated

BASES

ACTIONS

C.1, C.2, D.1, D.2, and D.3 (continued)

necessary to perform the Surveillances needed to demonstrate the OPERABILITY of the components. If, however, any required component is inoperable, then it must be restored to OPERABLE status. In this case, the Surveillance may need to be performed to restore the component to OPERABLE status. Actions must continue until all required components are OPERABLE.

move to previous page

The 4 hour Completion Time to restore at least one low pressure 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]

above the bottom of the suppression chamber

SURVEILLANCE REQUIREMENTS

SR 3.5.2.1 ~~and SR 3.5.2.2~~ [3]

required low pressure ECCS injection/spray subsystems are

The minimum water level of [~~12 ft 2 inches~~] 8.5 ft [4] required for the suppression pool is periodically verified to ensure that the suppression pool will provide adequate net positive suction head (NPSH) for the CS System and LPCI subsystem 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.

required low pressure injection/spray subsystems

and LPCI pump

and LPCI subsystems are

When suppression pool level is < [~~12 ft 2 inches~~] 8.5 ft [4], the CS ~~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 CS pump. Therefore, a verification that either the suppression pool water level is ≥ [~~12 ft 2 inches~~] 8.5 ft [4] or that CS is aligned to take suction from the CST and the CST contains ≥ [~~50,000~~] 110,000 [4] gallons of water, equivalent to [~~12~~] 7.5 ft, ensures that the ~~CS system~~ can supply at least [~~50,000~~] 50,000 [4] gallons of makeup water to the RPV. ~~The CS suction is uncovered at the [100,000] gallon level.~~ However, as noted, only one required CS subsystem may take credit for the CST option during OPDRVs. During OPDRVs, the volume in the CST may not provide adequate makeup if the RPV were completely drained. Therefore, only one CS subsystem is allowed to use the CST. This ensures the other required ECCS subsystem has adequate makeup volume.

low pressure ECCS injection/spray

(continued)

The water volume below a CCST level of 4 ft (approximately 60,000 gallons) is not considered available to the low pressure pumps since vortexing may occur.

BASES

SURVEILLANCE REQUIREMENTS

SR 3.5.2.1 and SR 3.5.2.2 (continued)

The 12 hour Frequency of these SRs was developed considering operating experience related to suppression pool water level 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 available in the control room, including alarms, to alert the operator to an abnormal suppression pool or CST water level condition.

SR 3.5.2.3, SR 3.5.2.4, and SR 3.5.2.5

The Bases provided for SR 3.5.1.1, SR 3.5.1.3, and SR 3.5.1.10 are applicable to SR 3.5.2.3, SR 3.5.2.4, and SR 3.5.2.5, respectively.

SR 3.5.2.6

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

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.5.2.4 (continued)

~~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 RPV draindown should occur.

during alignment and operation for decay heat removal,

capable of being

the LPCI mode and

REFERENCES

1. FSAR, Section ~~6.3.2.1~~.

3.1.2.1

Insert SR 3.5.2.3

2 INSERT SR 3.5.2.3

Alignment and operation for decay heat removal includes when the required RHR pump is not operating or when the system is being realigned from or to the RHR shutdown cooling mode. Because of the 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.

JUSTIFICATION FOR DEVIATIONS FROM NUREG-1433, 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, analysis description or licensing basis 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 the RPV water level. Under these conditions, the High Pressure Coolant Injection (HPCI) 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, where the coolant is distributed within the RPV through the feedwater sparger. 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, 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 a main steam line upstream of the associated inboard main steam line isolation valve.

The RCIC System is designed to provide core cooling for a wide range of reactor pressures 165 psig to 1255 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 from and to the CST 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 open 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 piping is kept full of water. The RCIC System is normally aligned to the CST. The height of water in the CST is sufficient to maintain the piping full of water up to the first isolation valve. The relative height of the feedwater line connection for RCIC is such that the water in the feedwater lines keeps the remaining portion of the RCIC discharge line full of water. Therefore, RCIC does not require a "keep fill" system.

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.

LCO

The OPERABILITY of the RCIC System provides adequate core cooling such that actuation of any of the low pressure ECCS subsystems is not required in the event of RPV isolation accompanied by a loss of feedwater flow. The RCIC System has sufficient capacity for maintaining RPV inventory during an isolation event.

APPLICABILITY

The RCIC System is required to be OPERABLE during 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 low pressure ECCS injection/spray subsystems can provide sufficient flow to the RPV.

(continued)

BASES (continued)

ACTIONS

A.1 and A.2

If the RCIC System is inoperable during MODE 1, or MODE 2 or 3 with reactor steam dome pressure > ~~[150]~~ psig, and the HPCI 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 reactor pressure since the HPCI System is the only high pressure system assumed to function during a loss of coolant accident (LOCA). OPERABILITY of HPCI is therefore verified ~~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 HPCI is out of service for maintenance or other reasons. It does not mean it is necessary to perform the Surveillances needed to demonstrate the OPERABILITY of the HPCI System. If the OPERABILITY of the HPCI System cannot be verified, however, Condition B must be ~~immediately~~ immediately entered. For transients and certain abnormal events with no LOCA, RCIC (as opposed to HPCI) is the preferred source of makeup coolant because of its relatively small capacity, which allows easier control of the 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. 2) that evaluated the impact on ECCS availability, assuming 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 similar functions of HPCI and RCIC, the AOTs (i.e., Completion Times) determined for HPCI 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 HPCI 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

(continued)

BASES

ACTIONS

B.1 and B.2 (continued)

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 3.5.3.1

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.

3

SR 3.5.3.2

(including the RCIC pump flow controller)

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 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. 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 once every 92 days. The Frequency of

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.5.3.2 (continued)

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.

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 system head 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. 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~~ psig to perform SR 3.5.3.4. Adequate steam flow is represented by ~~at least 1.25~~ turbine bypass valves open ~~or total steam flow ≥ 10⁶ lb/hr~~. Adequate reactor steam pressure must be available to perform these tests. 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 allowed 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 Surveillance 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 the Surveillance under 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~~ 24 month Frequency, which is based on the

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.5.3.3 and SR 3.5.3.4 (continued)

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 in order to verify its design function satisfactorily. This Surveillance verifies that, with a required system initiation signal (actual or simulated), the automatic initiation logic of the RCIC System will cause the system to operate as designed, including actuation of the system throughout its emergency operating sequence; that is, automatic pump startup and actuation of all automatic valves to their required positions. This test also ensures 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.

The 18 month Frequency is based on the need to perform the 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.

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.

(continued)

BASES (continued)

REFERENCES

1. ~~10 CFR 50, Appendix A, GDC 33.~~ 1

2. FSAR, Section ~~[5.5.6]~~ 5.4.6. 2

3. Memorandum from R.L. Baer (NRC) to V. Stello, Jr. (NRC), "Recommended Interim Revisions to LCOs for ECCS Components," December 1, 1975.

JUSTIFICATION FOR DEVIATIONS FROM NUREG-1433, 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, analysis description, or licensing basis 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. 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 reflect those changes made to the Specification. The following requirements have been renumbered, where applicable, to reflect the changes.

GENERIC NO SIGNIFICANT HAZARDS CONSIDERATION
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.

GENERIC NO SIGNIFICANT HAZARDS CONSIDERATION
ITS: SECTION 3.5 - ECCS AND RCIC SYSTEM

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-1433, 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:
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 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 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?

   The proposed change reduces the number of ADS valves required to be OPERABLE from five to four. 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 ADS valves are not assumed to be an initiator of any analyzed event. The ADS valves function to mitigate the consequences of analyzed events by reducing the reactor vessel pressure to allow other ECCS components to function as needed. The change is based on analysis summarized in UFSAR Section 6.3.3.1.4. This analysis demonstrates adequate core cooling is provided during a small break LOCA and a simultaneous battery failure (i.e., battery failure and resulting HPCI System failure) with two of the five ADS valves out-of-service. This change reflects the credit provided through the use of NRC approved methods for calculating more realistic (yet conservative) peak cladding temperatures during accident situations. The consequences of an accident are not affected by this change since it has been analyzed by appropriate methods.

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 proposed change does not involve a significant reduction in a margin of safety since sufficient ADS valves are maintained to ensure the safety analysis assumptions are met. The safety analysis shows that with a battery failure (i.e., battery failure and resulting HPCI System failure), three ADS valves are sufficient to lower reactor pressure to allow low pressure ECCS injection and cooling, thus ensuring the 10 CFR 50, Appendix K limits are maintained.

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 ECCS discharge line "keep filled" alarm 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 ECCS discharge line pressure is still required to be checked per SR 3.5.1.1. One method to perform the verification required by SR 3.5.1.1 would require the instrumentation to be calibrated within the required frequency (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 this change involve a significant reduction in a margin of safety?

   The proposed deletion of the ECCS discharge line "keep filled" alarm 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 discharge line pressure is still required to be checked per SR 3.5.1.1. One method to perform the verification required by SR 3.5.1.1 would require the instrumentation to be calibrated within the required frequency (OPERABLE). 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. 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 low pressure ECCS subsystems are not assumed to be an initiator of any analyzed event. While a HPCI System inadvertent injection is an analyzed event, this change does not negatively impact the event. As such the inoperability of ECCS systems will not increase the probability of any accident previously evaluated. In addition, while the ECCS equipment is used to mitigate the consequences of an accident, the proposed ACTIONS are bounded by the analyses summarized in the UFSAR Section 6.3.3, and therefore, does not involve any increase to 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 combination of inoperable ECCS has been previously evaluated and the length of the allowable outage time specified permitted is consistent with other comparable combinations of inoperable ECCS systems.

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 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.

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 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 one of the two required ECCS subsystems need not depend on suppression pool volume as a water source when aligned to an OPERABLE condensate storage tank, the one subsystem would not be rendered inoperable and the directions for suspending OPDRVs are adequately addressed by the ACTIONS for inoperable ECCS. Therefore, this change does not 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 initiation, response, and effectiveness of one of the two required ECCS subsystems need not depend upon, nor is it impacted by low suppression pool level. Further, 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.

L.3 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 or Refuel 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 or Refuel position without the explicit requirement to "lock" the reactor mode switch in position. Reactor mode switch positions other than Refuel or Shutdown 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 or Refuel position was specified in the Technical Specifications to ensure that the reactor mode switch was not inadvertently moved from the Shutdown or Refuel 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 or Refuel position without the explicit requirement to "lock" the reactor mode switch in Shutdown or Refuel. Reactor mode switch positions other than Refuel or Shutdown 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.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 proposed change would remove unnecessary additional performance of Surveillances which have been performed within their normally required Frequency. Not performing the Surveillances would not affect any equipment which is assumed to be an initiator of any analyzed event. Since each Surveillance will be performed on its normal Frequency, there is no impact on the capability of the system to perform its required safety function. It is overly conservative to assume that systems or components are inoperable when a Surveillance Requirement has not been performed. In fact, the opposite is the case; the vast majority of Surveillance Requirements performed demonstrate that systems or components are OPERABLE. Therefore, the proposed 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 create the possibility of a new or different kind of accident from any accident previously evaluated because the proposed change does not introduce a new mode of plant operation and does not involve physical modification of the plant.

3. Does this change involve a significant reduction in a margin of safety?

   The normal Frequency of each Surveillance has been shown, based on operating experience, to be adequate for assuring the equipment is available and capable of performing its intended function. Additionally, the requirements of proposed SR 3.0.4 (CTS 4.0.D) provide assurance the equipment is OPERABLE prior to entering the MODES for which it is required. Therefore, the proposed 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 has determined it does not involve a significant hazards consideration based on the following:

1. Does the change involve a significant increase in the probability or consequences of an accident previously evaluated?

   The purpose of the condensate storage tank (known as the contaminated condensate storage tank in the ITS) water level requirement in MODE 4 and 5 is to help ensure the ECCS can mitigate loss of core inventory accidents. The condensate storage tank water level is not an initiator of any analyzed event, therefore, the probability of any previously analyzed accident is not increased by this change. The proposed water level still ensures that all ECCS pumps can operate satisfactorily (net positive suction head and vortex prevention requirements are met), and adequate recirculation/makeup volume is provided. The proposed water level requirement will ensure there is a sufficient volume of water available for more than ten minutes with one ECCS pump operating at the required flow rate. This will provide time for the operators to obtain additional water supply for the condensate storage tank or obtain an alternate makeup source. Therefore, the consequences of an accident are not significantly increased by this change.

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 purpose of the condensate storage tank water level requirement in MODE 4 and 5 is to help ensure the ECCS can mitigate loss of coolant inventory events. Reducing the minimum condensate storage tank water level requirement does not involve a significant reduction in a margin of safety since the proposed volume in feet will still ensure that all ECCS pumps can operate satisfactorily (net positive suction head and vortex prevention requirements are met), and adequate recirculation/makeup volume is provided. The proposed water level requirement will ensure there is a sufficient volume of water available for more than ten minutes with one ECCS pump operating at the required flow rate. This will provide time for the operators to obtain additional water supply for the condensate storage tank or obtain an alternate makeup source.

NO SIGNIFICANT HAZARDS CONSIDERATION
ITS: 3.5.3 - RCIC SYSTEM

There were no plant specific less restrictive changes identified for this Specification.

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 to meet 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.