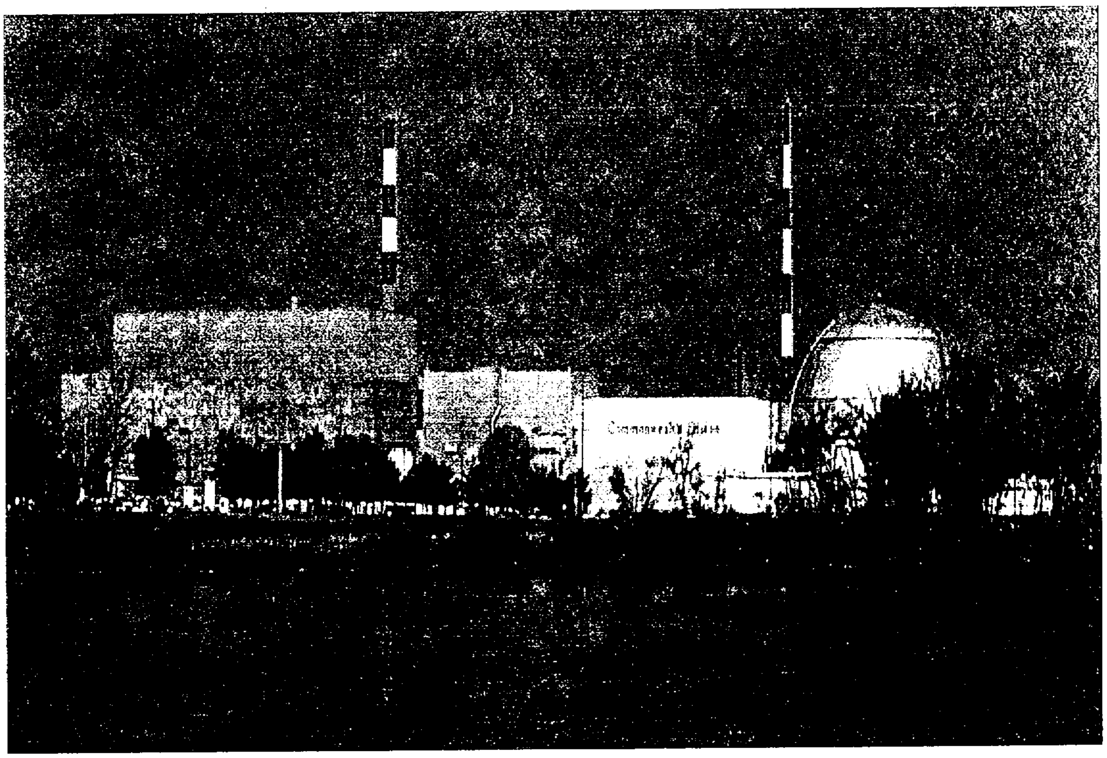
Improved Technical Specifications



Dresden Station

Volume 5:
Section 3.4 and Section 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,

<u>OR</u>

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. <u>AND</u> A.2 Be in MODE 3.	8 hours 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 8 safety valves shall be OPERABLE.

<u>AND</u>

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. <u>OR</u> Two or more relief valves inoperable. <u>OR</u> One or more safety valves inoperable.	B.1 Be in MODE 3. <u>AND</u> B.2 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) 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. $\leq$ 5 gpm unidentified LEAKAGE;

c. $\leq$ 25 gpm total LEAKAGE averaged over the previous 24 hour period; and

d. $\leq$ 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. <u>OR</u> Pressure boundary LEAKAGE exists.	C.1 Be in MODE 3. <u>AND</u> C.2 Be in MODE 4.	12 hours 36 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
SR 3.4.4.1 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 drywell floor drain sump monitoring system shall be OPERABLE.

APPLICABILITY: MODES 1, 2, and 3.

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. Drywell floor drain sump monitoring system inoperable.	A.1	Restore drywell floor drain sump monitoring system to OPERABLE status.	24 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	Perform a CHANNEL FUNCTIONAL TEST of drywell floor drain sump monitoring system instrumentation.	31 days
SR 3.4.5.2	Perform a CHANNEL CALIBRATION of drywell floor drain sump monitoring system instrumentation.	12 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
	<u>AND</u>	
	A.2 Restore DOSE EQUIVALENT I-131 to within limits.	48 hours
B. Required Action and associated Completion Time of Condition A not met. <u>OR</u> Reactor Coolant specific activity > 4.0 μCi/gm Dose EQUIVALENT I-131.	B.1 Determine DOSE EQUIVALENT I-131.	Once per 4 hours
	<u>AND</u>	
	B.2.1 Isolate all main steam lines.	12 hours
	<u>OR</u>	
		(continued)

ACTIONS

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

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.4.6.1	------------------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 Shutdown Cooling (SDC) System — Hot Shutdown

LCO 3.4.7 Two required SDC subsystems shall be OPERABLE, and, with no recirculation pump in operation, at least one SDC subsystem shall be in operation.

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

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

2. One required SDC subsystem may be inoperable for up to 2 hours for the performance of Surveillances.

APPLICABILITY: MODE 3, with reactor vessel coolant temperature less than the SDC cut-in permissive temperature.

ACTIONS

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

1. LCO 3.0.4 is not applicable.

2. Separate Condition entry is allowed for each SDC subsystem.

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

ACTIONS

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

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.4.7.1	-------------------NOTE------------------- Not required to be met until 2 hours after reactor vessel coolant temperature is less than the SDC cut-in permissive temperature. -- Verify one SDC subsystem or recirculation pump is operating.	12 hours

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.8 Shutdown Cooling (SDC) System – Cold Shutdown

LCO 3.4.8 Two required SDC subsystems shall be OPERABLE, and, with no recirculation pump in operation, at least one SDC subsystem shall be in operation.

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

1. Both required SDC subsystems may be not in operation during hydrostatic testing.

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

3. One required SDC 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 required SDC subsystems inoperable.	A.1	Verify an alternate method of decay heat removal is available for each inoperable required SDC subsystem.	1 hour <u>AND</u> Once per 24 hours thereafter

(continued)

ACTIONS

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

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
SR 3.4.8.1 Verify one SDC subsystem or recirculation pump is operating.	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. AND A.2 Determine RCS is acceptable for continued operation.	30 minutes 72 hours
B. Required Action and associated Completion Time of Condition A not met.	B.1 Be in MODE 3. AND B.2 Be in MODE 4.	12 hours 36 hours

(continued)

ACTIONS

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

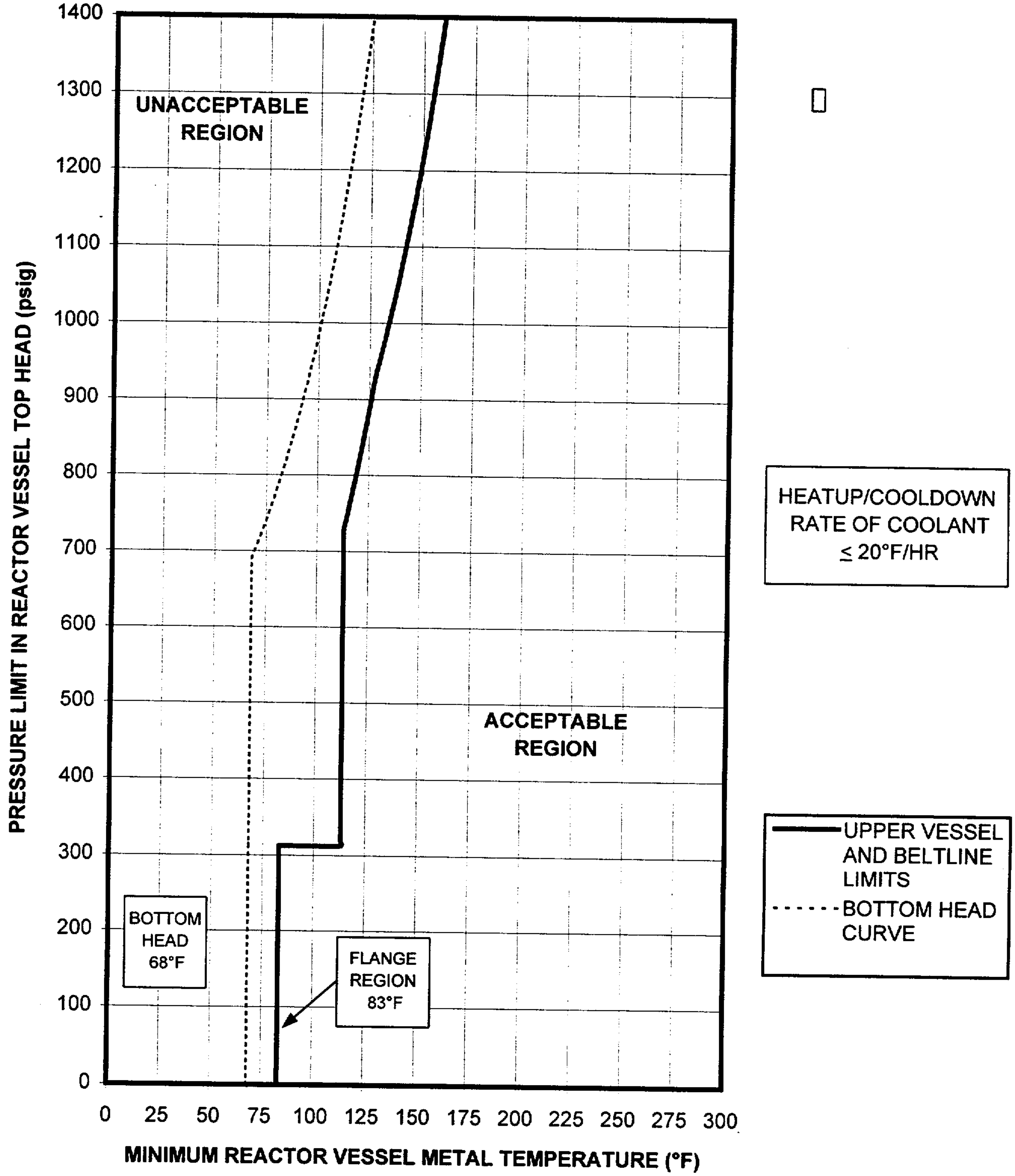


Figure 3.4.9-1 (Page 1 of 1)
Non-Nuclear Inservice Leak and Hydrostatic Testing Curve
(Valid to 32 EFPY)

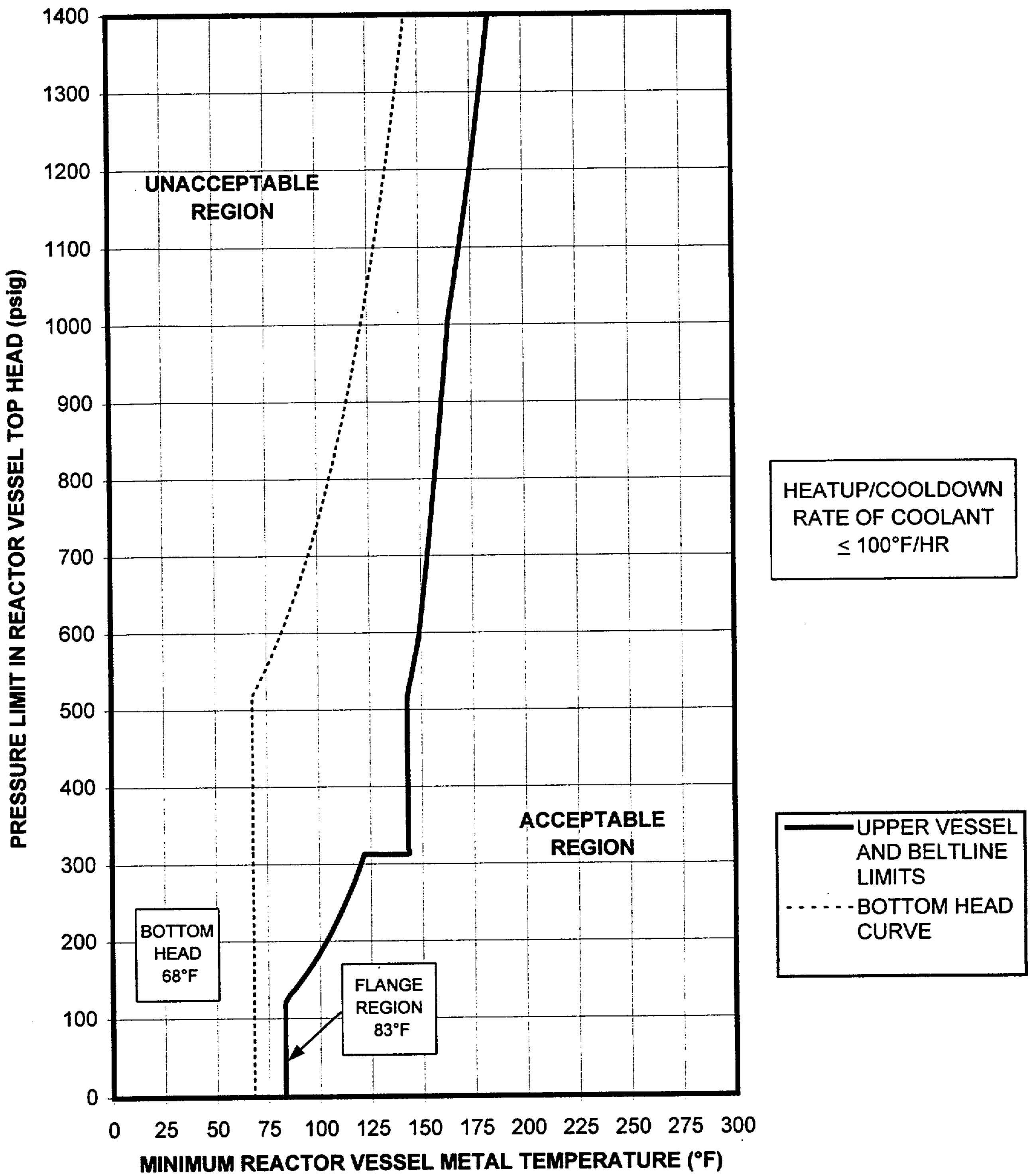


Figure 3.4.9-2 (Page 1 of 1)
Non-Nuclear Heatup/Cooldown Curve
(Valid to 32 EFPY)

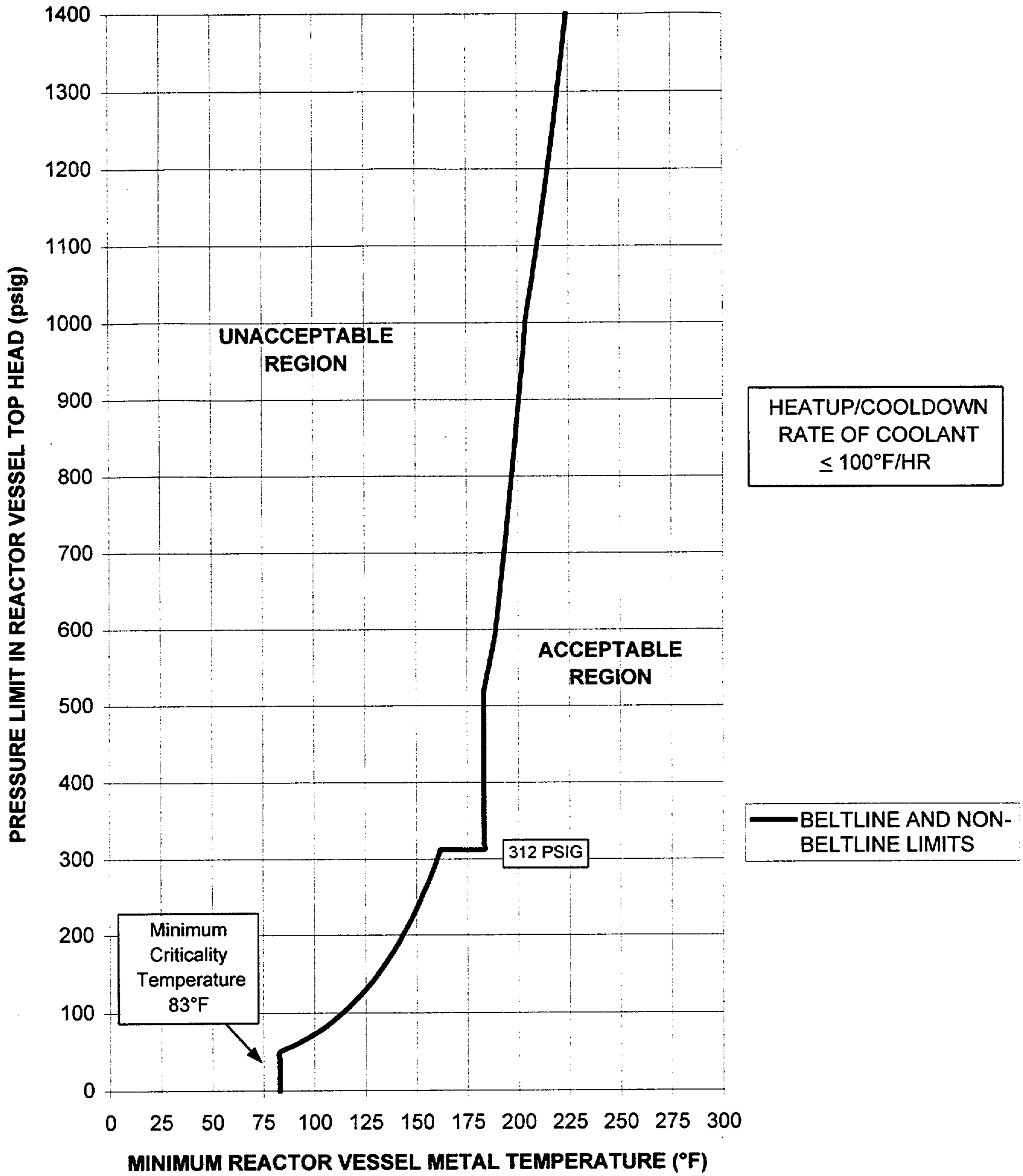


Figure 3.4.9-3 (Page 1 of 1)
Critical Operations Curve
(Valid to 32 EFPY)

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.10 Reactor Steam Dome Pressure

LCO 3.4.10 The reactor steam dome pressure shall be $\leq$ 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 $\leq$ 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. 1).

The transient analyses in Chapter 15 of the UFSAR have also been performed for single recirculation loop operation (Ref. 3) and demonstrate sufficient flow coastdown characteristics to maintain fuel thermal margins during the abnormal operational transients analyzed provided the MCPR requirements are modified. During single recirculation loop operation, modification to the Reactor Protection System (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 1.

(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 6.3.3.3. 2. UFSAR, Chapter 15. 3. UFSAR, Section 15.3.1.

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	<u>A.1</u> 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.

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.

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

B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.3 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 which 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 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. 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

(continued)

BASES

BACKGROUND (continued)	through a discharge line to a point below the minimum water level in the suppression pool. The safety valves discharge directly to the drywell. 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. 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. 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."
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, eight safety valves are assumed to operate in the safety mode. The relief valves and S/RV 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.

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)	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 eight 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 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.

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BASES

LCO (continued)	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, eight safety valves (not including the S/RV) and five relief valves (including the S/RV) 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 Shutdown Cooling (SDC) System is capable of dissipating the core heat. In MODE 4, decay heat is low enough for the Shutdown Cooling 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 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.

(continued)

BASES

ACTIONS (continued)

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.

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

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.4.3.2 (continued)

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 2.0 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.

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.

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.4.3.3 (continued)

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.
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, Section 3.1.2.4.1 (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. <u>Pressure Boundary LEAKAGE</u>

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

(continued)

BASES

LCO (continued)

b. Unidentified LEAKAGE

The 5 gpm of unidentified LEAKAGE is allowed as a reasonable minimum detectable amount that the 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." Sump level and flow rate are 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, Section 3.1.2.4.1.

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, Section 3.1.2.4.1 (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 measuring flow from the drywell floor drain sump. Although alternate methods of detecting RCS LEAKAGE are available, the sole means of quantifying LEAKAGE in the drywell is the drywell floor drain sump monitoring system.

The drywell floor drain sump monitoring system monitors the LEAKAGE collected in the floor drain sump. This unidentified LEAKAGE consists of LEAKAGE from control rod drives, valve flanges or packings, floor drains, the Reactor Building Closed Cooling Water System, and drywell air cooling unit condensate drains, 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 floor drain collector tank.

(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 Management Systems. The pumps alternate as lead and backup on each successive start. When a high level is reached in the floor drain sump, a level switch actuates to start the lead floor drain sump pump when the pump discharge valves are open. In the event the level continues to rise, a second level switch actuates to start the backup floor drain sump pump and initiates an alarm in the control room. When the level decreases to a low level, both floor drain sump pumps are stopped. A flow transmitter in the discharge line of the drywell floor drain sump pumps provides flow indication in the control room. In addition, a leak rate recorder is provided capable of identifying a 1 gpm change over an 8 hour period. The pumps can also be started from the control room.

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. 3 and 4). The leakage detection system inside the drywell is designed with the capability of detecting LEAKAGE less than the established LEAKAGE rate limits and providing appropriate alarm of excess LEAKAGE in the control room.

A control room alarm allows the operators to evaluate the significance of the indicated LEAKAGE and, if necessary, shut down the reactor for further investigation and corrective action. The allowed LEAKAGE rates are well below the rates predicted for critical crack sizes (Ref. 5). Therefore, these actions provide adequate response before a significant break in the RCPB can occur.

RCS leakage detection instrumentation satisfies Criterion 1 of 10 CFR 50.36(c)(2)(ii).

LCO

The drywell floor drain sump 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. Other monitoring systems (e.g., particulate, temperature) are

(continued)

BASES

LCO (continued)	available to the operators so closer examination can be made to determine the extent of any corrective action that may be required. With the drywell floor drain sump monitoring system inoperable, monitoring for LEAKAGE in the RCPB is degraded.
APPLICABILITY	In MODES 1, 2, and 3, the drywell floor drain sump monitoring system is required to be OPERABLE to support LCO 3.4.4. This Applicability is consistent with that for LCO 3.4.4.
ACTIONS	<u>A.1</u> With the drywell floor drain sump monitoring system inoperable, no other form of sampling can provide the equivalent information to quantify leakage. However, other monitoring systems are available that 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 24 hours. The 24 hour Completion Time of Required Action A.1 is acceptable, based on operating experience, considering the alternative forms of leakage detection that are still available. <u>B.1 and B.2</u> If the Required Action and associated Completion Time of Condition A 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.

(continued)

BASES (continued)

SURVEILLANCE REQUIREMENTS

SR 3.4.5.1

This SR is for the performance of a CHANNEL FUNCTIONAL TEST of the drywell floor drain sump monitoring system instrumentation. The test ensures that the system can perform its 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 3.4.5.2

This SR is for the performance of a CHANNEL CALIBRATION of the drywell floor drain sump monitoring system instrumentation channel (i.e., drywell floor drain sump pump discharge flow integrator). The calibration verifies the accuracy of the instrument string. The Frequency of SR 3.4.5.2 is based on the assumption of a 12 month calibration interval in the determination of the magnitude of equipment drift in the setpoint analysis.

REFERENCES

1. UFSAR, Section 3.1.2.4.1.

2. Regulatory Guide 1.45, May 1973.

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

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

5. UFSAR, Section 5.2.5.6.4.

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 ≤ 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	<u>A.1 and A.2</u> 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

(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 Shutdown Cooling (SDC) 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 $\leq$ 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 three redundant, manually controlled shutdown cooling subsystems (loops) of the SDC System provide decay heat removal. Each loop consists of one motor driven pump, a heat exchanger, and associated piping and valves. Each loop has 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 SDC heat exchangers transfer heat to the Service Water System via the Reactor Building Closed Cooling Water (RBCCW) System.

APPLICABLE SAFETY ANALYSES

Decay heat removal by operation of the SDC 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 SDC System meets Criterion 4 of 10 CFR 50.36(c)(2)(ii).

LCO

Two SDC subsystems are required to be OPERABLE, and when no recirculation pump is in operation, one SDC subsystem must be in operation. An OPERABLE SDC subsystem consists of one OPERABLE SDC pump, one heat exchanger, the associated piping and valves, and the necessary portions of the RBCCW System capable of providing cooling water to the heat exchanger and SDC pump seal cooler. The subsystems have a common suction source and common discharge piping. Thus, to meet the LCO, two loops must be OPERABLE. Since the piping is a passive component that is assumed not to fail, it is allowed to be

(continued)

BASES

LCO (continued)

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 SDC subsystem can provide the required cooling, but two subsystems are required to be OPERABLE to provide redundancy. Operation of one subsystem can maintain or reduce the reactor coolant temperature as required. To ensure adequate core flow to allow for accurate average reactor coolant temperature monitoring, nearly continuous operation is required.

Note 1 permits both SDC subsystems and recirculation pumps to not be in operation for a period of 2 hours in an 8 hour period. Note 2 allows one SDC subsystem to be inoperable for up to 2 hours for the performance of Surveillance tests. These tests may be on the affected SDC System or on some other plant system or component that necessitates placing the SDC 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 SDC subsystems or other operations requiring SDC flow interruption and loss of redundancy.

APPLICABILITY

In MODE 3 with reactor vessel coolant temperature below the SDC cut-in permissive temperature (i.e., the actual temperature at which the interlock resets) the SDC 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 vessel coolant temperature greater than or equal to the SDC cut-in permissive temperature, this LCO is not applicable. Operation of the SDC System in the shutdown cooling mode is not allowed above this temperature because the RCS temperature may exceed the design temperature of the shutdown cooling piping. Decay heat removal at reactor temperatures greater than or equal to the SDC cut-in permissive temperature is typically accomplished by condensing the steam in the main condenser.

(continued)

BASES

APPLICABILITY (continued)

The requirements for decay heat removal in MODES 4 and 5 are discussed in LCO 3.4.8, "Shutdown Cooling (SDC) System - Cold Shutdown"; LCO 3.9.8, "Shutdown Cooling (SDC) - High Water Level"; and LCO 3.9.9, "Shutdown Cooling (SDC) - 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 SDC 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 SDC subsystem.

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

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

(continued)

BASES

ACTIONS

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

With both required SDC subsystems inoperable, an alternate method of decay heat removal must be provided in addition to that provided for the initial SDC 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 and the Reactor Water Cleanup System (by itself or using feed and bleed in combination with the Control Rod Drive System or Condensate/Feed 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 SDC subsystem and no recirculation pump in operation, except as permitted by LCO Note 1, reactor coolant circulation by the SDC subsystem or recirculation pump must be restored without delay.

Until SDC or recirculation pump operation is re-established, an alternate method of reactor coolant circulation must be placed into service. This will provide the necessary circulation for monitoring coolant temperature. The 1 hour Completion Time is based on the coolant circulation function and is modified such that the 1 hour is applicable separately for each occurrence involving a loss of coolant circulation. Furthermore, verification of the functioning of the alternate method must be reconfirmed every 12 hours thereafter. This will provide assurance of continued temperature monitoring capability.

(continued)

BASES

ACTIONS

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

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

SURVEILLANCE REQUIREMENTS

SR 3.4.7.1

This Surveillance verifies that one SDC 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 SDC subsystem in the control room.

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

REFERENCES

None.

B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.8 Shutdown Cooling (SDC) 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 ≤ 212°F in preparation for performing Refueling or maintenance operations, or the decay heat must be removed for maintaining the reactor in the Cold Shutdown condition. The three redundant, manually controlled shutdown cooling subsystems (loops) of the SDC System provide decay heat removal. Each loop consists of one motor driven pump, a heat exchanger, and associated piping and valves. Each loop has 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 low pressure coolant injection path and associated recirculation loop. The SDC heat exchangers transfer heat to the Service Water System, via the Reactor Building Closed Cooling Water (RBCCW) System.
APPLICABLE SAFETY ANALYSES	Decay heat removal by operation of the SDC 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 SDC System meets Criterion 4 of 10 CFR 50.36(c)(2)(ii).
LCO	Two SDC subsystems are required to be OPERABLE, and when no recirculation pump is in operation, one SDC subsystem must be in operation. An OPERABLE SDC subsystem consists of one OPERABLE SDC pump, one heat exchanger, the associated piping and valves, and the necessary portions of the RBCCW System capable of providing cooling water to the heat exchanger and SDC pump seal cooler. The subsystems have a common suction source and common discharge piping. Thus, to meet the LCO, two loops must be OPERABLE. Since the piping is a passive

(continued)

BASES

LCO (continued)

component that is assumed not to fail, it is allowed to be common to both subsystems. 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 SDC subsystem can provide the required cooling, but two subsystems are required to be OPERABLE to provide redundancy. Operation of one subsystem can maintain or reduce the reactor coolant temperature as required. To ensure adequate core flow to allow for accurate average reactor coolant temperature monitoring, nearly continuous operation is required.

Note 1 allows both SDC subsystems to not be in operation during hydrostatic testing. This allowance is acceptable because adequate reactor coolant circulation will be maintained by operation of a reactor recirculation pump to ensure adequate core flow and since systems are available to control reactor coolant temperature. Note 2 permits both SDC subsystems and recirculation pumps to not be in operation for a period of 2 hours in an 8 hour period. Note 3 allows one SDC subsystem to be inoperable for up to 2 hours for the performance of Surveillance tests. These tests may be on the affected SDC System or on some other plant system or component that necessitates placing the SDC 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 SDC subsystems or other operations requiring SDC flow interruption and loss of redundancy.

APPLICABILITY

In MODE 4, the SDC System must be OPERABLE and one SDC subsystem shall be operated in the shutdown cooling mode to remove decay heat to maintain coolant temperature below 212°F. Otherwise, a recirculation pump is required to be in operation.

In MODES 1 and 2, and in MODE 3 with reactor vessel coolant temperature greater than or equal to the SDC cut-in permissive temperature, this LCO is not applicable. Operation of the SDC System in the shutdown cooling mode is not allowed above this temperature because the RCS temperature may exceed the design temperature of the

(continued)

BASES

APPLICABILITY (continued)

shutdown cooling piping. Decay heat removal at reactor temperatures greater than or equal to the SDC cut-in permissive temperature is typically accomplished by condensing the steam in the main condenser.

The requirements for decay heat removal in MODE 3 below the cut-in permissive temperature and in MODE 5 are discussed in LCO 3.4.7, "Shutdown Cooling (SDC) System — Hot Shutdown"; LCO 3.9.8, "Shutdown Cooling (SDC) — High Water Level"; and LCO 3.9.9, "Shutdown Cooling (SDC) — Low Water Level."

ACTIONS

A Note has been provided to modify the ACTIONS related to SDC 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 SDC subsystem.

A.1

With one of the two required SDC subsystems inoperable, except as permitted by LCO Notes 1 and 3, the remaining subsystem is capable of providing the required decay heat removal. However, the overall reliability is reduced. Therefore, an alternate method of decay heat removal must be provided. With both required SDC subsystems inoperable, an alternate method of decay heat removal must be provided in addition to that provided for the initial SDC subsystem inoperability. This re-establishes backup decay heat removal capabilities, similar to the requirements of the LCO. The 1 hour Completion Time is based on the decay heat removal function and the probability of a loss of the available decay heat removal capabilities. Furthermore,

(continued)

BASES

ACTIONS

A.1 (continued)

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 System and the Reactor Water Cleanup System (by itself or using feed and bleed in combination with the Control Rod Drive System or Condensate/Feed System).

B.1 and B.2

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

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

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BASES (continued)

SURVEILLANCE REQUIREMENTS	SR 3.4.8.1 This Surveillance verifies that one SDC 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 SDC 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,

(continued)

BASES

BACKGROUND (continued)

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

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

The non-nuclear heatup and cooldown curve applies during heatups with non-nuclear heat (e.g., recirculation pump heat) and during cooldowns when the reactor is not critical (e.g., following a scram). The curve provides the minimum reactor vessel metal temperatures based on the most limiting vessel stress.

The P/T criticality limits include the Reference 1 requirement that they be at least 40°F above the non-critical heatup curve or the 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 Dresden 2 and 3, 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 Dresden 2 and 3, 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 ≤ 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 ≤ 20°F in any 1 hour period when the RCS temperature and pressure are being maintained with 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 ≤ 145°F during recirculation pump startup in MODES 1, 2, 3, and 4; c. The temperature difference between the reactor coolant in the respective recirculation loop and in the reactor vessel is ≤ 50°F during recirculation pump startup in MODES 1, 2, 3, and 4;

(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 when the reactor head is tensioned.

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

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 3.4.9.4 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 ≤ 93°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 ≤ 113°F, monitoring of the flange temperature is required every 12 hours to ensure the temperature is within the specified limits.

(continued)

BASES

SURVEILLANCE REQUIREMENTS

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

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 Robert Pulsifer (NRC) to ComEd, "Issuance of Amendments 153 and 148 for Dresden 2 and 3," dated February 28, 1997.]
8. UFSAR, Section 15.4.4.

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	<u>SR 3.4.10.1</u> Verification that reactor steam dome pressure is ≤ 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

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

A.2 — LCO 3.4.1

b. Increase the MINIMUM CRITICAL POWER RATIO (MCPR) Operating Limit ~~by 0.01 per Specification 3.11.C, and~~ LA.1

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

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.2

3.6 - LIMITING CONDITIONS FOR OPERATION | 4.6 - SURVEILLANCE REQUIREMENTS

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

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

2. With no reactor coolant system
ACTION A 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.3 M.1

a ~~Except to permit testing in preparation for returning the pump to service.~~ L.1

A.1

3.6 - LIMITING CONDITIONS FOR OPERATION

C. Recirculation Pumps

LCO 3.4.1 Recirculation pump speed shall be maintained within:

SR3.4.1.1
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.2

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
2. ~~Trip one of the recirculation pumps and take the ACTION required by Specification 3.6.A.1.~~ L.2 A.6

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

DISCUSSION OF CHANGES
ITS: 3.4.1 - RECIRCULATION LOOPS OPERATING

ADMINISTRATIVE

A.1 In the conversion of the Dresden 2 and 3 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 instrumentation. Reference to the Trip Setpoints has been eliminated in the

ADMINISTRATIVE

A.4 (cont'd) 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 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.

M.2 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

TECHNICAL CHANGES - MORE RESTRICTIVE

M.2 (cont'd) 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 - LESS RESTRICTIVE

"Generic"

LA.1 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.

LA.2 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).

TECHNICAL CHANGES - LESS RESTRICTIVE

LA.2 (cont'd) 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 limit 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, 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.3 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.

"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 inadvertently started. In addition, the requirements in CTS 3.6.D (ITS 3.4.9, "Reactor Coolant System 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.

TECHNICAL CHANGES - LESS RESTRICTIVE (continued)

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

A.1

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

4.6 - SURVEILLANCE REQUIREMENTS

B. Jet Pumps

All jet pumps shall be demonstrated OPERABLE as follows: add proposed SR 3.4.2.1 Note 1 L.2

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.

3.6 - LIMITING CONDITIONS FOR OPERATION

4.6 - SURVEILLANCE REQUIREMENTS

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:~~ A2

L.2 add proposed SR 3.4.2.1 NOTE 1

a. SR 3.4.2.1.a The indicated recirculation pump flow in the operating loop differs by >10% from the established single recirculation 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

c. SR 3.4.2.1.b The indicated flow of any individual jet pump differs from established single recirculation loop patterns by >10%.

d. SR 3.4.2.1 NOTE 2 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 Dresden 2 and 3 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 $\leq$ 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 Action 2 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 require 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

PRIMARY SYSTEM BOUNDARY

3.6 - LIMITING CONDITIONS FOR OPERATION

LCO 3.4.3

E. Safety Valves (Excluding the Target Rock valve, 8) M.1

A.5 The safety valve function of the ~~9~~ reactor coolant system safety valves shall be OPERABLE ~~(in accordance with the specified code safety valve function lift settings(a) established as:)~~

~~1 safety valve(b) @1135 psig ± 1%~~
2 safety valves @1240 psig ± 1%
2 safety valves @1250 psig ± 1%
4 safety valves @1260 psig ± 1%

APPLICABILITY:

OPERATIONAL MODE(s) 1, 2 and 3.

ACTION:

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 (In accordance with the Inservice Testing Program) LA.3

~~1. Deleted~~

A.5 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~~ 8 ~~9~~ ~~safety valves are removed, set pressure tested and reinstalled or replaced with spares that have been previously set pressure tested and stored in accordance with manufacturer's recommendations.~~

A.5

SR 3.4.3.1 Verify the safety function lift setpoints (a) of the required safety valves are as follows: LA.1

~~(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.~~ A.5

3.6 - LIMITING CONDITIONS FOR OPERATION

E. Safety Valves

Excluding the Target Rock valve, the safety valve function of the reactor coolant system safety valves shall be OPERABLE.

APPLICABILITY:

OPERATIONAL MODE(s) 1, 2 and 3.

ACTION:

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

2. Deleted

4.6 - SURVEILLANCE REQUIREMENTS

E. Safety Valves

1. Deleted.

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

Verify the safety function lift setpoints[a] of the required safety valves are as follows:

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%

---

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.

c The surveillance interval has been extended to 60 months for Unit 3, Cycle 15 only, and the provisions of Specification 4.0.B are not applicable to the 60-month interval.

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
≤ 1112 psig
≤ 1112 psig
≤ 1135 psig
≤ 1135 psig
≤ 1135 psig[a]

APPLICABILITY:

OPERATIONAL MODE(s) 1, 2 and 3.

ACTION:

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

4.6 - SURVEILLANCE REQUIREMENTS

F. Relief Valves ⟨See ITS 3.3.6.3⟩

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.~~

⟨add proposed SRs 3.4.3.2 and 3.4.3.3⟩ A.4

a Target Rock combination safety/relief valve. ⟨See ITS 3.3.6.3⟩

A.1

A.2 general organization

3.6 - LIMITING CONDITIONS FOR OPERATION

4.6 - SURVEILLANCE REQUIREMENTS

See ITS 3.3.6.3

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

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.~~

ADMINISTRATIVE

A.1 In the conversion of the Dresden 2 and 3 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 Not used.

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.

A.5 The change to CTS LCO 3.6.E, for Unit 2, which reduces the number of safety valves required to be OPERABLE is provided in the Dresden 2 and 3 ITS consistent with the Technical Specifications change submitted to the NRC for approval per the ComEd License Amendment Request letter PSLTR 00-0061, dated February 29, 2000. As such, this change is considered administrative.

TECHNICAL CHANGES - MORE RESTRICTIVE

M.1 CTS LCO 3.6.E ensures that the appropriate number of safety valves are available to protect the reactor vessel from overpressure during upset conditions as required by the ASME Boiler and Pressure Vessel Code. Proposed LCO 3.6.E (see Discussion of Change A.5) excludes the Target Rock combination safety and relief valve from the safety valve function OPERABILITY requirements of the LCO, but does not specify the number of safety valve functions (safety valves) that are required to be OPERABLE. The number of required safety valves is determined from plant controlled documents. However, the NUREG-1433 presentation of CTS LCO 3.6.E (ISTS LCO 3.4.3) specifies the number (in brackets) of safety valves required to be OPERABLE in order to satisfy the LCO. Therefore, proposed ITS LCO 3.4.3 includes the plant specific requirement that 8 safety valves shall be OPERABLE. Since this change proposes to include a specific number of required safety valves in the ITS, the number of valves will no longer be controlled by ComEd, subject to the provisions of 10 CFR 50.59. Instead, the number of required safety valves will be controlled by the NRC, pursuant to 10 CFR 50.90. As such, this change represents an additional restriction on plant operation and is considered a more restrictive change.

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 Not used.

LA.3 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 demonstrate the Reactor Coolant System (RCS) safety valves are OPERABLE. However, the IST Program, required by 10 CFR 50.55a, provides

TECHNICAL CHANGES - LESS RESTRICTIVE

LA.3 (cont'd) requirements for the testing of all ASME Code Class 1, 2, and 3 valves in accordance with applicable codes, standards, and relief requests and is endorsed by the NRC for Dresden 2 and 3. Compliance with 10 CFR 50.55a, and as a result the IST Program and implementing procedures, is required by the Dresden 2 and 3 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.

"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 Dresden 2 and 3 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

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~~ 24 hour surveillance period. A.2 the previous
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. ACTION C With any PRESSURE BOUNDARY LEAKAGE, be in at least HOT SHUTDOWN within 12 hours and in COLD SHUTDOWN within the next 24 hours.
2. ACTION A 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 ACTION C 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~~: LA.1

moved to ITS 3.4.5

1. Sampling the primary containment atmospheric particulate radioactivity at least once per 12 hours[a], ~~and~~ A.3

SR 3.4.4.1 ~~2.~~ ~~Determining the primary containment sump flow rate~~ at least once per ~~8 hours, not to exceed~~ 12 hours. LA.1 L.1

moved to ITS 3.4.5

a Not a means of quantifying leakage. A.3

A.1

3.6 - LIMITING CONDITIONS FOR OPERATION | **4.6 - SURVEILLANCE REQUIREMENTS**

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

Action B a. Identify the source of leakage as not IGSCC susceptible material or reduce the leakage to within limits within 4 hours, or A.4

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

ADMINISTRATIVE

A.1 In the conversion of the Dresden 2 and 3 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 control room leak rate recorder.

RELOCATED SPECIFICATIONS

None

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:

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

2. The drywell floor drain sump system.

APPLICABILITY:

OPERATIONAL MODE(s) 1, 2 and 3.

ACTION:

R.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 A) 2. With the drywell floor drain sump system inoperable, restore the drywell floor drain sump system to OPERABLE status within 24 hours; ~~otherwise~~, be

ACTION B) in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

4.6 - SURVEILLANCE REQUIREMENTS

G. Leakage Detection Systems

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

1. ~~Performing the leakage determinations of Specification 4.6.H.~~ A.2 add proposed SR 3.4.5.1 M.1

SR 3.4.5.2 2. Performing a CHANNEL CALIBRATION of the drywell floor drain sump ~~pump discharge flow integrator~~ monitoring system at least once per ~~18~~ 12 months. LA.1 M.2 A.3

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~~ R.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

a ~~Not a means of quantifying leakage.~~ R.1

DISCUSSION OF CHANGES
ITS: 3.4.5 - RCS LEAKAGE DETECTION INSTRUMENTATION

ADMINISTRATIVE

A.1 In the conversion of the Dresden 2 and 3 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 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.

A.3 The Dresden 2 and 3 design includes a single qualified leakage detection system, although other methods of RCS leakage detection are available. The words, "drywell floor drain sump pump discharge flow integrator," in CTS 4.6.G.2 are proposed to be replaced with the qualified detection system name, "drywell floor drain sump monitoring system," for clarification and to provide consistency with the proposed changes to the LCO and ACTIONS. Therefore, the words, "monitoring system," have been added to CTS 4.6.G.2. Since this change only provides plant specific clarification of the existing requirements, the change is considered administrative.

TECHNICAL CHANGES - MORE RESTRICTIVE

M.1 ITS SR 3.4.5.1 has been added to CTS 4.6.G to require a CHANNEL FUNCTIONAL TEST to be performed on the drywell floor drain sump monitoring system on a 31 day frequency. This requirement ensures the monitor can perform its function and verifies the relative accuracy of the instrument string. 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 The Frequency of the CHANNEL CALIBRATION requirement for CTS 4.6.G.2, Drywell Floor Drain Sump Monitoring System, has been increased from 18 months to 12 months (proposed ITS SR 3.4.5.2). The proposed Frequency is acceptable since it is consistent with current plant calculations. This change to the CTS requirement constitutes a more restrictive change to help ensure that the drywell floor drain sump monitoring system is maintained OPERABLE.

TECHNICAL CHANGES - LESS RESTRICTIVE

"Generic"

LA.1 The detail in CTS 4.6.G.2 of what Drywell Floor Drain Sump Monitoring System instrumentation (pump discharge flow integrator) is subject to a CHANNEL CALIBRATION is proposed to be relocated to the Bases. This detail is not necessary to ensure that a CHANNEL CALIBRATION is performed. Proposed SR 3.4.5.2, in conjunction with the Bases, requires the CHANNEL CALIBRATION to verify the accuracy of the drywell floor drain sump pump discharge flow integrator instrument string. This is consistent with the intent of CTS 4.6.G.2 and provides assurance that the instrumentation is OPERABLE when required. 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"

None

RELOCATED SPECIFICATIONS

R.1 At Dresden 2 and 3, the primary containment atmosphere particulate radioactivity sampling system is not actually a system (i.e., a sensor, indicator, etc.) It consists of two containment penetrations (sampling and return lines) and their associated isolation valves, including two air operated primary containment isolation valves (PCIVs) per line. A device can be connected to the penetration lines to obtain grab samples. Grab samples of the primary containment atmosphere can also be obtained using the primary containment sample manifold. Once obtained, the grab samples are analyzed using appropriate laboratory detector/counting systems. Since the primary containment atmosphere particulate radioactivity sampling system is not, in itself, a leakage detection system, it does not satisfy Regulatory Guide 1.45 and is not capable of detecting a significant abnormal degradation of the reactor coolant pressure boundary prior to a design basis accident (DBA). Furthermore, the evaluation summarized in NEDO-31466 determined that the loss of the primary containment atmosphere particulate radioactivity sampling system is a non-significant risk contributor to core damage frequency and offsite release. Therefore, the requirements specified for this system do not satisfy the 10 CFR 50.36(c)(2)(ii) Technical Specification screening criteria as documented in the Application of Selection Criteria to the Dresden 2 and 3 Technical Specifications and will be relocated to the UFSAR. Changes to the UFSAR will be controlled in accordance with 10 CFR 50.59.

A.1 ITS 3.4.6

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 acitivity 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.

LCO 3.0.4

Required Actions A.1 and A.2 Note a The provisions of ~~Specification 3.0.D~~ are not applicable.

ADMINISTRATIVE

A.1 In the conversion of the Dresden 2 and 3 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

A.1

ITS 3.4.7

3.6 - LIMITING CONDITIONS FOR OPERATION

O. Shutdown Cooling - HOT SHUTDOWN

LCO 3.4.7

Two[a] shutdown cooling (SDC) loops shall be OPERABLE and, unless at least one recirculation pump is in operation, at least one shutdown cooling loop shall be in operation[(b)(c)], ~~with each loop consisting of at least:~~ A.2

1. ~~One OPERABLE SDC pump, and~~

2. ~~One OPERABLE SDC heat exchanger.~~ LA.1

APPLICABILITY:

OPERATIONAL MODE 3, with reactor vessel coolant temperature less than the SDC cut-in permissive setpoint.

ACTION:

add proposed ACTIONS Note 1 — L.1

add proposed ACTIONS Note 2 — A.3

ACTION A

1. With less than the above required SDC loops OPERABLE, immediately initiate corrective action to return the required loops to OPERABLE status as soon as possible. Within 1 hour ~~and at least once per 24 hours thereafter,~~ demonstrate the operability of at least one alternate method capable of decay heat removal for each inoperable SDC loop. Be in at least COLD SHUTDOWN within 24 hours[(d)]. A.4 A.5

4.6 - SURVEILLANCE REQUIREMENTS

O. Shutdown Cooling - HOT SHUTDOWN

SR 3.4.7.1

At least one SDC loop, one recirculation pump ~~or alternate method~~ shall be verified to be in operation ~~and circulating reactor coolant~~ at least once per 12 hours. LA.2

Required Action B.2

add proposed SR 3.4.7.1 Note — L.1

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

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

c The shutdown cooling loop may be removed from operation during hydrostatic testing. A.2

d ~~Whenever two or more SDC loops are inoperable, if unable to attain COLD SHUTDOWN as required by this ACTION, maintain reactor coolant temperature as low as practical by use of alternate heat removal methods.~~ A.5

3.6 - LIMITING CONDITIONS FOR OPERATION

ACTION B

2. With no SDC loop or recirculation pump in operation, immediately initiate corrective action to return at least one shutdown cooling loop or recirculation pump to operation as soon as possible. Within 1 hour establish reactor coolant circulation by an alternate method and monitor reactor coolant temperature and pressure at least once per hour.

4.6 - SURVEILLANCE REQUIREMENTS

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

ADMINISTRATIVE

A.1 In the conversion of the Dresden 2 and 3 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.O footnote (c) allowance to remove the SDC loop from operation during hydrostatic testing has been deleted since these tests are not performed during MODE 3 operation. Since the footnote does not provide any additional allowance, its removal is considered administrative.

A.3 The proposed ACTION Note 2, "Separate Condition entry is allowed for each...," has been added to CTS 3.6.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 SDC subsystems.

A.4 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 SDC 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.5 The CTS 3.6.O Action 1 footnote (d) requirement that if unable to attain cold shutdown when two SDC 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

None

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 SDC subsystem are proposed to be relocated to the Bases. The Bases will indicate that an OPERABLE SDC subsystem consists of an OPERABLE pump, heat exchanger, and the associated piping and valves. The details for subsystem OPERABILITY are not necessary in ITS 3.4.7. 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 the method in CTS 4.6.O of verifying operation of the SDC subsystem (circulating reactor coolant) is proposed to be relocated to the Bases. This detail is not necessary for assuring SDC subsystem is in operation. Proposed ITS 3.4.7.1 requires verification an SDC subsystem is operating and is adequate to ensure an SDC subsystem is circulating reactor coolant. Therefore, the relocated detail is not required to be in the ITS to provide adequate protection of the public health and safety. Changes to the Bases will be controlled by the provisions of the proposed Bases Control Program described in Chapter 5 of the ITS.

"Specific"

L.1 CTS 3.6.O requires one SDC loop to be operation in MODE 3, with reactor vessel temperature less than the SDC cut-in permissive setpoint. CTS 4.6.O requires a verification that a loop is in 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 SDC System cannot be placed in operation until after the applicable conditions necessary to open the SDC suction valves are met (the SDC suction valves are interlocked closed at high temperature). Therefore, entry into the conditions should be allowed while depending on the ACTIONS and without performing the Surveillance Requirement. Both LCO 3.0.4 and SR 3.0.4 must be "not applicable" to provide the necessary time to plant the system

TECHNICAL CHANGES - LESS RESTRICTIVE

L.1 (cont'd) in service following the reduction of pressure to below the cut-in permissive temperature setpoint. Therefore, a Note to the CTS 3.6.O Actions (ITS 3.4.7 ACTIONS Note 1) and a Note to CTS 4.6.O (proposed SR 3.4.7.1) have been added. Without this change, certain entries into the applicable operating conditions would result in intentional temporary noncompliance until the system is placed in service.

L.2 CTS 3.6.O footnote (a) allows one shutdown cooling loop to be inoperable for 2 hours, provided the other loop is OPERABLE and in operation. CTS 3.6.O footnote (b) allows the shutdown cooling pump to be removed from operation for up to 2 hours per 8 hour period, provided the other loop is OPERABLE. The requirements in CTS 3.6.O footnotes (a) and (b) (ITS 3.4.7 LCO Notes 1 and 2) are proposed to be changed to delete the "provided" requirements. The allowances of the Notes may be required even when no shutdown cooling loop remains OPERABLE or in operation. Some Surveillances result in the inoperability of both shutdown cooling loops (e.g., Surveillances on the common suction line valves). With one or more shutdown cooling loops inoperable, CTS 3.6.O Action 1 (ITS 3.4.7 ACTION A) requires an alternate method capable of decay heat removal to be established within 1 hour for each inoperable shutdown cooling loop and, with no SDC or recirculation pump in operation, CTS 3.6.O Action 2 (ITS 3.4.7 ACTION B) requires establishment of reactor coolant recirculation by an alternate method within 1 hour. If acceptable alternatives are available for decay heat removal (i.e., complying with Actions), the temporary allowances of the Notes should apply since the alternate methods must be capable of providing adequate decay heat removal.

RELOCATED SPECIFICATIONS

None

A.1 ITS 3.4.8

3.6 - LIMITING CONDITIONS FOR OPERATION

P. Shutdown Cooling - COLD SHUTDOWN

LCO 3.4.8

Two[a] shutdown cooling (SDC) loops shall be OPERABLE and, unless at least one recirculation pump is in operation, at least one shutdown cooling loop shall be in operation[b][c] ~~with each loop consisting of at least:~~

1. ~~One OPERABLE SDC pump, and~~
2. ~~One OPERABLE SDC heat exchanger.~~

LA.1

APPLICABILITY:

OPERATIONAL MODE 4.

ACTION: add proposed ACTIONS Note — A.2

ACTION A

1. With less than the above required SDC loops OPERABLE, within 1 hour and at least once per 24 hours thereafter, demonstrate the operability of at least one alternate method capable of decay heat removal for each inoperable SDC loop.

ACTION B

2. With no SDC loop or recirculation pump in operation, within 1 hour establish reactor coolant circulation by an alternate method and monitor reactor coolant temperature and pressure at least once per hour.

4.6 - SURVEILLANCE REQUIREMENTS

P. Shutdown Cooling - COLD SHUTDOWN

SR 3.4.8.1

At least one SDC loop, recirculation pump or alternate method shall be verified to be in operation and circulating reactor coolant at least once per 12 hours. LA.2

Required Action B.1

LCO Note 3 a One shutdown cooling loop may be inoperable for up to 2 hours for surveillance testing provided the other loop is OPERABLE and/in operation. L.1

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

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

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ADMINISTRATIVE

A.1 In the conversion of the Dresden 2 and 3 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 proposed ACTION Note, "Separate Condition entry is allowed for each SDC 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 SDC subsystem are proposed to be relocated to the Bases. The Bases will indicate that an OPERABLE SDC subsystem consists of an OPERABLE pump, heat exchanger, service water providing cooling to the heat exchanger, and the associated piping and valves. The details for subsystem OPERABILITY are not necessary in ITS 3.4.8. 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.

TECHNICAL CHANGES - LESS RESTRICTIVE (continued)

LA.2 The detail of the method in CTS 4.6.P of verifying operation of the SDC subsystem (circulating reactor coolant) is proposed to be relocated to the Bases. This detail is not necessary for assuring the SDC subsystem is in operation. Proposed SR 3.4.8 requires verification an SDC subsystem is circulating reactor coolant. Therefore, the relocated detail is not required to be in the ITS to provide adequate protection of the public health and safety. Changes to the Bases will be controlled by the provisions of the proposed Bases Control Program described in Chapter 5 of the ITS.

"Specific"

L.1 CTS 3.6.P footnote (a) allows one SDC loop to be inoperable for 2 hours, provided the other loop is OPERABLE and in operation. CTS 3.6.P footnote (b) allows the SDC pump to be removed from operation for up to 2 hours per 8 hour period, provided the other loop is OPERABLE. The requirements in CTS footnotes (a) and (b) (ITS 3.4.8 LCO Notes 1 and 2) are proposed to be changed to delete the "provided" requirements. The allowances of the Notes may be required even when no SDC loop remains OPERABLE or in operation. Some Surveillances result in the inoperability of both SDC loops (e.g., Surveillances on the common suction line valves). With one or more SDC loops inoperable, CTS 3.6.P Action 1 (ITS 3.4.8 ACTION A) requires an alternate method capable of decay heat removal to be established within 1 hour for each inoperable RHR shutdown cooling loop and, with no SDC or recirculation pump in operation, CTS 3.6.P, Action 2 (ITS 3.4.8 ACTION B) requires establishment of reactor coolant recirculation by an alternate method within 1 hour. If acceptable alternatives are available for decay heat removal (i.e., complying with the Actions), the temporary allowances of the Notes should apply since the alternate methods must be capable of providing adequate decay heat removal.

RELOCATED SPECIFICATIONS

None

ITS 3.4.9

A.1

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:

   a. The reactor vessel metal temperature and pressure shall be maintained within the Acceptable Regions as shown on Figure~~s~~ 3.6.K-1 ~~through 3.6.K-3~~ with the rate of change of the primary system coolant temperature ≤20°F per hour, or A.9

   b. SR 3.4.9.1 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-~~4~~ 2. A.9

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

   a. The reactor vessel metal temperature and pressure shall be maintained within the Acceptable Regions as shown on Figure 3.6.K-~~4~~ 2, and A.9

   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,

   a. SR 3.4.9.1 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-~~4~~ 2 A.9

2. SR 3.4.9.1 SR 3.4.9.2 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,

   a. The rate of change of the primary system coolant temperature to be within the limits, and L.1

   b. The reactor vessel metal temperature and pressure to be within the Acceptable Region on Figure 3.6.K-~~5~~ 3 A.9

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

PRIMARY SYSTEM BOUNDARY

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-~~5~~ 3, and

A.9

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.5

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

add proposed SR 3.4.9.7 Note

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

add proposed SR 3.4.9.6 Note

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

A.6

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.

APPLICABILITY:

At all times.

ACTION:

add proposed Conditions A and C Notes — A.2

With any of the above limits exceeded,

ACTIONS A and C

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

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.1 — L.2

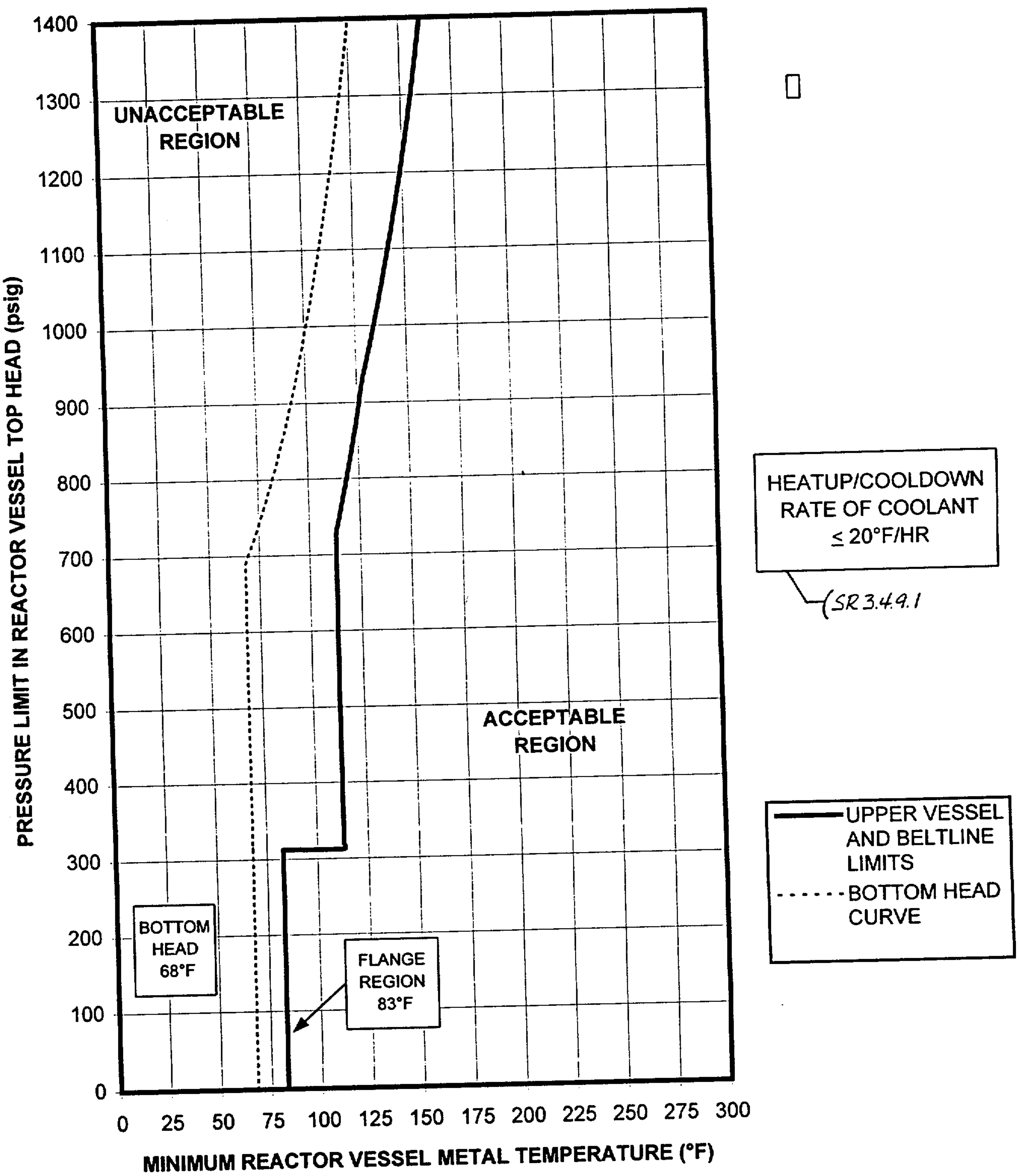
ACTION B

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

A.1

Figure 3.4.9-1

FIGURE 3.6.K-1 A.9

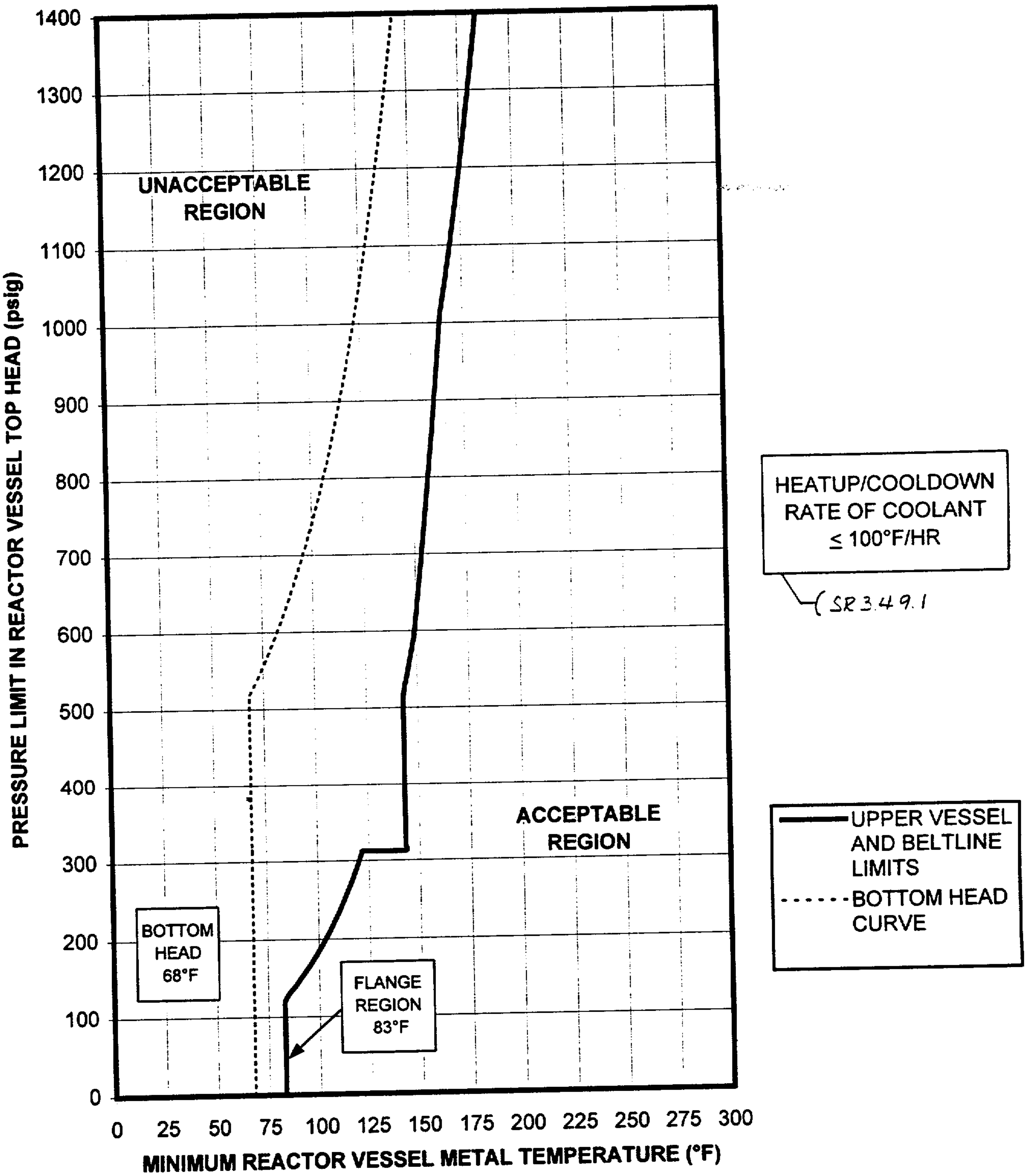


A.1

PRIMARY SYSTEM BOUNDARY

Figure 3.4.9-2

FIGURE 3.6.K-2 A.9

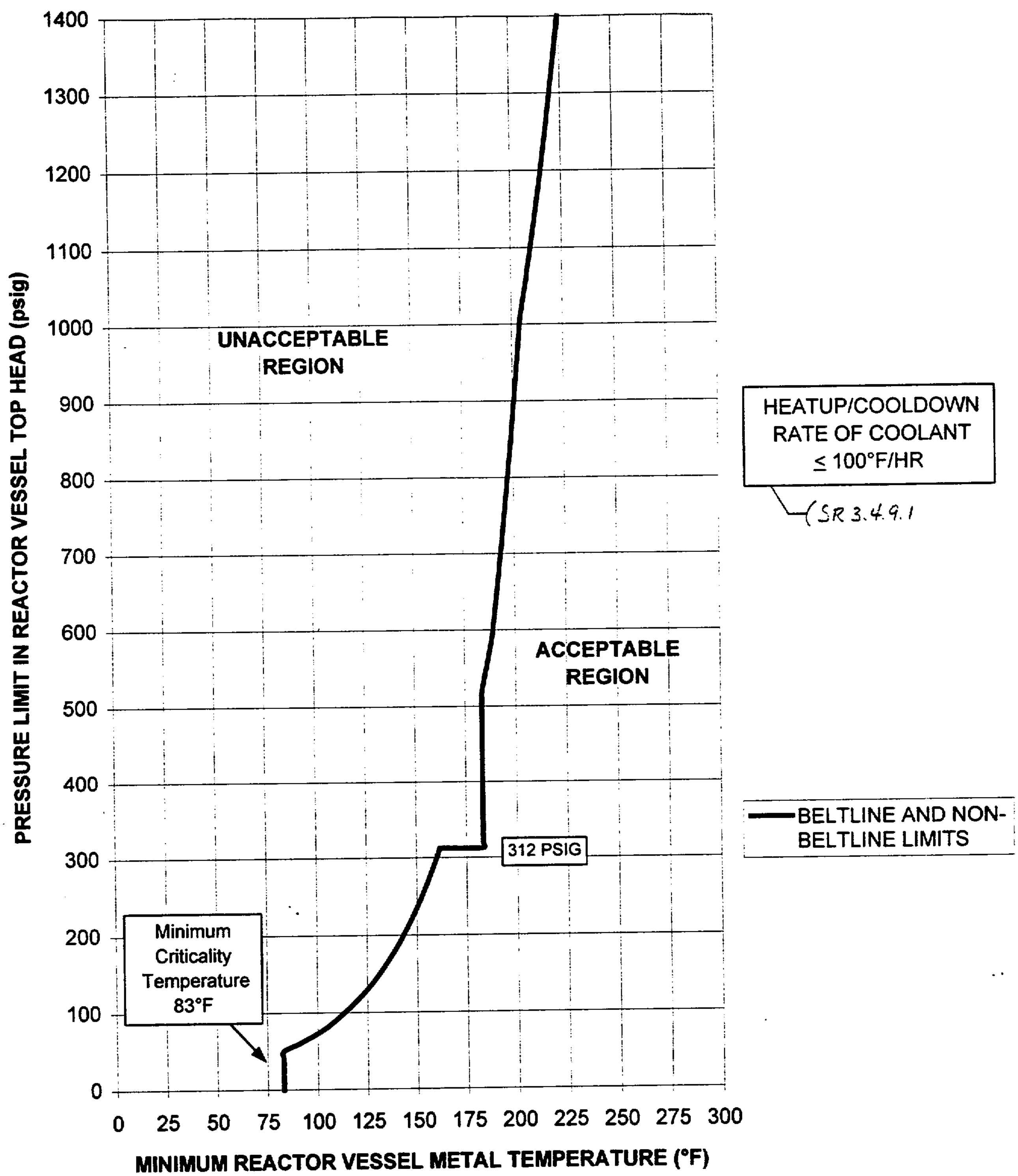


SR 3.4.9.1

A.1

Figure 3.4.9-3

FIGURE 3.6.K-3 A.9



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 and the bottom head coolant temperature is ~~within limits~~[a], ≤145°F and: M.2

SR 3.4.9.4

1. When both loops have been idle, unless the temperature differential between the reactor coolant within the idle loop to be started up and the coolant in the reactor pressure vessel is ~~within limits~~ ≤50°F, or M.1 A.7 M.1
2. When only one loop has been idle, unless the temperature differential between the reactor coolant within the idle and operating recirculation loops is ~~within limits~~ ≤50°F. A.8 M.1

APPLICABILITY:

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

ACTION: add proposed Conditions A and C Notes A.2

ACTIONS A and C — With temperature differences ~~and/or flow rates~~ exceeding the above limits, suspend startup of any recirculation loop, restore the parameter(s) to within limits within 30 minutes, and determine if the reactor coolant system is acceptable for continued operation within 72 hours. LA.2 A.3 L.2

ACTION B — Otherwise, be in HOT SHUTDOWN in 12 hours and COLD SHUTDOWN within the following 24 hours.

4.6 - SURVEILLANCE REQUIREMENTS

D. Idle Recirculation Loop Startup

The temperature differentials ~~and flow rate~~ shall be determined to be within the limits within 15 minutes prior to startup of an idle recirculation loop. LA.2

SR 3.4.9.3
SR 3.4.9.4

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

ADMINISTRATIVE

A.1 In the conversion of the Dresden 2 and 3 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.K Action 2 to "perform an engineering evaluation..."and the CTS 3.6.D Action to "determine if the Reactor Coolant System is acceptable for continued operation" are 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 CTS 3.6.K Action 1 and the CTS 3.6.D Action to "restore...within 30 minutes" is proposed to be revised to "initiate action to restore ...Immediately" for conditions other than MODES 1, 2, and 3. The existing Action would appear to provide a half hour in which pressure and temperature requirements could exceed the limits, even if capable of being returned to within limits. Also, if the parameters are incapable of being restored to within the limits within 30 minutes, the existing Action would appear to result in the requirement for an LER. The intent of the Action is believed to be more appropriately presented in ITS 3.4.9 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."

A.9 These changes to CTS 3/4.6.K are provided in the Dresden ITS consistent with the Technical Specifications Change Request submitted to the NRC for approval per ComEd letter PSLTR-00-0057, dated February 23, 2000. The changes identified revise the heatup, cooldown, and inservice test limitations for the

ADMINISTRATIVE

A.9 (cont'd) reactor pressure vessel of each unit to a maximum of 32 Effective Full Power Years. The proposed changes rely on recently approved American Society of Mechanical Engineers methodology for determining allowable pressure and temperature limits. A similar Technical Specifications amendment was recently issued for Duke Energy, Oconee Nuclear Station. As such, this change is administrative.

TECHNICAL CHANGES - MORE RESTRICTIVE

M.1 CTS LCO 3.6.D establishes the conditions for startup of an idle recirculation loop. The temperature limitations are not currently specified in the LCO since they are specified in the Dresden Administrative Technical Requirements (DATR) manual. As discussed in Discussion of Change A.7 above, the CTS 3.6.D requirements have been combined into the RCS P/T Limits Specification (ITS 3.4.9). As such, proposed ITS SRs 3.4.9.3 and 3.4.9.4 verify the temperature limitations for the startup of an idle loop have been met prior to starting the idle loop recirculation pump. The BWR ISTS, NUREG-1433, Revision 1, presentation of these SRs (NUREG SRs 3.4.10.3 and 3.4.10.4) references the Pressure and Temperature Limits Report (PTLR) to obtain the temperature limit values. Since a PTLR has not been approved by the NRC for Dresden 2 and 3, the proposed presentation of ITS SRs 3.4.9.3 and 3.4.9.4 removes references to the PTLR and includes the specific limit values as specified in the DATR. Since this change proposes to include specific limit values in the ITS, the limits will no longer be administratively controlled by ComEd, subject to the provisions of 10 CFR 50.59. Instead, the limits will be controlled by the NRC, pursuant to 10 CFR 50.90. As such, this change represents as additional restriction on plant operation and is considered a more restrictive change.

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 any reactor pressure 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 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.2 The details of the CTS 3.6.D Action and CTS 4.6.D relating to operational limits (loop flow) during a return to two recirculation pump operation from single recirculation loop operation are proposed to be relocated to the UFSAR. The single loop flow rate is considered an operational limit since it is not directly related to the ability of the system to perform its safety analysis functions. The flow rate is limited only to restrict reactor vessel internals vibration to within acceptable limits during restart of the second pump. These requirements are oriented toward maintaining long term OPERABILITY of the recirculation loops and do not necessarily have an immediate impact on their OPERABILITY. As such, the relocated details are not required to be in the ITS to provide adequate protection of the public health and safety. Changes to the UFSAR will be controlled by the provisions of the 10 CFR 50.59.

"Specific"

L.1 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-5) once within 15 minutes

TECHNICAL CHANGES - LESS RESTRICTIVE

L.1 (cont'd) 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.

L.2 CTS 3.6.K Action 2 and the CTS 3.6.D Action specify 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.

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.

APPLICABILITY: M.1

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

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.

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.

M.1

a Not applicable during anticipated transients.

ADMINISTRATIVE

A.1 In the conversion of the Dresden 2 and 3 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

CTS 3/4.6.N

R.1

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.

Page 1 of 1

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 Dresden 2 and 3 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.

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 Dresden 2 and 3 ITS Section 3.4, consistent with the BWR ISTS, NUREG-1433, Rev. 1. The revised Bases are as shown in the Dresden 2 and 3 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.

<CTS>

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.1 Recirculation Loops Operating

<3.6.A>
<3.6.A Act 1.b>
<3.6.A Act 1.c>
<3.6.A Act 1.d>
<3.6.C>

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

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

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 1

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

2

<Appl 3.6.A> APPLICABILITY: MODES 1 and 2.

ACTIONS

2 Insert ACTIONS

<3.6.A Act 1>

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

~~*Pending resolution of stability issue.~~

(continued)

<CTS>

2 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
2 <3.6.A Act 1>	D ~~B~~. Required Action and associated Completion Time of Condition ~~A~~ C not met. ~~OR~~ ~~No recirculation loops in operation.~~	D ~~B~~.1 Be in MODE 3.	12 hours

SURVEILLANCE REQUIREMENTS

	SURVEILLANCE		FREQUENCY
<3.6.C.1> <3.6.C.2> <4.6.C>	SR 3.4.1.1	--------------------NOTE-------------------- Not required to be performed until 24 hours after both recirculation loops are in operation. --- 3 Verify ~~recirculation loop~~ jet pump flow mismatch with both recirculation loops in operation is: 1 a. ≤ [10]% of rated core flow when operating at < [70]% of rated core flow; and b. ≤ [5]% of rated core flow when operating at ≥ [70]% of rated core flow.	24 hours

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.

<Appl 3.6.B> APPLICABILITY: MODES 1 and 2.

ACTIONS

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

<CTS>

SURVEILLANCE REQUIREMENTS

	SURVEILLANCE	FREQUENCY
<4.6.B> 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. -- 1 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

~~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 Dresden 2 and 3 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 Dresden 2 and 3 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 Dresden 2 and 3. 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/Relief Valves (S/RVs)

1 and

2

<3.6.E>
<3.6.F>

LCO 3.4.3 The safety function of [11] S/RVs shall be OPERABLE.

safety valves 1

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.F Act 2> 3 2 A. One [or two] [required] S/RV[s] inoperable. relief valve	A.1 Restore the [required] S/RV[s] to OPERABLE status.	14 days 2
<3.6.E Act> <3.6.F Act 2> <3.6.F Act 3> B. Required Action and associated Completion Time of Condition A not met. OR 2 One 3 [Three] or more [required] S/RVs inoperable. 1 safety valves Two or more relief valves inoperable OR	B.1 Be in MODE 3. AND B.2 Be in MODE 4.	12 hours 36 hours

<CTS>

SURVEILLANCE REQUIREMENTS

	SURVEILLANCE	FREQUENCY
<3.6.E> <4.6.E> [3] SR 3.4.3.1	Verify the safety function lift setpoints of the ~~[required]~~ ~~S/RVs~~ are as follows: [5] TSTF-29B change not adopted [1] Safety valves Number of ~~S/RVs~~ / Setpoint (psig) ~~[4] [1090 ± 32.7]~~ ~~[4] [1100 ± 33.0]~~ ~~[3] [1110 ± 33.3]~~ [2] 1 1135 ± 11.3; 2 1240 ± 12.4; 2 1250 ± 12.5; 4 1260 ± 12.6 ~~Following testing, lift settings shall be within ± 1%.~~ [4]	~~[~~In accordance with the Inservice Testing Program ~~or [18] months]~~ [2]
<DOC A.4> 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 ~~[required]~~ ~~S/RV~~ relief valve opens when manually actuated. [3] [1]	~~[18]~~ 24 months ~~[on a STAGGERED TEST BASIS for each valve solenoid]~~ [2] [3] [1]
<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 Dresden 2 and 3 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 Dresden 2 and 3 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]~~

25

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

1

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

24

<Appl 3.6.H> 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.a>	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 <3.6.H Act 3.a>	B. (continued)	B.2	Identify ~~Verify~~ source of unidentified LEAKAGE increase is not IGSCC susceptible material ~~service sensitive type 304 or type 316 austenitic stainless steel~~.	4 hours
<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 AND C.2	Be in MODE 3. 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.	12 ~~8~~ 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
	<u>AND</u>		
	A.2	Isolate the high pressure portion of the affected system from the low pressure portion by use of a second closed manual, 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
	<u>AND</u>		
	B.2	Be in MODE 4.	36 hours

1

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.

   Dresden 2 and 3 were licensed prior to 1979, and were not recipients 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 Dresden 2 and 3, and are not incorporated in the ITS. Subsequent Specifications are renumbered accordingly.

<CTS>

3.4 REACTOR COOLANT SYSTEM (RCS)

1 5

3.4.6 RCS Leakage Detection Instrumentation

<3.6.G> LCO 3.4.6 The ~~following RCS leakage detection instrumentation~~ shall be OPERABLE: 2

a. Drywell floor drain sump monitoring system; [and]

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

c. ~~Primary containment air cooler condensate flow rate monitoring system].~~ 3

<Appl 3.6.G> APPLICABILITY: MODES 1, 2, and 3.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
<3.6.G Act 2> A. Drywell floor drain sump monitoring system inoperable.	~~-----NOTE----- LCO 3.0.4 is not applicable.~~ 2 A.1 Restore drywell floor drain sump monitoring system to OPERABLE status.	TSTF-60 not adopted ~~30 days~~ 24 hours

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
B. Required primary containment atmospheric monitoring system inoperable.	-------------NOTE------------- LCO 3.0.4 is not applicable. -----------------------------	2
	B.1 Analyze grab samples of primary containment atmosphere.	Once per 12 hours
	AND	
	B.2 Restore required primary containment atmospheric monitoring system to OPERABLE status.	30 days
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 3.4.5.1.	Once per 8 hours

3

(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
2 <3.6.G Act 2> B E. Required Action and associated Completion Time of Condition A, B, [C, or D] not met. 3	B E.1 Be in MODE 3. AND E.2 Be in MODE 4. B	12 hours 36 hours
F. All required leakage detection systems inoperable.	F.1 Enter LCO 3.0.3.	Immediately. 4

3

<CTS>

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
~~SR 3.4.6.1~~	~~Perform a CHANNEL CHECK of required primary containment atmospheric monitoring system.~~	~~12 hours~~ 2
<Doc M.1> 1 L SR 3.4.~~6~~.~~2~~ 5	Perform a CHANNEL FUNCTIONAL TEST of ~~required leakage detection~~ drywell floor drain sump monitoring system instrumentation. 5	31 days
<4.6.G> 1 SR 3.4.~~6~~.~~3~~ 5 2 1	Perform a CHANNEL CALIBRATION of ~~required leakage detection~~ drywell floor drain sump monitoring system instrumentation. 5	~~[18]~~ 12 months 6

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

3. The bracketed requirement/information has been deleted since it is not applicable to Dresden 2 and 3. The following requirements have been renumbered, where applicable, to reflect this deletion.

4. The requirement to enter LCO 3.0.3 if all required leakage detection systems are inoperable is not applicable to Dresden 2 and 3. Each unit has a single leakage detection system, and its inoperability is addressed by ITS 3.4.5 ACTION A. If the Required Action and Completion Time are not met, ITS 3.4.5 ACTION B requires a plant shutdown.

5. The Dresden 2 and 3 design includes a single qualified leakage detection system, although other methods of RCS leakage detection are available. The words, "required leakage detection," in ITS SRs 3.4.5.1 and 3.4.5.2 have been replaced with the qualified detection system name, "drywell floor drain sump monitoring system" for clarification and to provide consistency with the proposed changes to the LCO and ACTIONS.

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

<CTS>

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.~~7~~ 6 RCS Specific Activity — [1]

<3.6.J> LCO 3.4.~~7~~ 6 The specific activity of the reactor coolant shall be limited to DOSE EQUIVALENT I-131 specific activity ≤ [0.2] 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
OR	AND	
Reactor Coolant specific activity > [4.0] μCi/gm Dose EQUIVALENT I-131. [2]	B.2.1 Isolate all main steam lines.	12 hours
	OR	
		(continued)

<CTS>

ACTIONS

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

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
<4.6.J> 1 SR 3.4.7.1 6	---NOTE--- Only required to be performed in MODE 1. --- Verify reactor coolant DOSE EQUIVALENT I-131 specific activity is ≤ [0.2] μCi/gm.	7 days 2

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.

all changes are [1] unless otherwise indicated

<CTS>

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.~~8~~ 7 ~~Residual Heat Removal (RHR)~~ Shutdown Cooling (SDC) System—Hot Shutdown

<3.6.D> LCO 3.4.~~8~~ 7 Two required ~~RHR shutdown cooling~~ SDC subsystems shall be OPERABLE, and, with no recirculation pump in operation, at least one ~~RHR shutdown cooling~~ required SDC subsystem shall be in operation.

NOTES

1. Both required ~~RHR shutdown cooling~~ SDC subsystems and recirculation pumps may ~~be removed from~~ not be in operation for up to 2 hours per 8 hour period. TSTF-153 [4]

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

<Appl 3.6.D> APPLICABILITY: MODE 3, with reactor ~~steam dome pressure~~ vessel coolant temperature ~~≥~~ less than ~~[~~the ~~RHR~~ SDC cut-in permissive ~~pressure]~~ temperature. [3]

ACTIONS

NOTES

1. LCO 3.0.4 is not applicable.

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

CONDITION	REQUIRED ACTION	COMPLETION TIME
<3.6.D Act 1> A. One or two required ~~RHR shutdown cooling~~ SDC subsystems inoperable.	A.1 Initiate action to restore ~~RHR shutdown cooling~~ SDC subsystem(s) to OPERABLE status. AND	Immediately (continued)

all changes are 1 unless otherwise indicated

<CTS>

ACTIONS

	CONDITION	REQUIRED ACTION	COMPLETION TIME
<3.6.0 Act 1>	A. (continued)	A.2 Verify an alternate method of decay heat removal is available for each inoperable required SDC ~~RHR shutdown cooling~~ subsystem.	1 hour
		AND	
		A.3 Be in MODE 4.	24 hours
<4.6.0> <3.6.0 Act 2>	B. No required SDC ~~RHR shutdown cooling~~ subsystem in operation. AND No recirculation pump in operation.	B.1 Initiate action to restore one required SDC ~~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

all changes are 1 unless otherwise indicated

<CTS>

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
2 7 <4.6.0> SR 3.4.8.1	-------------------NOTE------------------- Not required to be met until 2 hours after reactor ~~steam dome pressure~~ vessel coolant temperature is ≤ less than the ~~RHR~~ SDC cut in permissive ~~pressure~~ temperature. ---	3
	Verify one ~~RHR shutdown cooling~~ SDC subsystem or recirculation pump is operating.	12 hours

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

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

4. TSTF-153 revised the RHR Shutdown Cooling System-Hot Shutdown LCO (ISTS LCO 3.4.8) Note 1, which provides an exception to the requirement for the required pump to be in operation, to provide a clarification of the intent of the Note consistent with the requirement being excepted. The justification for TSTF-153 described that the change was necessary to eliminate ambiguity that could lead to errors or improper enforcement. However, the change can now lead to a misinterpretation of the allowance of the Note. Specifically, the Note can now be interpreted as <u>requiring</u> the required subsystems or pumps to not be in operation for up to 2 hours per 8 hour period, i.e., they must be taken out of operation. The intent of the Note (as described in the associated Bases) is to <u>allow (but not require)</u> the required subsystems or pumps to not be in operation for up to 2 hours per 8 hour period. Therefore, the Note is revised to allow the subsystems or pumps to be "not in operation" for up to 2 hours per 8 hour period.

all changes are 1 *unless otherwise indicated*

<CTS>

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.9 8 Residual Heat Removal (RHR) Shutdown Cooling (SDC) System—Cold Shutdown

2

<3.6.P> LCO 3.4.9 8 Two required RHR shutdown cooling SDC subsystems shall be OPERABLE, and, with no recirculation pump in operation, at least one RHR shutdown cooling SDC subsystem shall be in operation.

NOTES

3 — 2 ~~1~~. Both required ~~RHR shutdown cooling~~ SDC subsystems and recirculation pumps may ~~be removed from~~ not be in operation for up to 2 hours per 8 hour period. TSTF-153 4

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

1. Both required SDC subsystems may be not in operation during hydrostatic testing. 3

<Appl 3.6.P> APPLICABILITY: MODE 4.

ACTIONS

NOTE

Separate Condition entry is allowed for each shutdown cooling subsystem.

CONDITION	REQUIRED ACTION	COMPLETION TIME
<3.6.P Act 1> A. One or two required ~~RHR shutdown cooling~~ SDC subsystems inoperable.	A.1 Verify an alternate method of decay heat removal is available for each inoperable required ~~RHR shutdown cooling~~ SDC subsystem.	1 hour AND Once per 24 hours thereafter

(continued)

all changes are 1 unless otherwise indicated

<CTS>

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
SDC <3.6.P Act 2> <4.6.P> B. No RHR shutdown cooling subsystem in operation. AND No recirculation pump in operation.	B.1 Verify reactor coolant circulating by an alternate method.	1 hour from discovery of no reactor coolant circulation AND Once per 12 hours thereafter
	AND B.2 Monitor reactor coolant temperature.	Once per hour

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
SDC <4.6.P> SR 3.4.8.1 (2 8) Verify one RHR shutdown cooling subsystem or recirculation pump is operating.	12 hours

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

3. Note 1 to ITS LCO 3.4.8 has been added consistent with the current licensing basis. The subsequent Notes have been renumbered to reflect this addition.

4. TSTF-153 revised the RHR Shutdown Cooling System-Cold Shutdown LCO (ISTS LCO 3.4.9) Note 1, which provides an exception to the requirement for the required pump to be in operation, to provide a clarification of the intent of the Note consistent with the requirement being excepted. The justification for TSTF-153 described that the change was necessary to eliminate ambiguity that could lead to errors or improper enforcement. However, the change can now lead to a misinterpretation of the allowance of the Note. Specifically, the Note can now be interpreted as <u>requiring</u> the required subsystems or pumps to not be in operation for up to 2 hours per 8 hour period, i.e., they must be taken out of operation. The intent of the Note (as described in the associated Bases) is to <u>allow (but not require)</u> the required subsystems or pumps to not be in operation for up to 2 hours per 8 hour period. Therefore, the Note is revised to allow the subsystems or pumps to be "not in operation" for up to 2 hours per 8 hour period.

<CTS>

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.10 RCS Pressure and Temperature (P/T) Limits

1 9

<3.6.K> 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

<Appl 3.6.K> APPLICABILITY: At all times.

ACTIONS

	CONDITION	REQUIRED ACTION	COMPLETION TIME
<3.6.K Act 1> <3.6.K Act 2> <3.6.D Act>	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. 3 or	A.1 Restore parameter(s) to within limits. AND A.2 Determine RCS is acceptable for continued operation.	30 minutes 72 hours
<3.6.K Act 3> <3.6.D Act>	B. Required Action and associated Completion Time of Condition A not met.	B.1 Be in MODE 3. AND B.2 Be in MODE 4.	12 hours 36 hours

(continued)

<CTS>

ACTIONS (continued)

	CONDITION	REQUIRED ACTION	COMPLETION TIME
<3.6.K Act 1> <3.6.K Act 2> <3.6.D Act>	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 <3.6.K.1> <3.6.K.2> <4.6.K.1> <4.6.K.2> <3.6.K.3.b> 2	SR 3.4.~~10~~.1	-------------------NOTE------------------- Only required to be performed during RCS heatup and cooldown operations and RCS inservice leak and hydrostatic testing. -- : a. Verify ~~Verify~~ RCS pressure ~~,~~ and 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
1 9 <4.6.K.2> <3.6.K.3.a> 2	SR 3.4.~~10~~.2	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

(continued)

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 temperature and pressure are being maintained within the limits of Figure 3.4.9-1. 2

<CTS>

SURVEILLANCE REQUIREMENTS (continued)

	SURVEILLANCE	FREQUENCY
1 9 <3.6.D> TSTF-35 <4.6.D>	SR 3.4.~~10~~.3 ------NOTE------ Only required to be met in MODES 1, 2, 3, and 4 during recirculation pump startup ~~with reactor steam dome pressure ≥ 25 psig~~. 4	
2	Verify the difference between the bottom head coolant temperature and the reactor pressure vessel (RPV) coolant temperature is ~~within the limits specified in the PTLR~~ ≤ 145°F.	Once within 15 minutes prior to each startup of a recirculation pump
1 9 <3.6.D.1> TSTF-35 <3.6.D.2> <4.6.D>	SR 3.4.~~10~~.4 ------NOTE------ Only required to be met in MODES 1, 2, 3, and 4 during recirculation pump startup.	
2 6 TSTF-353 changes not adopted	Verify the difference between the reactor coolant temperature in the recirculation loop to be started and the RPV coolant temperature is ~~within the limits specified in the PTLR~~ ≤ 50°F.	Once within 15 minutes prior to each startup of a recirculation pump
1 9 <3.6.K.4> <4.6.K.4.6>	SR 3.4.~~10~~.5 ------NOTE------ Only required to be performed when tensioning the reactor vessel head bolting studs.	
2	Verify reactor vessel flange and head flange temperatures are ~~within the limits specified in the PTLR~~ ≥ 83°F.	30 minutes

(continued)

<CTS>

SURVEILLANCE REQUIREMENTS (continued)

		SURVEILLANCE	FREQUENCY
1 9 <3.6.K.4> <4.6.K.4.a.2)> 5 2	SR 3.4.~~10~~.6	---NOTE--- Not required to be performed until 30 minutes after RCS temperature ≤ ~~80~~ 93 °F in MODE 4. --- Verify reactor vessel flange and head flange temperatures are ~~within the limits specified in the PTLR~~ ≥ 83°F.	30 minutes
1 9 <3.6.K.4> <4.6.K.4.a.1)> 5 2	SR 3.4.~~10~~.7	---NOTE--- Not required to be performed until 12 hours after RCS temperature ≤ ~~100~~ 113 °F in MODE 4. --- Verify reactor vessel flange and head flange temperatures are ~~within the limits specified in the PTLR~~ ≥ 83°F.	12 hours

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

<CTS>

<Fig. 3.6.K-1>

Insert Figure 3.4.9-1

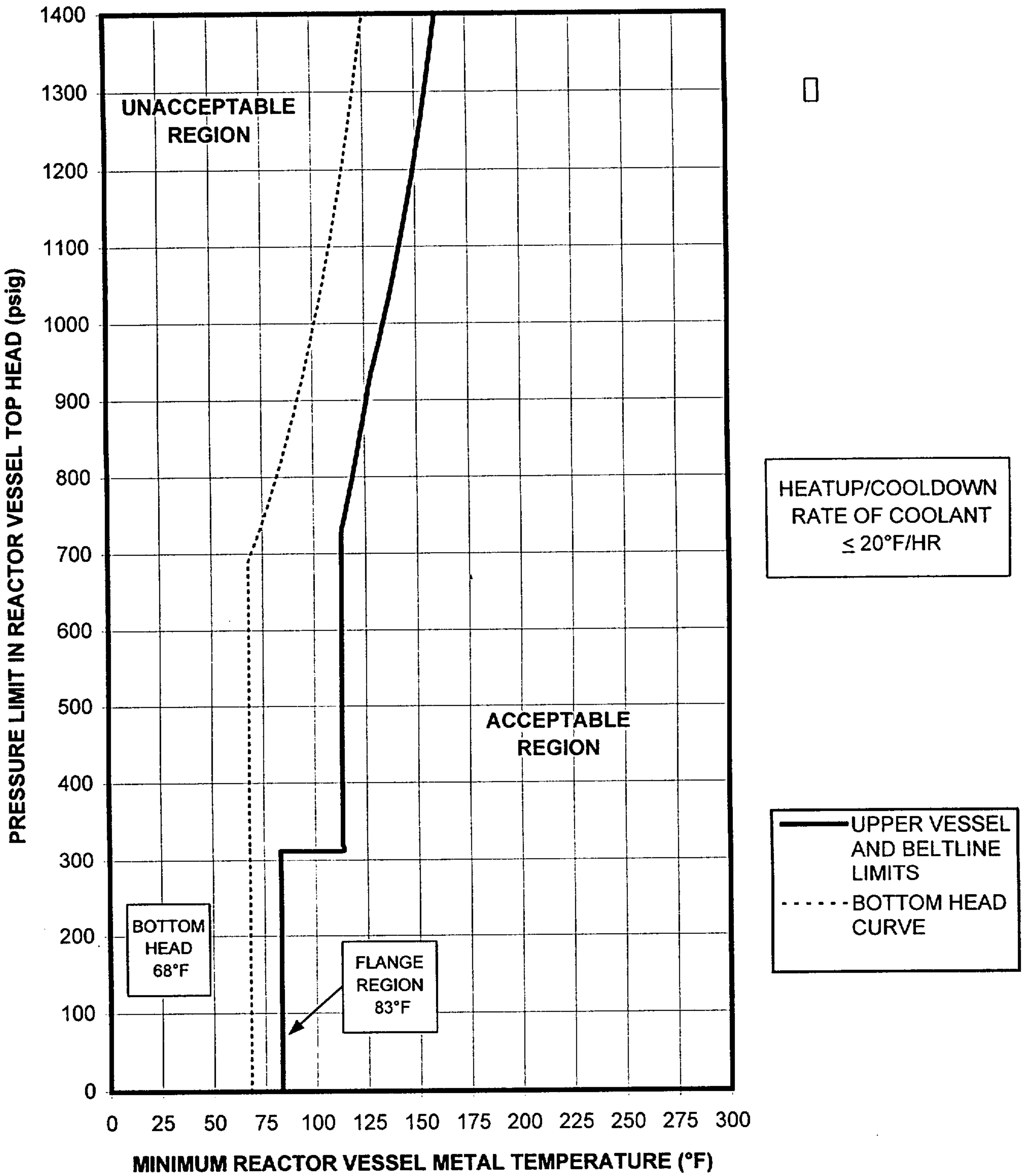


Figure 3.4.9-1 (Page 1 of 1)
Non-Nuclear Inservice Leak and Hydrostatic Testing Curve
(Valid to 32 EFPY)

<CTS>

<Fig. 3.6.K-2>

Insert Figure 3.4.9-2

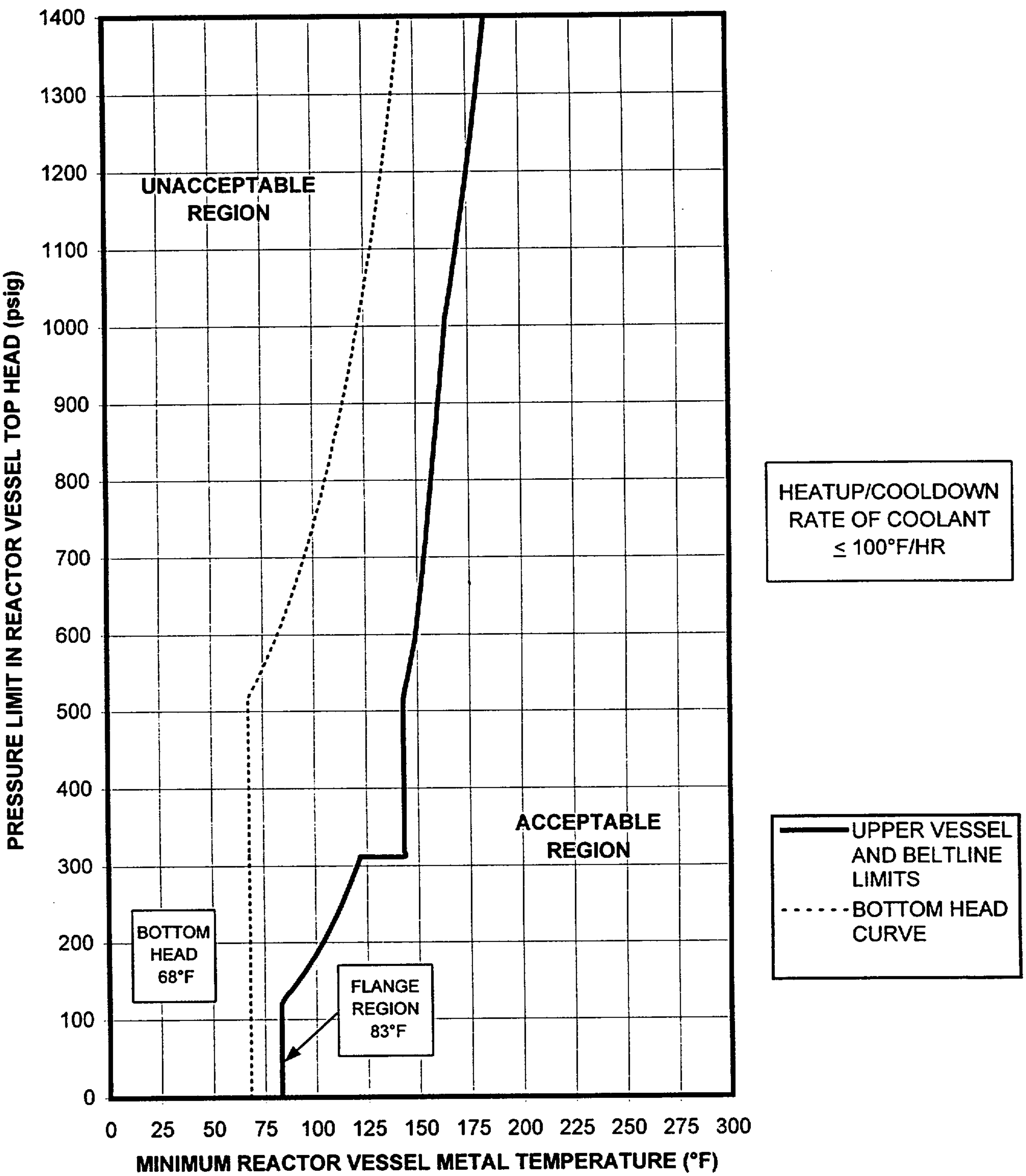


Figure 3.4.9-2 (Page 1 of 1)
Non-Nuclear Heatup/Cooldown Curve
(Valid to 32 EFPY)

<CTS>

<Fig. 3.6.K-3>

Insert Figure 3.4.9-3

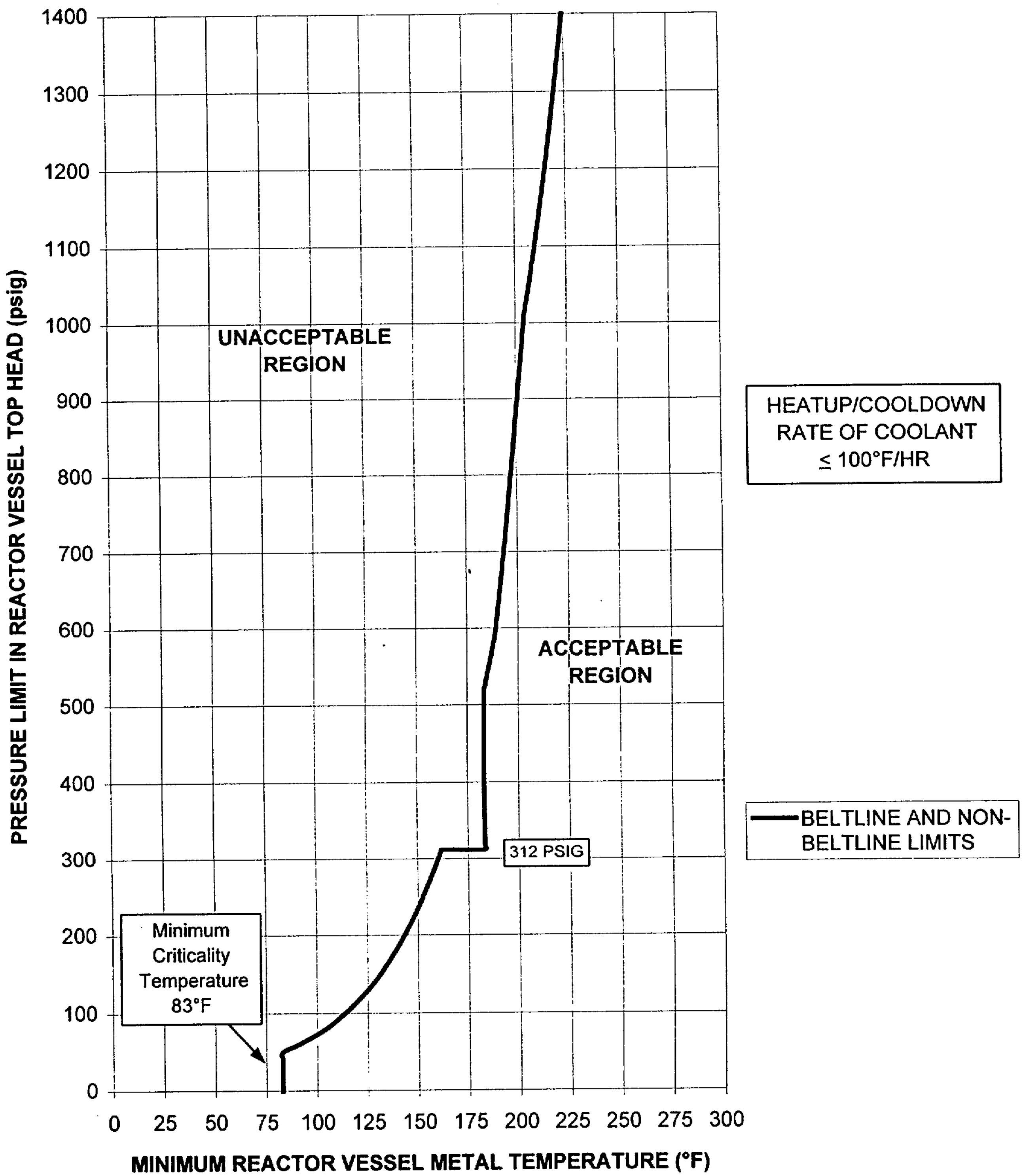


Figure 3.4.9-3 (Page 1 of 1)
Critical Operations Curve
(Valid to 32 EFPY)

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, Dresden 2 and 3 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 Dresden 2 and 3 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 Dresden 2 and 3 Technical Specifications (i.e., current licensing basis does not include these Surveillances). The coolant temperature verifications are only required in the CTS when starting an idle recirculation pump, and these verifications have been maintained in the ITS.

<CTS>

O 1

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.10 Reactor Steam Dome Pressure

1 O

1005 2

<3.6.L> LCO 3.4.10 The reactor steam dome pressure shall be ≤ [1020] psig.

<Appl 3.6.L> APPLICABILITY: MODES 1 and 2.

ACTIONS

	CONDITION	REQUIRED ACTION	COMPLETION TIME
<3.6.L Act>	A. Reactor steam dome pressure not within limit.	A.1 Restore reactor steam dome pressure to within limit.	15 minutes
<3.6.L Act>	B. Required Action and associated Completion Time not met.	B.1 Be in MODE 3.	12 hours

SURVEILLANCE REQUIREMENTS

	SURVEILLANCE		FREQUENCY
<4.6.L>	SR 3.4.10.1	Verify reactor steam dome pressure is ≤ [1020] psig.	12 hours

1 O 1005 2

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

1

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 (at a faster rate) 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.

2 2 1

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

1

(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. ~~2~~ 1). [1]

[2] The transient analyses ~~of~~ in Chapter 15 of the U 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~~ and the Rod Block Monitor [3] ~~setpoints~~ Allowable Values [4] is also required to account for the different relationships between recirculation drive flow and reactor core flow. The APLHGR and MCPR ~~setpoints~~ limits for single loop operation are specified in the COLR. The APRM ~~flow biased simulated THERMAL POWER setpoint~~ Neutron Flux-High Allowable Value is in LCO 3.3.1.1, "Reactor Protection System (RPS) Instrumentation." [4] The Rod Block Monitor - Upscale Allowable Value is in LCO 3.3.2.1, "Control Rod Block Instrumentation." [3]

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 [4] 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~~ Alternatively, with [4] 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~~ [3] APRM ~~Flow Biased Simulated Thermal Power~~ Neutron Flux—High ~~setpoint~~ Allowable Value (LCO 3.3.1.1), and the Rod Block Monitor - Upscale Allowable Value (LCO 3.3.2.1) [3] ~~may~~ must [4] be applied to allow continued operation consistent with the assumptions of Reference ~~2~~ 1. [1]

(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.

3 Insert ACTIONS

ACTIONS

A.1 B.1 and C.1

for reasons other than Condition A or B (e.g., one loop "not in operation") 3

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.

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 for greater than 2 hours (i.e., Required Action B.1 has been taken). ~~The loop with the lower flow must be considered not in operation.~~ 3 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 ~~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. 4

3

The 2 hour and 24 hour Completion Times ~~is~~ 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.

3 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)

$\boxed{3}$ 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)

D 3

B.1

With

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

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 ~~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. 6 3

The mismatch is measured in terms of percent of rated core flow. If the flow mismatch exceeds the specified limits, the loop with the lower flow is considered ~~inoperable~~ not in operation. 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. 3 5

(continued)

BASES (continued)

1 U 3 6

REFERENCES

1. UFSAR, Section 6.3.3.

2. UFSAR, ~~Section [5.5.1.4]~~. Chapter 15

~~3. [Plant specific analysis for single loop operation.]~~

3. UFSAR, Section 15.3.1.

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

4. Changes have been made to more closely match the LCO requirement.

5. Typographical/grammatical error corrected.

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

B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.2 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.

1

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. The total flow then passes through the jet pump diffuser section into the area below the core (lower plenum), gaining sufficient head in the process to drive the required flow upward through the core.

1

and resulting partial pressure recovery

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~~ 3 of ~~the NRC Policy Statement~~ (10 CFR 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

[3] An inoperable jet pump can increase the blowdown area and reduce the capability ~~of~~ to reflood~~ing~~ 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

(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.

5 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
4 may nozzles. A change in the relationship indicate~~s~~ 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

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. 5

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 ~~when~~ until 24 hours after 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. The 24 hours is an acceptable time to establish conditions appropriate to perform this SR. 7 7 3

(continued)

BASES (continued)

REFERENCES

1. FSAR, Section [6.3].

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

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

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 ~~/~~ and Relief Valves ~~(S/RVs)~~

Each unit is designed with nine safety valves, one of which 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~~ 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).

The Safety valves and S/RVs are located on the main steam lines between the reactor vessel and the first isolation valve within the drywell. The ~~S/RVs~~ Safety valves ~~can~~ actuate ~~by either of two modes: the safety mode or the relief mode.~~ [2] 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.~~ Insert BKGD-1 This satisfies the Code requirement. Insert BKGD-2 [2]

The safety valves discharge directly to the drywell.

~~Each~~ The relief valve and S/RV discharge~~s~~ steam through a discharge line to a point below the minimum water level in the suppression pool. Insert BKGD-3 ~~The S/RVs that provide the relief mode are the low-low set (LLS)~~ Two of the five relief valves are the low set relief valves and ~~the~~ all of the relief valves, including the S/RV, are Automatic Depressurization System (ADS) valves. The ~~LLS~~ low set relief requirements are specified in LCO 3.6.1.6, "~~Low-Low Set (LLS) Valves~~ Low Set Relief 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, ~~[11]~~ eight ~~S/RVs~~ Safety valves are assumed to operate in the safety mode. The relief valves and S/RV are not credited to function during this event. [2] The analysis results demonstrate that the design ~~S/RV~~ Safety valve 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 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. 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. 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.

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.

2 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. 4 2

Safety and relief valves 1

1 Safety and relief valves

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

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

LCO

eight 1

safety valves

1 Safety valve

The safety function of (11) 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).

2

1 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.

2 U

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.

2 Insert LCO

APPLICABILITY

1 eight safety valves (not including the S/RV) and five relief valves (including the S/RV)

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.

safety and relief valves 1

Shutdown Cooling (SDC) 2

(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.

2

BASES

Shutdown Cooling

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.

2 and MCPR

are 1 s

Safety and relief are

1

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.

3

3

2

1 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 1

B.1 and B.2

safety valves 1

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

1 reliefs

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

One 3

safety valves 1

(continued)

Safety and Relief Valves 1

S/RVs

BASES (continued)

SURVEILLANCE REQUIREMENTS

SR 3.4.3.1

1 Safety valves, including the S/RV, Safety valve and

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[s], 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 setpoint is ± [3]% for OPERABILITY; however, the valves are reset to ± 1% during the Surveillance to allow for drift.

4 ty 3 1 s are 1 TSTF-298 change not adopted

The 18 month Frequency was selected because this Surveillance must be performed during shutdown conditions and is based on the time between refuelings. 2

SR 3.4.3.2

relief valve, including the 1

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/RVs divert steam flow upon opening. Sufficient time is therefore allowed after the required pressure and flow are achieved to perform this test. Adequate pressure at which this test is to be performed is [920] psig (the pressure recommended by the valve manufacturer). Adequate steam flow is represented by [at least 1.25] turbine bypass valves open[, or total steam flow ≥ 10^6 lb/hr]. [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

1 relief valve or the S/RV s 1

3 300 2.0 4 9 Unit is 2 (Ref. 5) 2

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.4.3.2 (continued)

conditions for testing and provides a reasonable time to complete the SR. If a valve 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.

1 the S/RV

3 24

The [18] month on a STAGGERED TEST BASIS Frequency ensures that each solenoid for each S/RV is alternately tested. The 18 month Frequency was developed based on the S/RV tests required by the ASME Boiler and Pressure Vessel Code, Section XI (Ref. 3). Operating experience has shown that these components usually pass the Surveillance when performed at the 18 month Frequency. Therefore, the Frequency was concluded to be acceptable from a reliability standpoint.

1 24 relief valve 1 5 2

Insert SR 3.4.3.3 1

REFERENCES

2 3 U

1. FSAR, Section [5.2.2.2.4].
4 2. FSAR, Section [15]. Chapter 2
5 3. ASME, Boiler and Pressure Vessel Code, Section XI.

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

1 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~~ UFSAR, Section 3.1.2.4.1 (Ref~~s~~. 1~~, 2, and 3~~). 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 ~~(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. <u>Pressure Boundary LEAKAGE</u>

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

(continued)

BASES

LCO (continued)

b. Unidentified LEAKAGE

The 5 gpm of unidentified LEAKAGE is allowed as a reasonable minimum detectable amount that the ~~containment air monitoring,~~ drywell floor drain sump ~~level~~ flow rate 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. [1]

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

[2] An unidentified LEAKAGE increase of > 2 gpm within the previous 24 ~~[4]~~ hour period indicates a potential flaw in the RCPB and must be quickly evaluated to determine the source and extent of the LEAKAGE. The increase is measured relative to the steady state value; temporary changes in LEAKAGE rate as a result of transient conditions (e.g., startup) are not considered. As such, the 2 gpm increase limit is only applicable in MODE 1 when operating pressures and temperatures are established. Violation of this LCO could result in continued degradation of the RCPB.

APPLICABILITY

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

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

(continued)

BASES (continued)

ACTIONS

A.1

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

B.1 and B.2

24 3

An unidentified LEAKAGE increase of > 2 gpm within a ~~4~~ hour period is an indication of a potential flaw in the RCPB and must be quickly evaluated. Although the increase does not necessarily violate the absolute unidentified LEAKAGE limit, certain susceptible components must be determined not to be the source of the LEAKAGE increase within the required Completion Time. For an unidentified LEAKAGE increase greater than required limits, an alternative to reducing LEAKAGE increase to within limits (i.e., reducing the LEAKAGE rate such that the current rate is less than the "2 gpm increase in the previous ~~[4]~~ hours" limit; either by isolating the source or other possible methods) is to ~~evaluate 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.~~ 24 2

3 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~~ 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.

1

SURVEILLANCE REQUIREMENTS

SR 3.4.4.1

an alternate method which may be used to quantify LEAKAGE is calculating flowrates using sump pump run times

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." 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, an 8 hour Frequency for this Surveillance is appropriate for identifying LEAKAGE and for tracking required trends (Ref. 6).

3 5 12 4 1 4

REFERENCES

1. ~~10 CFR 50, Appendix A, GDC 30~~. UFSAR, Section 3.1.2.4.1
2. GEAP-5620, "Failure Behavior in ASTM A106B Pipes Containing Axial Through-Wall Flaws," April 1968.
3. NUREG-76/067, "Investigation and Evaluation of Cracking in Austenitic Stainless Steel Piping of Boiling Water Reactor Plants," October 1975.
4. ~~FSAR, Section [5.2.7.5.2]~~.
5. ~~Regulatory Guide 1.45~~.
6. Generic Letter 88-01, Supplement 1, February 1992.

4 1 4 5 1

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)

1

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)

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.

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

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)

1

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.

B 3.4 REACTOR COOLANT SYSTEM (RCS)

1 5
B 3.4.~~6~~ RCS Leakage Detection Instrumentation

BASES

BACKGROUND

2 UFSAR, Section 3.1.2.4.1

~~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.

2 measuring flow from the drywell floor drain sump

2 Although alternate methods of detecting RCS LEAKAGE are available, the sole

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 Reactor Building

2 Leakage into the drywell floor drain sump is pumped through a piping header that penetrates the containment wall to the floor drain collector tank.

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 Reactor Building 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.~~

(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. 2 transmitter

2 Insert BKGD 1 A flow indicator in the discharge line of the drywell floor drain sump pumps provides flow indication in the control

Insert BKGD 2 room. The pumps can also be started from the control room.

The primary containment air 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).

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. 3

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 3 2 and 5). Each of the leakage detection systems inside the 2 4 drywell is designed with the capability of detecting LEAKAGE less than the established LEAKAGE rate limits and providing appropriate alarm of excess LEAKAGE in the control room.

A control room alarm allows the operators to evaluate the significance of the indicated LEAKAGE and, if necessary, shut down the reactor for further investigation and corrective action. The allowed LEAKAGE rates are well below 5 2 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)

$\boxed{2}$ INSERT BKGD 1

Two drywell floor drain sump pumps take suction from the drywell floor drain sump and discharge to the Liquid Radioactive Waste Management Systems. The pumps alternate as lead and backup on each successive start. When a high level is reached in the floor drain sump, a level switch actuates to start the lead floor drain sump pump when the pump discharge valves are open. In the event the level continues to rise, a second level switch actuates to start the backup floor drain sump pump and initiates an alarm in the control room. When the level decreases to a low level, both floor drain sump pumps are stopped.

$\boxed{2}$ INSERT BKGD 2

In addition, a leak rate recorder is provided capable of identifying a 1 gpm change over an 8 hour period.

BASES

APPLICABLE SAFETY ANALYSES (continued)

RCS leakage detection instrumentation satisfies Criterion 1 of ~~the NRC Policy Statement.~~ 10 CFR 50.36(c)(2)(ii) [2]

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 (e.g., particulate, temperature) [2] are available [2] ~~provide early alarms~~ to the operators so closer examination can ~~of other detection systems will~~ be made to determine the extent of any corrective action that may be required. With the ~~leakage detection systems~~ drywell floor drain sump monitoring system [1] inoperable, monitoring for LEAKAGE in the RCPB is degraded.

APPLICABILITY

In MODES 1, 2, and 3, ~~leakage detection systems are~~ the drywell floor drain sump monitoring system is [1] 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]~~ [3] other monitoring systems are available that [2] 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 ~~8~~ 12 [1] hours (SR 3.4.4.1), operation may continue for ~~30 days~~ 24 hours [1]. The ~~30 day~~ 24 hour [1] Completion Time of Required Action A.1 is acceptable, based on operating experience, considering the ~~multiple~~ alternative [4] 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.~~ [1]

(continued)

BASES

ACTIONS (continued)

B.1 and B.2

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.] 1

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.

C.1

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. 1 3

(continued)

BASES

ACTIONS (continued)

~~D.1 and D.2~~

~~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.~~ 1 3

~~The Required Actions are modified by a Note that states that the provisions of LCO 3.0.4 are not applicable. As a result, a MODE change is allowed when both the gaseous and particulate primary containment atmospheric monitoring channels and air cooler condensate flow rate are inoperable. This allowance is provided because other instrumentation is available to monitor RCS leakage.~~

1 B

E.1 and E.2

4 the and 1

If ~~any~~ Required Action of Condition A, ~~B, [C, or D]~~ cannot be met ~~within the~~ associated Completion Time, the plant must be brought to a MODE in which the LCO does not apply. To achieve this status, the plant must be brought to at least MODE 3 within 12 hours and 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.

~~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.~~ 1

(continued)

BASES (continued)

SURVEILLANCE REQUIREMENTS

SR 3.4.6.1

~~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.~~ [1]

SR 3.4.6.2 (5) (1)

This SR is for the performance of a CHANNEL FUNCTIONAL TEST of the ~~required RCS leakage detection~~ drywell floor drain sump monitoring system instrumentation. The test ensures that the ~~monitors~~ system can perform ~~their~~ its function in the desired manner. The test also verifies the ~~alarm setpoint and~~ relative accuracy of the instrument string. The Frequency of 31 days considers instrument reliability, and operating experience has shown it proper for detecting degradation. [1] [2]

SR 3.4.6.3 (5) (2) [1]

This SR is for the performance of a CHANNEL CALIBRATION of ~~required leakage detection~~ the drywell floor drain sump monitoring system instrumentation channel~~s~~ (i.e., drywell floor drain sump pump discharge flow integrator). 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.~~ SR 3.4.5.2 is based on the assumption of a 12 month calibration interval in the determination of the magnitude of equipment drift in the setpoint analysis. [1] [2]

REFERENCES

1. ~~10 CFR 50, Appendix A, GDC 30~~ UFSAR, Section 3.1.2.4.1. [2]
2. Regulatory Guide 1.45, May 1973.
~~3. FSAR, Section [5.2.7.2.1].~~ [2]
3. GEAP-5620, "Failure Behavior in ASTM A106B Pipes Containing Axial Through-Wall Flaws," April 1968. (3) [4]
4. NUREG-75/067, "Investigation and Evaluation of Cracking in Austenitic Stainless Steel Piping of Boiling Water Reactor Plants," October 1975. (4)
5. UFSAR, Section ~~[5.2.7.5.2]~~ 5.2.5.6.4. (5) [5]

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 . Dresden 2 and 3. 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 3.4.6 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. 2

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 2

(continued)

BASES

Insert ASA 2

APPLICABLE SAFETY ANALYSES (continued)

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

The limit~~s~~ on specific activity ~~are~~ is a value~~s~~ from a parametric evaluation of typical site locations. ~~These~~ limit~~s~~ ~~are~~ is conservative because the evaluation considered more restrictive parameters than for a specific site, such as the location of the site boundary and the meteorological conditions of the site. 3

RCS specific activity satisfies Criterion 2 of ~~the NRC Policy Statement~~ 10 CFR 50.36 (c)(2)(ii). 2

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 and GDC 19 of 10 CFR 50, Appendix A (Ref. 3). 4 2

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)

$\boxed{2}$ Insert ASA

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).

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 10 CFR 50 Appendix A (Ref. 3)) during a postulated MSLB accident. 2

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

6 1

SR 3.4.~~7~~.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 U 2. FSAR, Section ~~[~~15.~~[.40]~~ 6.4. 4

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.

all changes are [1] unless otherwise indicated

B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.~~8~~ 7 ~~Residual Heat Removal (RHR)~~ Shutdown Cooling (SDC) 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~~ 212°F. ~~This decay heat removal is~~ the decay heat must be removed for [2] in preparation for performing ~~refueling or~~ Cold Shutdown maintenance operations, or ~~for~~ ~~keeping~~ maintaining the reactor in the Hot Shutdown condition. [3]

The ~~two~~ three [3] redundant, ~~manually~~ ~~controlled~~ shutdown cooling subsystems ~~of the RHR System~~ SDC one [3] provide decay heat removal. Each [3] loop consists of ~~two~~ one motor driven pump~~s~~, a heat exchanger, and associated piping and valves. ~~Both~~ loop~~s~~ (loops [2]) ~~have~~ has 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~~ SDC heat exchangers transfer heat to the ~~RHR Service Water System (LCO 3.7.1, "Residual Heat Removal Service Water (RHRSW) System")~~ via the Reactor Building Closed Cooling Water (RBCCW) System [3].

APPLICABLE SAFETY ANALYSES

Decay heat removal by operation of the ~~RHR~~ SDC 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 (5) 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~~ SDC System [2] (9) (4) is retained as a Technical Specification. 10 CFR 50.36(c)(2)(ii) [4]

LCO

Two ~~RHR shutdown cooling~~ SDC subsystems are required to be OPERABLE, and when no recirculation pump is in operation, one ~~shutdown cooling~~ SDC subsystem must be in operation. An OPERABLE ~~RHR shutdown cooling~~ SDC subsystem consists of one OPERABLE ~~RHR~~ SDC pump, one heat exchanger, ~~and~~ the associated piping and valves, and the necessary portions of the RBCCW System capable of providing cooling water to the heat exchanger and SDC pump seal cooler. The ~~two~~ subsystems have a common suction source and ~~are allowed to have a common heat exchanger and~~ [3]

(continued)

all changes are 1 unless otherwise indicated

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.

is a — It is — is — 3 — SDC — 2 — TSTF -153 — and recirculation pumps

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.

SDC — not be in operation

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.

5 — temperature — vessel coolant temperature — SDC — temperature — SDC — 5 — −

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

vessel coolant temperature — − — SDC — 5 — temperature — temperatures — −

(continued)

all changes are 1 unless otherwise indicated

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.~~ 3

6 The requirements for decay heat removal in MODES 4 and 5 are discussed in LCO 3.4.~~9~~ 8, "~~Residual Heat Removal (RHR)~~ SDC Shutdown Cooling System—Cold Shutdown"; LCO 3.9.8, "~~Residual Heat Removal (RHR)~~ Shutdown Cooling (SDC)—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. 7

A second Note has been provided to modify the ACTIONS related to ~~RHR shutdown cooling~~ SDC 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, and A.3

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

(continued)

allchanges are 1 unless otherwise indicated

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 SDC ~~RHR/shutdown cooling~~ capability. Therefore, an alternate method of decay heat removal must be provided.

With both SDC ~~RHR/shutdown cooling~~ subsystems inoperable, an required alternate method of decay heat removal must be provided in addition to that provided for the initial SDC ~~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 ~~Spent Fuel Pool Cooling System~~ and the Reactor Water Cleanup System 3 (by itself or using feed and bleed in combination with the Control Rod Drive System or Condensate/Feed 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 SDC ~~RHR shutdown cooling~~ subsystem and no recirculation pump in operation, except as permitted by LCO Note 1, reactor coolant circulation by the SDC ~~RHR shutdown cooling~~ subsystem or recirculation pump must be restored without delay.

Until SDC ~~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)

all changes are 1 unless otherwise indicated

BASES

ACTIONS

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

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.

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

SURVEILLANCE REQUIREMENTS

SR 3.4.~~8~~7.1

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

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

REFERENCES

None.

JUSTIFICATION FOR DEVIATIONS FROM NUREG-1433, REVISION 1
ITS BASES: 3.4.7 - 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. The proper LCO number has been included.

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

all changes are 1 unless otherwise indicated

B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.~~9~~ 8 ~~Residual Heat Removal (RHR)~~ Shutdown Cooling (SDC) 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~~ 212 °F ~~. This decay heat removal is~~ the decay heat must be removed for in preparation for performing refueling or maintenance operations, or ~~for~~ ~~keeping~~ maintaining the reactor in the Cold Shutdown condition. 2 3

The ~~two~~ three redundant, manually controlled shutdown cooling subsystems (loops) of the ~~RHR~~ SDC System provide decay heat removal. Each loop consists of ~~two~~ one motor driven pump~~s~~, a heat exchanger, and associated piping and valves. ~~Both~~ Each loop~~s~~ ~~have~~ has 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 low pressure coolant injection path and associated recirculation loop. The ~~RHR~~ heat exchangers transfer heat to the ~~RHR~~ SDC Service Water System, via the Reactor Building Closed Cooling Water (RBCCW) System. 3

APPLICABLE SAFETY ANALYSES

Decay heat removal by operation of the ~~RHR~~ SDC 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.~~ (5) 2 (9) (4) (10 CFR 50.36(c)(2)(ii)) 4

LCO

Two ~~RHR shutdown cooling~~ SDC subsystems are required to be OPERABLE, and when no recirculation pump is in operation, one ~~RHR shutdown cooling~~ SDC subsystem must be in operation. An OPERABLE ~~RHR shutdown cooling~~ SDC subsystem consists of one OPERABLE ~~RHR~~ SDC pump, one heat exchanger, ~~and~~ the associated piping and valves, and the necessary portions of the RBCCW System capable of providing cooling water to the heat exchanger and SDC pump seal cooler. 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~~ 3

(continued)

all changes are 1 unless otherwise indicated

BASES

LCO (continued)

~~in one loop of 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. ~~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~~ SDC 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.

is a; it is; is; 3; 2; and recirculation pumps; not be in operation; TSTF -53; Insert Note 1

Note ~~1~~ 2 permits both ~~RHR shutdown cooling~~ SDC subsystems to ~~be shut down~~ not be in operation for a period of 2 hours in an 8 hour period. Note ~~2~~ 3 allows one ~~RHR shutdown cooling~~ SDC 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 4, the ~~RHR Shutdown Cooling~~ SDC System ~~may~~ must be OPERABLE and one SDC subsystem shall be operated in the shutdown cooling mode to remove decay heat to maintain coolant temperature below ~~200~~ 212°F. Otherwise, a recirculation pump is required to be in operation.

5; 3

In MODES 1 and 2, and in MODE 3 with reactor ~~steam dome~~ vessel coolant temperature ~~pressure~~ greater than or equal to the ~~RHR~~ cut in permissive ~~pressure~~ temperature, this LCO is not applicable. Operation of the ~~RHR~~ SDC System in the shutdown cooling mode is not allowed above this ~~pressure~~ temperature because the RCS ~~pressure~~ temperature may exceed the design ~~pressure~~ temperature of the shutdown cooling piping. Decay heat removal at reactor ~~pressures~~ temperatures greater than or equal to the ~~RHR~~ cut in permissive ~~pressure~~ temperature is typically accomplished by condensing

– 2; – 2

(continued)

$\boxed{1}$ INSERT NOTE 1

Note 1 allows both SDC subsystems to not be in operation 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.

all changes are [1] unless otherwise indicated

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.~~ [3]

The requirements for decay heat removal in MODE 3 below the cut-in permissive ~~pressure~~ temperature [2] and in MODE 5 are discussed in LCO 3.4.~~8~~ 7 [6] "~~Residual Heat Removal (RHR)~~ Shutdown Cooling (SDC) System—Hot Shutdown"; LCO 3.9.8, "~~Residual Heat Removal~~ (~~RHR~~) SDC—High Water Level"; and LCO 3.9.9, "~~Residual Heat Removal (RHR)~~ SDC—Low Water Level."

ACTIONS

A Note has been provided to modify the ACTIONS related to ~~RHR shutdown cooling~~ SDC 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~~ SDC subsystem.

A.1

With one of the two required ~~RHR shutdown cooling~~ SDC subsystems inoperable, except as permitted by LCO Note ~~2~~ 1 and 3, the remaining [5] 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 required ~~RHR shutdown cooling~~ SDC subsystems inoperable, an alternate method of decay heat removal must be provided in addition to that provided for the initial ~~RHR shutdown cooling~~ SDC subsystem inoperability. This re-establishes backup decay heat removal capabilities, similar to the requirements of the LCO. The 1 hour

(continued)

all changes are 1 unless otherwise indicated

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~~ Condensate/Feed and Main Steam System 3 and the Reactor Water Cleanup System (by itself or using feed and bleed in combination with the Control Rod Drive System or Condensate/Feed System).

B.1 and B.2

With no ~~RHR shutdown cooling~~ SDC subsystem and no recirculation pump in operation, except as permitted by LCO Note ~~1~~ 5 and 2 and until ~~RHR~~ SDC or recirculation pump operation is re-established, an alternate method of reactor coolant circulation must be placed into service. This will provide the necessary circulation for monitoring coolant temperature. The 1 hour Completion Time is based on the coolant circulation function and is modified such that the 1 hour is applicable separately for each occurrence involving a loss of coolant circulation. Furthermore, verification of the functioning of the alternate method must be reconfirmed every 12 hours thereafter. This will provide assurance of continued temperature monitoring capability.

During the period when the reactor coolant is being circulated by an alternate method (other than by the required ~~RHR Shutdown Cooling~~ SDC 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.

(continued)

all changes are 1 unless otherwise indicated

BASES (continued)

SURVEILLANCE REQUIREMENTS

SR 3.4.~~9~~8.1

This Surveillance verifies that one ~~RHR shutdown cooling~~ SDC 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.

JUSTIFICATION FOR DEVIATIONS FROM NUREG-1433, REVISION 1
ITS BASES: 3.4.8 - 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.

B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.(10) RCS Pressure and Temperature (P/T) Limits

1 (9)

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 | criticality, 1

The (PTLR) contains P/T limit curves for heatup, cooldown, and 2 inservice leakage 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

2 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. 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 non-critical 2

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

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~~ 1 approved the curves and limits required by this Specification. Since the P/T limits are not derived from any DBA, there are no acceptance

(continued)

$\boxed{2}$ 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.

$\boxed{2}$ 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 Dresden 2 and 3, 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 Dresden 2 and 3, 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~~ 10 CFR 50.36(c)(2)(ii). 2

LCO

The elements of this LCO are: 1

a. RCS pressure, and temperature, ~~and~~ heatup and ~~or~~ cooldown rate 5 are ≤ 100°F in any 1 hour period are within the limits specified in ~~the PTLR~~ Figures 3.4.9-1, 3.4.9-2, and 3.4.9-3 during RCS heatup, cooldown, and inservice leak and hydrostatic testing, 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. 1

b. The temperature difference between the reactor vessel bottom head coolant and the reactor pressure vessel (RPV) coolant is ~~within the limit of the PTLR~~ ≤ 145°F during recirculation pump startup ~~, and during increases in THERMAL POWER or loop flow while operating at low THERMAL POWER or loop flow~~ in MODES 1, 2, 3, and 4; 5

c. The temperature difference between the reactor coolant in the respective recirculation loop and in the reactor vessel ~~meets the limit of the PTLR~~ is ≤ 50°F 1 during recirculation pump startup ~~, and during increases in THERMAL POWER or loop flow while operating at low THERMAL POWER or loop flow~~ in MODES 1, 2, 3, and 4; 5

d. RCS pressure and temperature are within the criticality limits specified in ~~the PTLR~~ Figure 3.4.9-3, prior to achieving criticality; and 1

e. The reactor vessel flange and the head flange temperatures are ~~within the limits of the PTLR~~ ≥ 83°F when tensioning the reactor vessel head bolting studs and when the reactor head is tensioned. 5

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~~age~~ and 1 hydrostatic testing P/T limit curves. Thus, the LCO for the

(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~~s~~, size~~s~~, and orientation~~s~~ of flaws in the vessel material. 3

APPLICABILITY

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

ACTIONS

A.1 and A.2

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

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 2

(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 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. 212 2

Insert C.1 and C.2 4

SURVEILLANCE REQUIREMENTS

9 1

SR 3.4.10.1

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.

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.

(continued)

$\boxed{4}$ 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 9

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.

1 TSTF-353 changes not adopted

SR 3.4.10.3 and SR 3.4.10.4 9 1

Differential temperatures within the applicable PTLR limits 2 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 3.4.10.4 is to compare the temperatures of the operating recirculation loop and the idle loop. 9 1

1 9 and SR 3.4.9.4 have 5

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.

5 TSTF-35 The Note(s) also state(s) the SR(s) are only required to be met during a recirculation pump startup since this is when the stresses occur.

SR 3.4.10.5, SR 3.4.10.6, and SR 3.4.10.7 9 1

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) 9

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 and every 30 minutes thereafter 5 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~~ 93 1 °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~~ 113 1 °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.

Insert SR 3.4.9.5, SR 3.4.9.6, and SR 3.4.9.7 4

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.~~ [Letter from Robert Pulsifer (NRC) to ComEd, "Issuance of Amendments 153 and 148 for Dresden 2 and 3," dated February 28, 1997.] 2
8. U FSAR, Section ~~15.1.26~~ 15.4.4. 6

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 ≤ 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.

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 requirements.

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

B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.10 Reactor Steam Dome Pressure
0

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 analysis 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 4 5

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

Insert ASA

Reactor steam dome pressure satisfies the requirements of Criterion 2 of ~~the NRC Policy Statement~~ 10 CFR 50.36(c)(2)(ii). 4 4

LCO

The specified reactor steam dome pressure limit of ≤ ~~[1020]~~ 1005 psig ensures the plant is operated within the assumptions of the ~~transient analyses~~ reactor overpressure analysis. Operation above the limit may result in a transient response more severe than analyzed. 3 4

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)

4 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

4 APPLICABILITY (continued)

MODES, the reactor may be generating significant steam and ~~the design basis accidents and transients are bounding~~.

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. ~~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.~~ 6

B.1

If the reactor steam dome pressure cannot be restored to within the limit within the associated Completion Time, the plant must be brought to a MODE in which the LCO does not apply. To achieve this status, the plant must be brought to at least MODE 3 within 12 hours. The allowed Completion Time of 12 hours is reasonable, based on operating experience, to reach MODE 3 from full power conditions in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.4.~~12~~ 10.1 1

Verification that reactor steam dome pressure is ≤ ~~[1020]~~ 1005 3 psig ensures that the initial condition~~s~~ of the ~~design basis accidents and transients~~ vessel overpressure protection analysis 4 ~~are~~ is 4 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 4	U 1. FSAR, Section [5.2.2.2.4].	1
Chapter	2. FSAR, Section [15].	3

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. 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.

5. Typographical/grammatical error corrected.

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.

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 Dresden 2 and 3 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 the pump to service has been 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 the pumps are 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 help ensure that thermal stresses from the startup of a idle recirculation pump will not exceed design allowances. Therefore, this change does not significantly increase the consequences of any accident previously evaluated.

2. Does the change create the possibility of a new or different kind of accident from any accident previously evaluated?

   The proposed 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.

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.

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.

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

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

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 - SHUTDOWN COOLING SYSTEM — HOT SHUTDOWN

L.1 CHANGE

In accordance with the criteria set forth in 10 CFR 50.92, ComEd has evaluated this proposed Technical Specifications change and determined it does not represent a significant hazards consideration. The following is provided in support of this conclusion.

1. Does the change involve a significant increase in the probability or consequences of an accident previously evaluated?

   The proposed change allows time to place the system in service after reaching the applicable conditions. Since the system can not physically be placed in service until the cut-in permissive temperature setpoint is reached, this change only allows the activity to take place without resorting to intentional noncompliance with the requirements. Since no actual change to the operation of the plant is involved, the proposed change will not increase the probability or consequences of any accident previously evaluated.

2. Does the change create the possibility of a new or different kind of accident from any accident previously evaluated?

   The proposed change introduces no new mode of plant operation and it does not involve physical modification to the plant. Therefore, it does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. Does this change involve a significant reduction in a margin of safety?

   This change does not involve a significant reduction in a margin of safety since the proposed change only allows time to conduct the necessary manipulations to place the required system in service.

L.2 CHANGE

In accordance with the criteria set forth in 10 CFR 50.92, ComEd has evaluated this proposed Technical Specifications change and determined it does not represent a significant hazards consideration. The following is provided in support of this conclusion.

1. Does the change involve a significant increase in the probability or consequences of an accident previously evaluated?

   The proposed change will allow the required shutdown cooling (SDC) loops to be inoperable for 2 hours and both SDC loops and recirculation pumps to be not in operation for 2 hours per 8 hour period. Currently, the 2 hour allowances are only applicable if one SDC loop is OPERABLE. While the UFSAR evaluates the loss of all SDC, the event is not an assumed accident. In addition, the change still requires alternate methods for decay heat removal for each inoperable SDC loop to be in place and reactor coolant to be recirculating when the 2 hour allowance is used. The alternate methods must each be fully capable of removing decay heat load, thus the method is essentially equivalent to the SDC loops in this respect. In addition, the current Technical Specifications allow use of the alternate methods in lieu of the SDC loops for an unlimited amount of time. Therefore, this change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. Does the change create the possibility of a new or different kind of accident from any accident previously evaluated?

   The proposed change introduces no new mode of plant operation and it does not involve physical modification to the plant. Therefore, it does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. Does this change involve a significant reduction in a margin of safety?

   The proposed change will allow the required SDC loops to be inoperable for up to 2 hours and both SDC loops and pumps to be not in operation for 2 hours per 8 hour period. Currently, the 2 hour allowances are only applicable if one SDC loop is OPERABLE. The change does not affect the requirement to have an alternate method capable of decay heat removal for each inoperable SDC loop and the reactor coolant to be recirculating. Each alternate method must be fully capable of removing the decay heat load and circulating reactor coolant, thus the alternate methods are essentially equivalent to the SDC loops in this respect. In addition, the current Technical Specifications allow use of the alternate methods in lieu of the SDC loops for an unlimited amount of time. Therefore, this change does not involve a reduction in a margin of safety.

L.1 CHANGE

In accordance with the criteria set forth in 10 CFR 50.92, ComEd has evaluated this proposed Technical Specifications change and determined it does not represent a significant hazards consideration. The following is provided in support of this conclusion.

1. Does the change involve a significant increase in the probability or consequences of an accident previously evaluated?

   The proposed change will allow the required shutdown cooling (SDC) loops to be inoperable for 2 hours and both SDC loops and recirculation pumps to be not in operation for 2 hours per 8 hour period. Currently, the 2 hour allowances are only applicable if one SDC loop is OPERABLE. While the UFSAR evaluates the loss of all SDC, the event is not an assumed accident. In addition, the change still requires alternate methods for decay heat removal for each inoperable SDC loop to be in place and reactor coolant to be recirculating when the 2 hour allowance is used. The alternate methods must each be fully capable of removing the decay load, thus the method is essentially equivalent to the SDC loops in this respect. In addition, the current Technical Specifications allow use of the alternate methods in lieu of the SDC loops for an unlimited amount of time. Therefore, this change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. Does the change create the possibility of a new or different kind of accident from any accident previously evaluated?

   The proposed change introduces no new mode of plant operation and it does not involve physical modification to the plant. Therefore, it does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. Does this change involve a significant reduction in a margin of safety?

   The proposed change will allow the required SDC loops to be inoperable for up to 2 hours and both SDC loops and recirculation pumps to be not in operation for 2 hours per 8 hour period. Currently, the 2 hour allowances are only applicable if one SDC loop is OPERABLE. The change does not affect the requirement to have an alternate method capable of decay heat removal for each inoperable SDC loop and the reactor coolant to be recirculating. Each alternate method must be fully capable of removing the decay heat load and circulating reactor coolant, thus the alternate methods are essentially equivalent to the SDC loops in this respect. In addition, the current Technical Specifications allow use of the alternate methods in lieu of the SDC loops for an unlimited amount of time. 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?

   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-5) 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 of a previously analyzed accident. 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 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.1 CHANGE

3. (continued)

operations curve (CTS Figure 3.6.K-5) 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.

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 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.

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 ISOLATION CONDENSER (IC) 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 $\leq$ 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. <u>OR</u> 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 IC System is OPERABLE.	Immediately
	AND		
	F.2	Restore HPCI System to OPERABLE status.	14 days
G. HPCI System inoperable. AND One low pressure ECCS injection/spray subsystem is inoperable or Condition C entered.	G.1	Restore HPCI System to OPERABLE status.	72 hours
	OR		
	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. <u>OR</u> Two or more required ADS valves inoperable.	I.1 Be in MODE 3. <u>AND</u> I.2 Reduce reactor steam dome pressure to $\leq$ 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. <u>OR</u> HPCI System and one or more required ADS valves inoperable. <u>OR</u> 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	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
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 / $\geq$ 4500 gpm / 1 / $\geq$ 90 psig LPCI / $\geq$ 14,500 gpm / 3 / $\geq$ 20 psig	In accordance with the Inservice Testing Program

(continued)

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
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 $\leq$ 1005 and $\geq$ 920 psig, the HPCI pump can develop a flow rate $\geq$ 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 $\leq$ 180 psig, the HPCI pump can develop a flow rate $\geq$ 5000 gpm against a system head corresponding to reactor pressure.	24 months
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

(continued)

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
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 ISOLATION CONDENSER (IC) 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 $\geq$ 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. AND C.2 Restore one required ECCS injection/spray subsystem to OPERABLE status.	Immediately 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
	<u>AND</u>		
	D.2	Initiate action to restore one standby gas treatment subsystem to OPERABLE status.	Immediately
	<u>AND</u>		
	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 $\geq$ 10 ft 4 inches; 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 $\geq$ 21 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	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 ISOLATION CONDENSER (IC) SYSTEM

3.5.3 IC System

LCO 3.5.3 The IC 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. IC System inoperable.	A.1	Verify by administrative means High Pressure Coolant Injection System is OPERABLE.	Immediately
	<u>AND</u>		
	A.2	Restore IC System to OPERABLE status.	14 days
B. Required Action and associated Completion Time not met.	B.1	Be in MODE 3.	12 hours
	<u>AND</u>		
	B.2	Reduce reactor steam dome pressure to $\leq$ 150 psig.	36 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.5.3.1	Verify the IC System: a. shellside water level $\geq$ 6 feet; and b. shellside water temperature $\leq$ 210°F.	24 hours
SR 3.5.3.2	Verify each IC 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	Verify the IC System actuates on an actual or simulated automatic initiation signal.	24 months
SR 3.5.3.4	Verify IC System heat removal capability to remove design heat load.	60 months

B 3.5 EMERGENCY CORE COOLING SYSTEMS (ECCS) AND ISOLATION CONDENSER (IC) 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) 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 Containment Cooling Service Water System. Depending on the

(continued)

BASES

BACKGROUND (continued)

location and size 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 14 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.

The LPCI System is composed of two LPCI subsystems (loops) (Ref. 2). Each subsystem consists 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 two LPCI subsystems are interconnected via the two, normally open, LPCI System cross-tie valves. 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

(continued)

BASES

BACKGROUND (continued)

started (simultaneously and immediately when normal AC power is available, and sequentially, with A and C pumps after approximately 4 seconds and B and D pumps after approximately 9 seconds, when emergency AC power is available). LPCI System valves 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, "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 the reactor vessel.

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 valves are 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 isolation valve. When the HPCI System is aligned to the suppression pool the "keep fill" system must be aligned to the HPCI discharge line.

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 ≤ 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 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. 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 electromatic ADS 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, 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.

(continued)

BASES (continued)

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).

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

(continued)

BASES

ACTIONS

B.1 (continued)

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 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).

(continued)

BASES

ACTIONS (continued)

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 IC 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 IC System will automatically provide core cooling at most reactor operating pressures. Verification of IC 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 IC is out of service for maintenance or other reasons. It does not mean to perform the Surveillances needed to demonstrate the OPERABILITY of the IC System. If the OPERABILITY of the IC 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.

(continued)

BASES

ACTIONS

G.1 and G.2 (continued)

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, operation is only allowed for a limited time. The 14 day Completion Time is based on a reliability study cited in Reference 10 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

(continued)

BASES

ACTIONS

I.1 and I.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.

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.

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

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.5.1.2 (continued)

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.

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.

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.5.1.4 (continued)

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) and are bounded by the requirements of SR 3.5.1.5. 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 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

(continued)

BASES

SURVEILLANCE REQUIREMENTS

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

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.

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

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.5.1.8 (continued)

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.

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 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 3.5.1.9

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

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.5.1.9 (continued)

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 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.

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.5.1.10 (continued)

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 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.

(continued)

BASES

REFERENCES (continued)

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. EMF-97-025(P) Revision 1, LOCA Break Spectrum for Dresden Unit 2 and 3, dated May 30, 1997.

B 3.5 EMERGENCY CORE COOLING SYSTEMS (ECCS) AND ISOLATION CONDENSER (IC) SYSTEM

B 3.5.2 ECCS - Shutdown

BASES

BACKGROUND	A description of the Core Spray (CS) System and the Low Pressure Coolant Injection (LPCI) 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 the similar injection capacity in relation to a CS subsystem.

(continued)

BASES (continued)

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 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.

(continued)

BASES

ACTIONS

A.1 and B.1 (continued)

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 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

(continued)

BASES

ACTIONS

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

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 10 ft 4 inches 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 $<$ 10 ft 4 inches, 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 $\geq$ 10 ft 4 inches or that required low pressure ECCS injection/spray subsystems are aligned to take suction from the CCST and the CCST contains $\geq$ 140,000 gallons of water, equivalent to 21 ft, ensures that the required low pressure ECCS injection/spray subsystems can supply at least 50,000 gallons of makeup water to the RPV. The CS and LPCI suctions are uncovered at the 90,000 gallon level. 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

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.5.2.1 (continued)

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 involves verification that those valves capable of potentially being mispositioned are in the correct position. This SR does not apply to valves that cannot be inadvertently misaligned, such as check valves. The 31 day Frequency is appropriate because the valves are operated under procedural control and the probability of their being mispositioned during this time period is low.

REFERENCES

1. UFSAR, Section 6.3.3.4.1.

B 3.5 EMERGENCY CORE COOLING SYSTEMS (ECCS) AND ISOLATION CONDENSER (IC) SYSTEM

B 3.5.3 IC System

BASES

BACKGROUND

The IC System is not part of the ECCS; however, the IC System is included with the ECCS section because of their similar functions.

The IC System is designed to operate either automatically or manually following reactor pressure vessel (RPV) isolation to provide adequate core cooling. Under these conditions, the High Pressure Coolant Injection (HPCI) and IC systems perform similar functions.

The IC System (Ref.1) is a passive high pressure system comprised of one natural circulation heat exchanger, two AC motor-operated isolation valves, two D.C. motor-operated isolation valves, and two tube side high point vent isolation valves to main steam line "A". The IC System functions as a heat sink for decay heat removal from the reactor vessel following reactor scram and isolation from the main condenser. This function prevents overheating of the reactor fuel, controls reactor pressure, and limits the loss of reactor coolant through the relief valves. The IC System is automatically initiated by sustained reactor vessel high pressure and, once activated, remains in operation until manually removed from service.

The isolation condenser shell contains two tube bundles. When the IC System is in operation, both tube bundles are in service.

The IC System is designed to provide core cooling for reactor pressure $\geq$ 150 psig. The shell side of the condenser has a minimum water level of 6 feet which provides an inventory of $\geq$ 18,700 gallons. This minimum level provides $\geq$ 11,300 gallons (approximately 3 feet) of water above the top of the tube bundles. The shell side water temperature must be $\leq$ 210°F. During normal plant operations, when the system is in standby, makeup is from the clean demineralized water storage tank. Makeup during IC System operation can be provided from the Condensate

(continued)

BASES

BACKGROUND (continued)	Transfer System. Since during operation of the IC System, water in the shell will boil, the condenser is vented to the atmosphere via one line.
APPLICABLE SAFETY ANALYSES	The function of the IC System is to respond to main steam line isolation events by providing core cooling to the reactor. Although the IC System is an Engineered Safety Feature System, no credit is taken in the accident analyses for IC 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 IC 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. The IC System reduces the loss of RPV inventory during an isolation event.
APPLICABILITY	The IC System is required to be OPERABLE during MODE 1, and MODES 2 and 3 with reactor steam dome pressure > 150 psig, since IC is the primary non-ECCS 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, IC is not required to be OPERABLE since the low pressure ECCS injection/spray subsystems can provide sufficient core cooling.
ACTIONS	<u>A.1 and A.2</u> If the IC 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 IC System must be restored to OPERABLE status within 14 days. In this Condition, loss of the IC 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 immediately when the IC System is inoperable. This may be performed as an administrative check, by examining logs or

(continued)

BASES

ACTIONS

A.1 and A.2 (continued)

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 immediately entered. For transients and certain abnormal events with no LOCA, IC (as opposed to HPCI) is an acceptable source of core cooling which also limits the loss of the RPV water level. Therefore, a limited time is allowed to restore the inoperable IC 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 IC, the AOTs (i.e., Completion Times) determined for HPCI are also applied to IC.

B.1 and B.2

If the IC 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 $\leq$ 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

This SR verifies the water volume and temperature in the shell side of the IC to be sufficient for proper operation. Based on a scram from 2552.3 MWt (101% RTP), a minimum water level of 6 feet at a temperature of $\leq$ 210°F in the condenser

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.5.3.1 (continued)

provides sufficient decay heat removal capability for 20 minutes of operation without makeup water, before beginning to uncover the tube bundles. The volume and temperature allow sufficient time for the operator to provide makeup to the condenser.

The 24 hour Frequency is based on operating experience related to the trending of the parameter variations during normal operation.

SR 3.5.3.2

Verifying the correct alignment for manual, power operated, and automatic valves in the IC flow path provides assurance that the proper flow path will exist for IC 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 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 IC System. This Frequency has been shown to be acceptable through operating experience.

SR 3.5.3.3

The IC 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 IC System will cause the system to

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.5.3.3 (continued)

operate as designed; that is, actuation of all automatic valves to their required positions. The LOGIC SYSTEM FUNCTIONAL TEST performed in LCO 3.3.5.2 overlaps this Surveillance to provide complete testing of the assumed design 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.

SR 3.5.3.4

Verifying the proper flow path and heat exchange capacity for IC System operation ensures the capability of the IC System to remove the design heat load. This SR verifies the IC System capability to remove heat consistent with the design requirements of 252.5×10^6 Btu/hr. The IC System capacity is equivalent to the decay heat rate 5 minutes after a reactor scram.

The 60 month Frequency is based on engineering judgement, and has been shown to be acceptable through operating experience.

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:~~ LA.1

~~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.~~

M.1 — S ~~2. The low pressure coolant injection (LPCI) subsystem comprised of:~~ LA.1

~~a. Four OPERABLE LPCI pumps, and~~

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

~~3. The high pressure cooling injection (HPCI) system consisting of:~~ LA.1

~~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.~~

LCO 3.5.1 4. The automatic depressurization system (ADS) with at least ~~5~~ 4 OPERABLE ADS valves. L.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:

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

SR 3.5.1.2 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. LA.2

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

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.

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

~~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

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

ACTION E

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~~ Enter LCO 3.0.3 M3

2. For the LPCI subsystem: A.2

ACTION A

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

ACTION E

4.5 - SURVEILLANCE REQUIREMENTS

SR 3.5.1.5

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

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~~(c)~~. A.3

3. At least once per ~~18~~ 24 months: LD.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.

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 ~~350~~ 180 psig~~(c)~~. A.3 LA.3 M.4

APPLICABILITY

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

d ~~The provisions of Specification 3.9.A, Actions 4.a or 6.b are applicable to the LPCI subsystem such that with an inoperable diesel generator, for the remaining OPERABLE diesel generator, both LPCI pumps (and their associated flow path) associated with that OPERABLE diesel generator, shall be OPERABLE. 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 and flow is adequate to perform the test. A.3

3.5 - LIMITING CONDITIONS FOR OPERATION

M.1 — (a) A.2 ACTION B ACTION C

b. With the LPCI subsystem otherwise inoperable(d), provided that both CS subsystems are OPERABLE, restore the LPCI subsystem to OPERABLE status within 7 days or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours. M.1 add proposed ACTION D

ACTION E

ACTION J

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.~~ M.3 Enter LCO 3.0.3

ACTION F

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

ACTION I

add proposed ACTION G L.3

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

4.5 - SURVEILLANCE REQUIREMENTS

~~2) The pump suction is automatically transferred from the condensate storage tank to the suppression chamber on a condensate storage tank water level - low signal and on a suppression chamber water level - high signal.~~ LA.3

~~c. Performing a CHANNEL CALIBRATION of the CS and LPCI system discharge line "keep filled" alarm instrumentation.~~ L.2

~~d. Deleted.~~ LD.1

4. At least once per ~~18~~ 24 months for the ADS:

SR 3.5.1.9

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

SR 3.5.1.10

b. Manually opening each required ADS valve L.1 ~~when the reactor steam dome pressure is ≥150 psig~~(c) 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

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

SR 3.5.1.10 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. A3

3.5 - LIMITING CONDITIONS FOR OPERATION

4.5 - SURVEILLANCE REQUIREMENTS

ACTION H — status within 14 days or be in at

ACTION I — least HOT SHUTDOWN within the next 12 hours and reduce reactor steam dome pressure to ≤150 psig within the following 24 hours.

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

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

← add proposed ACTION J — A.4, M.3

ADMINISTRATIVE

A.1 In the conversion of the Dresden 2 and 3 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 (d), 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 (d) 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 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.

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.

TECHNICAL CHANGES - MORE RESTRICTIVE

M.2 (cont'd) 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 alternate 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 1.b or 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 the 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. 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 - MORE RESTRICTIVE (continued)

M.4 CTS Surveillance Requirement 4.5.A.3.b.1) requires verifying the HPCI system develops a flow of ≥ 5000 gpm against a system head corresponding to reactor vessel pressure, when steam is being supplied to the turbine between 150 and 350 psig. Proposed ITS SR 3.5.1.7 requires verifying the system flow is ≥ 5000 gpm against a system head corresponding to reactor pressure with reactor pressure ≤ 180 psig. The requirement for steam supply pressure to be ≤ 180 psig has been added consistent with requirements at Quad Cities. The requirement to test at the lower pressure is an additional restriction on plant operation.

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 contaminated 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

TECHNICAL CHANGES - LESS RESTRICTIVE

LA.3 (cont'd) 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.

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 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

TECHNICAL CHANGES - LESS RESTRICTIVE

LD.1 (cont'd) 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 OPERABILITY. 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 these conditions, 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 and the remaining OPERABLE low pressure 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

TECHNICAL CHANGES - LESS RESTRICTIVE

L.3 (cont'd) 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 Dresden 2 and 3 will be controlled by plant procedures. Regulations provide sufficient control of these provisions for their removal from Technical Specifications.

RELOCATED SPECIFICATIONS

None

A.1

ITS 3.5.2

SR 3.5.2.2
SR 3.5.2.3
SR 3.5.2.4

3.5 - LIMITING CONDITIONS FOR OPERATION

LCO 3.5.2

B. Emergency Core Cooling System - Shutdown low pressure ECCS injection/spray

At least two ~~of the following four~~ subsystems~~/loops~~ shall be OPERABLE:

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~~ 50,000 available gallons of water.

SR 3.5.2.1.b

L.5

LA.1

2. ~~One or both low pressure coolant injection (LPCI) subsystem loops with a subsystem loop comprised of:~~
   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

SR 3.5.2.5

LD.1

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

SR 3.5.2.4

2. Each LPCI pump develops the required flow when tested pursuant to Specification 4.0.E.

add proposed flow rate and head conditions for one pump

M.1

A.1

ITS 3.5.2

3.5 - LIMITING CONDITIONS FOR OPERATION

4.5 - SURVEILLANCE REQUIREMENTS

(2) ~~When the suppression chamber water level is less than the limit or is drained, from the condensate storage tank~~ — LA.1

SR 3.5.2.1.b — containing at least ~~140,000~~ 50,000 — L.5
available gallons of water.

APPLICABILITY:

OPERATIONAL MODE(s) 4 and 5[a].

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 — A.4

ACTION D — SECONDARY CONTAINMENT INTEGRITY within the next 8 hours — A.3

Applicability — a The ECCS is not required to be OPERABLE provided that the reactor vessel head is removed, the cavity is flooded, — A.5 — and water level is maintained within the limits of Specification 3.10.G and 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.1 ITS 3.5.2

3.5 - LIMITING CONDITIONS FOR OPERATION

C. Suppression Chamber

moved to ITS 3.6.2.2 A.6

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' 6.5" above the bottom of the suppression chamber.

SR 3.5.2.1.a (10 ft, 4 inches) A.8 LA.1

2. In OPERATIONAL MODE(s) 4 and 5[(a)] with a contained volume equivalent to a water level of ~~≥8'~~ above the bottom of the suppression chamber, except that the suppression chamber level may be less than the limit provided that: L.2

   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, L.3

SR 3.5.2.1.b

   c. The condensate storage tank contains ≥~~140,000~~ available gallons of water, and — (50,000) L.5 — Add Note to SR 3.5.2.1.b L.2

LCO 3.5.2

   d. The ECCS systems are OPERABLE per Specification 3.5.B.

APPLICABILITY:

A.6 moved to ITS 3.6.2.2

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

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' 6.5".

2. For OPERATIONAL MODE(s) 4 or 5[(a)], at least once per 12 hours: A.7 A.8

SR 3.5.2.1.a

   a. The water level to be ≥~~8'~~, or (10 ft, 4 inches)

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

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. A.5 M.2

A.1 ITS 3.5.2

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 ITS 3.6.2.2

L.2 Add proposed Required Action A.1

Required Action C.1 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.~~ L.1 L.3

ACTION D Establish SECONDARY CONTAINMENT INTEGRITY within 8 hours. A.4 A.3

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

ADMINISTRATIVE

A.1 In the conversion of the Dresden 2 and 3 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 ISTS, 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 (a) 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.

ADMINISTRATIVE (continued)

A.8 The required suppression chamber water level of "≥ 8'" specified in CTS 3.5.C.2 and CTS 4.5.C.2.a is being changed to "≥ 10 ft 4 inches." This change is provided in the Dresden 2 and 3 ITS consistent with the Technical Specifications Change Request submitted to the NRC for approval per a ComEd letter, dated May 20, 1999. As such, this 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 allowances 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 of CTS 3/4.5.B relating to system OPERABILITY (in this case 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.

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 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

TECHNICAL CHANGES - LESS RESTRICTIVE

L.2 (cont'd) 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 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

TECHNICAL CHANGES - LESS RESTRICTIVE

L.4 (cont'd) 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 ([18] 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

A.1

ITS 3.5.3

3.5 - LIMITING CONDITIONS FOR OPERATION

D. Isolation Condenser

LCO 3.5.3 The isolation condenser (IC) system shall be OPERABLE.

APPLICABILITY:

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

ACTION: Immediately

ACTION A — With the IC system inoperable, operation may continue provided the HPCI system is OPERABLE; restore the IC system to OPERABLE status within 14 days or

ACTION B — be in at least HOT SHUTDOWN within the next 12 hours and reduce reactor steam dome pressure to ≤150 psig within the following 24 hours.

4.5 - SURVEILLANCE REQUIREMENTS

D. Isolation Condenser

The IC system shall be demonstrated OPERABLE:

1. SR 3.5.3.1 At least once per 24 hours by verifying the shell side water volume and the shell side water temperature to be within limits.
2. SR 3.5.3.2 At least once per 31 days by verifying that each valve, manual, power operated or automatic in the flow path that is not locked, sealed or otherwise secured in position, is in its correct position.
3. SR 3.5.3.3 At least once per 18 24 months by verifying the IC system actuates on an actual or simulated automatic initiation signal. LD.1
4. SR 3.5.3.4 At least once per 5 years by verifying the system heat removal capability. M.1

Page 1 of 1

ADMINISTRATIVE

A.1 In the conversion of the Dresden 2 and 3 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 CTS 4.5.D.4 requires verifying the IC System heat removal capability at least once per 5 years. Proposed ITS SR 3.5.3.4 retains this requirement and in addition specifies acceptance criteria of removal of the design heat load. Although consistent with the current plant requirements, the addition of these acceptance criteria in Technical Specifications is considered more restrictive.

TECHNICAL CHANGES - LESS RESTRICTIVE

"Generic"

LD.1 The Frequency for performing CTS 4.5.D.3 (proposed SR 3.5.3.3) has been extended from 18 months to 24 months. The IC system functional test (proposed SR 3.5.3.3) ensures that a system initiation signal (actual or simulated) to the automatic initiation logic of IC will cause the system or subsystems to operate as designed, including actuation of the system throughout its emergency operating sequence, 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 this test normally passes its Surveillance 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

TECHNICAL CHANGES - LESS RESTRICTIVE

LD.1 (cont'd) minimal. This conclusion is based on the following evaluation. The increased interval between SR performances is acceptable because IC is not a system that is taken credit for in the safety analysis. Additionally, the functions performed by IC can be performed by HPCI, and Technical Specifications do not permit HPCI and IC to be inoperable concurrently. Therefore, the impact of this change, if any, on system availability is small. 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 as implemented in SR 3.5.3.3. In addition, the proposed 24 month Surveillance Frequency, if performed at the maximum interval allowed by proposed SR 3.0.2 (30 months) does not invalidate any assumptions in the plant licensing basis.

"Specific"

None

RELOCATED SPECIFICATIONS

None

DISCUSSION OF CHANGES
ITS: SECTION 3.5 - ECCS AND IC 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 Dresden 2 and 3 ITS Section 3.5, consistent with the BWR ISTS, NUREG-1433, Rev. 1. The revised Bases are as shown in the Dresden 2 and 3 ITS Bases.

all changes are [3] unless otherwise identified

<CTS>

3.5 EMERGENCY CORE COOLING SYSTEMS (ECCS) AND ~~REACTOR CORE ISOLATION COOLING (RCIC)~~ ISOLATION CONDENSER (IC) SYSTEM [1]

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 [2] ~~safety/~~ relief valves shall be OPERABLE.

<Appl 3.5.A> <3.5.A Footnote (b)> 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. [2]

ACTIONS

Insert ACTIONS A, B, C and D

TSTF-318 change not shown

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> ~~B.~~ E. Required Action and associated Completion Time of Condition A, B, C, or D not met.	~~B.~~ E.1 Be in MODE 3.	12 hours
	AND	
	~~B.~~ E.2 Be in MODE 4.	36 hours
<3.5.A Act 3> ~~C.~~ F. HPCI System inoperable.	~~C.~~ F.1 Verify by administrative means ~~RC~~ IC System is OPERABLE. [1]	~~1 hour~~ Immediately [13] TSTF-301
	AND	
	~~C.~~ F.2 Restore HPCI System to OPERABLE status.	14 days

(continued)

3 INSERT ACTIONS A, B, C, & D

<CTS>

<3.5.A Act 2.a> A. One Low Pressure Coolant Injection (LPCI) pump inoperable.	A.1 Restore LPCI pump to OPERABLE status.	30 days
<3.5.A Act 1.a> <3.5.A Act 2.b> 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
<Doc M.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 [3] unless otherwise identified

<CTS>

ACTIONS (continued)

	CONDITION	REQUIRED ACTION		COMPLETION TIME
<DOC L.3>	*G*. HPCI System inoperable. AND One low pressure ECCS injection/spray subsystem is inoperable *[TSTF-318] or Condition A entered*.	*G*.1 OR *G*.2	Restore HPCI System to OPERABLE status. Restore low pressure ECCS injection/spray subsystem*(s)* to OPERABLE status. *[TSTF-318]*	72 hours 72 hours
[14] C *<3.5.A Act 4.a>*	*H*. One *required* ADS valve inoperable.	*H*.1	Restore ADS valve to OPERABLE status.	14 days
	~~F. One ADS valve inoperable.~~ ~~AND~~ ~~One low pressure ECCS injection/spray subsystem inoperable.~~	~~F.1~~ ~~OR~~ ~~F.2~~	~~Restore ADS valve to OPERABLE status.~~ ~~Restore low pressure ECCS injection/spray subsystem to OPERABLE status.~~	~~72 hours~~ ~~72 hours~~ *[15]* *TSTF-318 change not shown*
<3.5.A Act 3> *<3.5.A Act 4.a>* *<3.5.A Act 4.b>* *[5]*	*I*. *[4]* Two or more *required* ADS valves inoperable. OR Required Action and associated Completion Time of Condition *C*, *E*, *G*, *H* or *F* not met.	*I*.1 AND *I*.2	Be in MODE 3. Reduce reactor steam dome pressure to ≤ [150] psig. *[2]*	12 hours 36 hours

(continued)

all changes are [3] unless otherwise identified

<CTS>

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
<3.5.A Act 1.b> <3.5.A Act 2.c> <Doc A.4> <Doc M.3> J H. 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 [4] ADS valves inoperable. OR One or more low pressure ECCS injection/spray subsystems inoperable and one or more required ADS valves inoperable. [15]	H.1 Enter LCO 3.0.3.	Immediately

<CTS>

SURVEILLANCE REQUIREMENTS

	SURVEILLANCE	FREQUENCY
<4.5.A.1.a.1> SR 3.5.1.1	Verify, for each ECCS injection/spray subsystem, the piping is filled with water from the pump discharge valve to the injection valve.	31 days
<4.5.A.1.a.2> 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.~~ 6 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~~ 7
~~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~~ 8
<Doc M.2> [SR 3.5.1.~~5~~ 3] 8	Verify ~~each LPCI inverter output voltage is ≥ [570] V and ≤ [630] V while supplying the respective bus~~. 9 Correct breaker alignment to the LPCI swing bus	31 days 2

(continued)

<CTS>

SURVEILLANCE REQUIREMENTS (continued)

	SURVEILLANCE 14	FREQUENCY
<DOC M.2> SR 3.5.1.~~6~~ 8 4	~~NOTE~~ ~~Not required to be performed if performed within the previous 31 days.~~ 8 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]. 2	In accordance with the Inservice Testing Program ~~Once each startup prior to exceeding 25% RTP~~
<4.5.A.2.a> <4.5.A.2.b> SR 3.5.1.~~7~~ 8 5	1 test line pressure Verify the following ECCS pumps develop 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~~ 2
<4.5.A.2.c> SR 3.5.1.~~8~~ 8 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] ≤ ~~[1020]~~ 1005 and ≥ [920] psig, the HPCI pump can develop a flow rate ≥ ~~[4250]~~ 5000 gpm [against a system head corresponding to reactor pressure]. 2	In accordance with the Inservice Testing Program 10 ~~92 days~~

Table for SR 3.5.1.7 (renumbered 5):

SYSTEM	FLOW RATE	NO. OF PUMPS	[~~SYSTEM HEAD~~ CORRESPONDING TO A REACTOR PRESSURE OF]
Core Spray	≥ ~~[4250]~~ 4500 gpm	[1]	≥ ~~[113]~~ 90 psig
LPCI	≥ ~~[17,000]~~ 14,500 gpm	~~[2]~~ 3	≥ [20] psig

(continued)

<CTS>

SURVEILLANCE REQUIREMENTS (continued)

	SURVEILLANCE		FREQUENCY
<4.5.A.3.b>	SR 3.5.1.~~9~~ 7 8	---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]~~ 180 psig, the HPCI pump can develop a flow rate ≥ ~~[4250]~~ 5000 gpm ~~[~~against a system head corresponding to reactor pressure~~]~~. 2	~~18~~ 24 months 11
<4.5.A.3.a>	SR 3.5.1.~~10~~ 8 8	---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 2
<4.5.A.4.a>	SR 3.5.1.~~11~~ 9 8	---NOTE--- Valve actuation may be excluded. --- Verify the ADS actuates on an actual or simulated automatic initiation signal.	~~[18]~~ 24 months 2
<4.5.A.4.b>	SR 3.5.1.~~12~~ 10 8	---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. 4	~~18~~ 24 months ~~[on a STAGGERED TEST BASIS for each valve solenoid]~~ 11

Insert SR 3.5.1.11 12

12 INSERT SR 3.5.1.11

<CTS>

<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

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. 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 to 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.

4. The word "required" has been added consistent with its use throughout the ITS (only four of the five installed ADS valves are required).

5. Change made to be consistent with the Writer's Guide.

6. ITS SR 3.5.1.2 Note for consideration of LPCI Operability when aligned for decay heat removal (RHR) has been deleted. The use of LPCI for this function is not applicable at Dresden 2 and 3.

7. 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 electromatic relief (EMR) 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.

8. The brackets have been removed and the information deleted since it does not apply. Subsequent SRs have been renumbered as required.

9. 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 Dresden 2 and 3 design.

10. 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.

11. 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.

12. 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.

13. 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 the IC System is not inoperable at this time, the Required Action is met. However, since the Completion Time starts upon entry into this Condition, if the IC System 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 the IC System being inoperable, even though the IC System 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 Dresden 2 and 3 Technical Specifications has been used. This change is similar to the change approved in TSTF-301, Rev. 0.

14. The Frequency of ISTS SR 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.

15. 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.

16. 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 CONDENSER 1

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>
<3.5.B Footnote (a)>
<Appl 3.5.C>
<3.5.C Footnote (a)>

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. 2

ACTIONS

	CONDITION	REQUIRED ACTION	COMPLETION TIME
<3.5.B Act 1> <DOC L.2>	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. AND C.2 Restore one required ECCS injection/spray subsystem to OPERABLE status. 5	Immediately 4 hours

(continued)

<CTS>

ACTIONS (continued)

	CONDITION	REQUIRED ACTION	COMPLETION TIME
<3.5.B Act 2> <3.5.C Act 2>	D. Required Action C.2 and associated Completion Time not met.	D.1 Initiate action to restore [secondary] containment to OPERABLE status. 2	Immediately
		AND	
		D.2 Initiate action to restore one standby gas treatment subsystem to OPERABLE status.	Immediately 2
		AND	
		D.3 Initiate action to restore isolation capability in each required [secondary] containment penetration flow path not isolated. 2	Immediately

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~~ 3

(continued)

<CTS>

SURVEILLANCE REQUIREMENTS (continued)

	SURVEILLANCE	FREQUENCY
<3.5.B.1.a.2> <3.5.B.2.b.2> <3.5.C.2> <3.5.C.2.c> <4.5.C.2>	SR 3.5.2.~~2~~ 1 [3] Verify, for each required ~~core spray (CS)~~ ECCS injection/spray [3] subsystem, the: a. Suppression pool water level is ≥ ~~12~~ 10 ft ~~2~~ 4 inches ~~]~~ [2]; or b. ----NOTE---- Only one required ~~CS~~ EC injection/spray [3] subsystem may take credit for this option during OPDRVs. ---- Contaminated [1] ~~c~~Condensate storage tank water level is ≥ ~~12 ft~~ 21 ft [2]	12 hours
<4.5.B>	SR 3.5.2.~~3~~ 2 [3] Verify, for each required ECCS injection/spray subsystem, the piping is filled with water from the pump discharge valve to the injection valve.	31 days
<4.5.B>	SR 3.5.2.~~4~~ 3 [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. ----~~ [4] Verify each required ECCS injection/spray subsystem manual, power operated, and automatic valve in the flow path, that is not locked, sealed, or otherwise secured in position, is in the correct position.	31 days

(continued)

<CTS>

SURVEILLANCE REQUIREMENTS (continued)

	SURVEILLANCE	FREQUENCY
<4.5.B> <4.5.B.2> 1 3 4 1	SR 3.5.2.~~[5]~~ 4 Verify each required ECCS pump develops the specified flow rate [against a ~~system head~~ test line pressure corresponding to the specified reactor pressure]. TEST LINE PRESSURE ~~[SYSTEM HEAD]~~ SYSTEM / FLOW RATE / NO. OF PUMPS / CORRESPONDING TO A REACTOR PRESSURE OF] CS ≥ ~~[4250]~~ 4500 gpm [1] ≥ ~~[113]~~ psig LPCI ≥ ~~[7700]~~ 4500 gpm [1] ≥ [20] psig	[In accordance with the Inservice Testing Program ~~or~~ ~~92 days~~] 90 2
<4.5.B> 3 5	SR 3.5.2.~~[6]~~ 5 -------------------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 2

1. 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.

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

3. 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.

4. ISTS SR 3.5.2.4 (ITS SR 3.5.2.3) Note for consideration of LPCI OPERABILITY when aligned for decay heat removal (RHR) has been deleted. The use of LPCI for this function is not applicable at Dresden 2 and 3.

5. The word "required" has been added consistent with its use throughout the ITS (not all ECCS subsystems are required in MODES 4 and 5).

all changes are [1] *unless otherwise identified*

<CTS>

3.5 EMERGENCY CORE COOLING SYSTEMS (ECCS) AND ~~REACTOR CORE ISOLATION COOLING (RCIC)~~ ISOLATION CONDENSER (IC) SYSTEM

3.5.3 RCIC System

<3.5.D> LCO 3.5.3 The RCIC System shall be OPERABLE.

<Appl 3.5.D> APPLICABILITY: MODE 1,
MODES 2 and 3 with reactor steam dome pressure > [150] psig. [2]

ACTIONS

	CONDITION	REQUIRED ACTION	COMPLETION TIME
<3.5 D Act>	A. RCIC System inoperable.	A.1 Verify by administrative means High Pressure Coolant Injection System is OPERABLE.	~~1 hour~~ Immediately [5] TSTF-301
		AND	
		A.2 Restore RCIC System to OPERABLE status.	14 days
<3.5.D Act>	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. [2]	36 hours

all changes are $\boxed{1}$ unless otherwise identified

<CTS>

SURVEILLANCE REQUIREMENTS

CTS		SURVEILLANCE	FREQUENCY
<4.5.D.1>	SR 3.5.3.1	Verify the RCIC System ~~piping is filled with water from the pump discharge valve to the injection valve.~~ : a. shellside water level ≥ 6 feet; and b. shellside water temperature ≤ 210 °F.	~~31 days~~ 24 hours
<4.5.D.2>	SR 3.5.3.2	Verify each RCIC System manual, power operated, and automatic valve in the flow path, that is not locked, sealed, or otherwise secured in position, is in the correct position.	31 days
	~~SR 3.5.3.3~~	~~-------------------NOTE------------------- Not required to be performed until 12 hours after reactor steam pressure and flow are adequate to perform the test. -- Verify, with [reactor pressure] ≤ [1020] psig and ≥ [920] psig, the RCIC pump can develop a flow rate ≥ [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] ≤ [165] psig, the RCIC pump can develop a flow rate ≥ [400] gpm [against a system head corresponding to reactor pressure].~~	~~[18] months~~

$\boxed{3}$

(continued)

all changes are 1 unless otherwise identified

<CTS>

SURVEILLANCE REQUIREMENTS (continued)

	SURVEILLANCE	FREQUENCY
<4.5.D.3>	SR 3.5.3.~~5~~ 3 [3] ~~-------NOTE-------~~ ~~Vessel injection may be excluded.~~ [6] Verify the ~~RC~~IC System actuates on an actual or simulated automatic initiation signal.	~~[18]~~ 24 [2] months
<4.5.D.4>	SR 3.5.3.4 Verify IC System heat removal capability to remove design heat load.	60 months [4]

JUSTIFICATION FOR DEVIATIONS FROM NUREG-1433, REVISION 1
ITS: 3.5.3 - IC 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. Brackets have been removed and proper plant specific information/values have been provided.

3. ISTS SR 3.5.3.3 and SR 3.5.3.4 have been deleted since no pump flows are applicable to the IC System. Subsequent SRs have been renumbered as required.

4. ITS SR 3.5.3.4, to verify IC System heat removal capability has been added consistent with current Technical Specification Requirements.

5. 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 the IC System 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 IC System ACTION has already expired. Thus a unit shutdown would be required immediately upon discovery of HPCI being inoperable, even though the HPCI System Required Action (ITS 3.5.1, Required Action F.1) appears to allow 1 hour to verify IC System OPERABILITY. To avoid this confusion, the original time allowed by BWR ISTS, NUREG-1433, Revision 0 and the current Dresden 2 and 3 Technical Specifications has been used. This change is similar to the change approved in TSTF-301, Rev. 0.

6. The Note in ISTS SR 3.5.3.5 (ITS SR 3.5.3.3) has been deleted since the IC System design does not provide an alternate test line and therefore actuation of the system will result in vessel injection.

B 3.5 EMERGENCY CORE COOLING SYSTEMS (ECCS) AND ~~REACTOR CORE ISOLATION COOLING (RCIC) SYSTEM~~ ISOLATION CONDENSER (IC) 1

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 ~~l~~ow ~~p~~ressure ~~c~~oolant ~~i~~njection (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 (~~C~~ST), it is capable of providing a source of water for the HPCI, LPCI and and CS systems. C 2

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/~~relief valves and safety/relief valve (S/RV~~s~~) 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. C 2

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~~ Containment Cooling Service Water System. Depending on the location and size of 2

(continued)

all changes are 2 unless otherwise identified

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.~~ 3

4 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 immediately automatically started when normal AC power is available and approximately 14 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.

The LPCI ~~is an independent operating mode of the RHR~~ subsystem System ~~. There are~~ is composed of two LPCI subsystems (loops) ~~(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. The two two, normally open, LPCI subsystems ~~can be~~ are interconnected via the ~~RHR~~ System cross tie valves. Insert BKGD-1 ~~however, the cross tie valve is maintained closed with its power removed to prevent loss of both LPCI subsystems during a LOCA.~~ 9 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 sequentially, with simultaneously and (~~C pump~~) immediately when normal AC power is available, and A and C pumps after approximately 4 seconds and B and D pumps approximately ~~(10)~~ seconds ~~after~~ when 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~~s~~. When the RPV pressure drops sufficiently, the LPCI flow to the RPV, via the

(continued)

$\boxed{2}$ 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 [2] unless otherwise identified

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~~. (C)

the reactor vessel

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. (S) (C)

150 psig to 1120 psig

steam supply

are

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 [2] unless otherwise identified

BASES

BACKGROUND (continued)

When the HPCI System is aligned to the suppression pool the "Keep fill" system must be aligned to the HPCI discharge line.

(C) 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 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 ~~7 of the 11 S/RVs~~ 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. Each of ~~the~~ S/RVs used for automatic depressurization is equipped with one air accumulator and associated inlet check valves. The accumulator provides the pneumatic power to actuate the valves.

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 $\leq$ 2200°F;

b. Maximum cladding oxidation is $\leq$ 0.17 times the total cladding thickness before oxidation;

c. Maximum hydrogen generation from a zirconium water reaction is $\leq$ 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 2 unless otherwise identified

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.

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 ECCS satisfy Criterion 3 of ~~the NRC Policy Statement~~ 10CFR50.36 (c)(2)(ii).

LCO

Each ECCS injection/spray subsystem and ~~seven~~ four 1 electromatic ADS valves are required to be OPERABLE. The SIRV 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, 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. 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

(continued)

all changes are 1 unless otherwise identified

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."

Insert ACTION A

ACTIONS

B ~~A~~.1

a LPCI subsystem is inoperable for reasons other than Condition A, or a CS subsystem is inoperable

If ~~any one~~ low pressure ECCS injection/spray ~~subsystem is inoperable~~, the inoperable subsystem must be restored to OPERABLE status within 7 days. In this Condition, the remaining OPERABLE subsystems provide adequate core cooling during a LOCA. However, overall ECCS reliability is reduced, because a single failure in one of the remaining OPERABLE subsystems, concurrent with a LOCA, may result in the ECCS not being able to perform its intended safety function. The 7 day Completion Time is based on a reliability study (Ref. ~~12~~ 11 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 (i.e., Completion Times).

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 4

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~~ IC 2 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)

[I] 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).

[I] 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 pérform 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).

all changes are 1 unless otherwise identified

BASES

ACTIONS

F

~~C~~.1 and ~~C~~.2 (continued)

conjunction with ADS. Also, the ~~RCIC~~ IC System will automatically provide ~~makeup water~~ core cooling 2 at most reactor operating pressures. Verification of ~~RCIC~~ IC OPERABILITY ~~within 1 hour~~ immediately 4 TSTF-301 TSTF-301 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~~ IC 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~~ IC System. If the OPERABILITY of the ~~RCIC~~ IC System cannot be verified, however, Condition ~~H~~ I must be immediately entered. ~~If a single active~~ In the event of 2 component fail~~s~~ures 2 concurrent with a design basis LOCA, there is a potential, depending on the specific failure(s), that the minimum required ECCS equipment will not be available. A 14 day Completion Time is based on a reliability study cited in Reference ~~12~~ 11 2 and has been found to be acceptable through operating experience.

G

~~D~~.1 and ~~D~~.2

If any one low pressure ECCS injection/spray subsystem(s), or one LPCI pump in both LPCI subsystems, TSTF-318 is inoperable in addition to an inoperable HPCI System, the inoperable low pressure ECCS injection/spray subsystem(s) 4 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) 4 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 ~~12~~ 11 2 and has been found to be acceptable through operating experience.

H

~~E~~.1

The LCO requires ~~seven~~ four ADS valves to be OPERABLE in order to provide the ADS function. Reference ~~13~~ 12 2 contains the results

(continued)

all changes are 1 unless otherwise identified

BASES

ACTIONS H E.1 (continued)

two Z S

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.

three Z 11

~~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 change not shown

I

G.1 and G.2

G H required

F

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)

I J H.1

When multiple ECCS subsystems are inoperable, as stated in Condition H, 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)

all changes are [1] unless otherwise identified

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.

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.

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.

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

(continued)

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.~~

1

SR 3.5.1.~~5~~ 3

of the correct breaker alignment to the LPCI swing

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.

5 2

SR 3.5.1.~~6~~ 4 1

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.

6

The ~~specified~~ Frequency of this SR 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~~ in accordance with the Inservice Testing Program.

1

(continued)

all changes are 1 unless otherwise identified

BASES

4

SURVEILLANCE REQUIREMENTS

SR 3.5.1.6 (continued) any recirculation pump discharge

~~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. both 5 2

5 6 7

SR 3.5.1.7, SR 3.5.1.8, and SR 3.5.1.9

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.

2 and are bounded by the requirements of SR 3.5.1.5

test line pressure or

have been

analytically

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 1.25 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

4 against a system head corresponding to reactor pressure

INSERT SR 3.5.1.6 4

5 2 6 7 4

(continued)

4 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 [1] unless otherwise identified

BASES

5 6 7

SURVEILLANCE REQUIREMENTS

SR 3.5.1.~~7~~, SR 3.5.1.~~8~~, and SR 3.5.1.~~9~~ (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. (Insert SR 3.5.1.7) [4]

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 (24) ~~18~~ month Frequency for SR 3.5.1.~~9~~ 7 is based on the need to perform the Surveillance under the conditions that apply [4] ~~just prior to or~~ during a startup from a plant outage. Operating experience has shown that these components usually pass the SR when performed at the ~~18~~ month Frequency, which is based on the refueling cycle. Therefore, the Frequency was concluded to be acceptable from a reliability standpoint. (24)

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 positions. This SR also ensures that the HPCI System will automatically restart on an RPV low – 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 CST 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. [2] (C) [4]

(24) 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.

(continued)

4 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 1 unless otherwise identified

BASES

8

SURVEILLANCE REQUIREMENTS

SR 3.5.1.10 (continued)

Operating experience has shown that these components usually pass the SR when performed at the 18 month Frequency, which is based on the refueling cycle. Therefore, the Frequency was concluded to be acceptable from a reliability standpoint. 24

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.

9

SR 3.5.1.11

valves

The ADS designated ~~S/RVs~~ are required to actuate automatically upon receipt of specific initiation signals. A system functional test is performed to demonstrate that the mechanical portions of the ADS function (i.e., solenoids) operate as designed when initiated either by an actual or simulated initiation signal, causing proper actuation of all the required components. SR 3.5.1.12 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. 10

24 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 valve actuation. 4 ~~This prevents an RPV pressure blowdown.~~

since the valves are individually tested in accordance with SR 3.5.1.10 4

10

SR 3.5.1.12

A manual actuation of each ADS valve is performed to verify that the valve and solenoid are functioning properly and

(continued)

all changes are 1 unless otherwise identified

BASES

10

SURVEILLANCE REQUIREMENTS

SR 3.5.1.12 (continued)

valve 4

that no blockage exists in the ~~S/RV~~ 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~~ (300) psig (the pressure recommended by the valve manufacturer). Adequate steam flow is represented by at least ~~1.25~~ (2) turbine bypass valves open ~~or total steam flow ≥ 10⁶ lb/hr~~. 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. 5

The Frequency of ~~18~~ (24) months ~~on a STAGGERED TEST BASIS ensures that both solenoids for each ADS valve are alternately tested. The Frequency~~ 2 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 3.5.1.10

REFERENCES

2 U 1

1. FSAR, Section 6.3.2.~~2.3~~.
2. FSAR, Section 6.3.2.2~~.4~~. 5

(continued)

1 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 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.

all changes are [2] unless otherwise identified

BASES

REFERENCES (continued)

3. FSAR, Section [6.3.2.~~2.1~~]. 3
4. FSAR, Section [6.3.2.~~2.2~~]. 4
5. ~~FSAR, Section [15.2.8].~~
6. FSAR, Section [15.6.4].
7. FSAR, Section [15.6.5]. 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]~~

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

12 13. ~~FSAR, Section [6.3.3.3].~~

EMF-97-025(P), Revision 1, LOCA Break Spectrum Analysis for Dresden 2 and 3, dated May 30, 1997.

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 Depressization 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 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. This discussion has been deleted since it discusses the RCIC System, which is not part of plant design.

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.

all changes are 2 unless otherwise identified

B 3.5 EMERGENCY CORE COOLING SYSTEMS (ECCS) AND ~~REACTOR CORE ISOLATION COOLING (RCIC)~~ ISOLATION CONDENSER (IC) SYSTEM 1

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 (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. ~~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 valves is not required to be closed.~~

(continued)

BASES

LCO (continued)

~~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)

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. 3 move from Next Page 3

1 required 1

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~~ is available ~~OPERABLE; and~~ secondary containment isolation capability (i.e., one secondary containment isolation valve and associated instrumentation are OPERABLE or other acceptable administrative controls to assure isolation capability) 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

3

3 1 (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)

(continued)

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. 3

SURVEILLANCE REQUIREMENTS

~~SR 3.5.2.1 and~~ SR 3.5.2.~~2~~ 1

10 4 4 above the bottom of the suppression chamber 2

The minimum water level of ~~12~~ ft ~~2~~ inches 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. 2 C 2

4 10 they 4 4

2 and LPCI subsystems are

When suppression pool level is < ~~12~~ ft ~~2~~ inches, 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 or that ~~CS~~ ~~is~~ aligned to take suction from the CST and the CST contains ≥ ~~150,000~~ gallons of water, equivalent to ~~12~~ ft, ensures that the ~~CS System~~ can supply at least ~~50,000~~ 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.

and LPCI pump 2 C 4 4 10 are 1 C 1 140,000 50,000 90,000 4

required low pressure ECCS injection/spray subsystems 2 21

and LPCI 1 s are C

low pressure ECCS injection/spray 1

(continued)

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 3.5.2.5, and SR 3.5.2.6

The Bases provided for SR 3.5.1.1, SR 3.5.1.7, and SR 3.5.1.10 are applicable to SR 3.5.2.3, SR 3.5.2.5, and SR 3.5.2.6, respectively.

SR 3.5.2.4

Verifying the correct alignment for manual, power operated, and automatic valves in the ECCS flow paths provides assurance that the proper flow paths will exist for ECCS operation. This SR does not apply to valves that are locked, sealed, or otherwise secured in position, since these valves were verified to be in the correct position prior to locking, sealing, or securing. A valve that receives an initiation signal is allowed to be in a nonaccident position provided the valve will automatically reposition in the proper strcke 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

3 1

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.~~ 2

REFERENCES

1 u 3.4.1 4

1. FSAR, Section [6.3.2].

JUSTIFICATION FOR DEVIATIONS FROM NUREG-1433, REVISION 1
ITS BASES: 3.5.2 - ECCS — SHUTDOWN

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

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

all changes are $\boxed{1}$ *unless otherwise identified*

B 3.5 EMERGENCY CORE COOLING SYSTEMS (ECCS) AND ~~REACTOR CORE ISOLATION COOLING (RCIC)~~ SYSTEM *ISOLATION CONDENSER (IC)*

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.~~

Insert 3.5.3 BKGD-1 $\boxed{2}$

~~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 pressure~~s~~ ~~(165 psig to 1155 psig)~~ *≥ 150* psig.

Insert 3.5.3 BKGD-2

~~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)

$\boxed{2}$ INSERT 3.5.3 BKGD-1

The IC System (Ref.1) is a passive high pressure system comprised of one natural circulation heat exchanger, two AC motor-operated isolation valves, two D.C. motor-operated isolation valves, and two tube side high point vent isolation valves to main steam line "A". The IC System functions as a heat sink for decay heat removal from the reactor vessel following reactor scram and isolation from the main condenser. This function prevents overheating of the reactor fuel, controls reactor pressure, and limits the loss of reactor coolant through the relief valves. The IC System is automatically initiated by sustained reactor vessel high pressure and, once activated, remains in operation until manually removed from service.

The isolation condenser shell contains two tube bundles. When the IC System is in operation, both tube bundles are in service.

$\boxed{2}$ INSERT 3.5.3 BKGD-2

The shell side of the condenser has a minimum water level of 6 feet which provides an inventory of ≥ 18,700 gallons. This minimum level provides ≥ 11,300 gallons (approximately 3 feet) of water above the top of the tube bundles. The shell side water temperature must be ≤ 210°F. During normal plant operations, when the system is in standby, makeup is from the clean demineralized water storage tank. Makeup during IC System operation can be provided from the Condensate Transfer System. Since during operation of the IC System, water in the shell will boil, the condenser is vented to the atmosphere via one line.

all changes are 1 unless otherwise identified

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.~~ 2

APPLICABLE SAFETY ANALYSES

The function of the RCIC System is to respond to ~~transient~~ main steam line isolation events by providing makeup coolant core cooling to the reactor. Although the RCIC System is not an Engineered Safety Feature System, and no credit is taken in the safety accident 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.~~ satisfies Criterion 4 of 10 CFR 50.36(c)(2)(ii) 3

2

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~~ reduces the loss of RPV inventory during an isolation event. 2

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~~ core cooling. 2

(continued)

all changes are 1 unless otherwise identified

BASES (continued)

ACTIONS

A.1 and A.2

TSTF-301

4

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 immediately ~~within 1 hour~~ 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 entered. For transients and certain abnormal events with no LOCA, RCIC (as opposed to HPCI) is an acceptable ~~the preferred~~ source of core cooling ~~makeup coolant because of its relatively small capacity~~, which also limits the loss ~~allows easier control~~ of the RPV water level. Therefore, a limited time is allowed to restore the inoperable RCIC to OPERABLE status.

2

2

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

4

(continued)

all changes are $\boxed{1}$ unless otherwise identified

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

$\boxed{2}$ Insert 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 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.~~ $\boxed{2}$

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)

2 INSERT SR 3.5.3.1

This SR verifies the water volume and temperature in the shell side of the IC to be sufficient for proper operation. Based on a scram from 2552.3 MWt (101% RTP), a minimum water level of 6 feet at a temperature of $\leq$ 210°F in the condenser provides sufficient decay heat removal capability for 20 minutes of operation without makeup water, before beginning to uncover the tube bundles. The volume and temperature allow sufficient time for the operator to provide makeup to the condenser.

The 24 hour Frequency is based on operating experience related to the trending of the parameter variations during normal operation.

all changes are 1 unless otherwise identified

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 is tested both at 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 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 ≥ 10^6 lb/hr]. Therefore, sufficient time is allowed after adequate pressure and flow are achieved to perform these SRs. Reactor startup is allowed prior to performing the low pressure Surveillance because the reactor pressure is low and the time 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.

A 92 day Frequency for SR 3.5.3.3 is consistent with the Inservice Testing Program requirements. The 18 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 month Frequency, which is based on the

(continued)

all changes are 1 unless otherwise identified

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~~ 3

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~~ design function. 2

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 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.~~

Insert SR 3.5.3.4

(continued)

$\boxed{1}$ INSERT 3.5.3.4

SR 3.5.3.4

Verifying the proper flow path and heat exchange capacity for IC System operation ensures the capability of the IC System to remove the design heat load. This SR verifies the IC System capability to remove heat consistent with the design requirements of 252.5 x 10^6 Btu/hr. The IC System capacity is equivalent to the decay heat rate 5 minutes after a reactor scram.

The 60 month Frequency is based on engineering judgement, and has been shown to be acceptable through operating experience.

BASES (continued)

2

REFERENCES

1. ~~10 CFR 50, Appendix A, GDC 33.~~

(1) → 2. UFSAR, Section 5.5.6. 4.6 — 4

(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 - IC SYSTEM

1. Changes have been made to reflect those changes made to the Specifications. 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 proper 10 CFR 50.36(c)(2)(ii) criterion has been used. The current wording was developed prior to the issuance of the change to 10 CFR 50.36, which uses criterion 4 for the current words of the NUREG.

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

GENERIC NO SIGNIFICANT HAZARDS CONSIDERATION
ITS: SECTION 3.5 - ECCS AND IC SYSTEM

ADMINISTRATIVE CHANGES
("A.x" Labeled Comments/Discussions)

In accordance with the criteria set forth in 10 CFR 50.92, ComEd has evaluated this proposed Technical Specifications change and determined it does not represent a significant hazards consideration. The following is provided in support of this conclusion.

1. Does the change involve a significant increase in the probability or consequences of an accident previously evaluated?

   The proposed change involves reformatting, renumbering, and rewording the existing Technical Specifications. The reformatting, renumbering, and rewording process involves no technical changes to the existing Technical Specifications. As such, this change is administrative in nature and does not impact initiators of analyzed events or assumed mitigation of accident or transient events. Therefore, this change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. Does the change create the possibility of a new or different kind of accident from any accident previously evaluated?

   The proposed change does not involve a physical alteration of the plant (no new or different type of equipment will be installed) or changes in methods governing normal plant operation. The proposed change will not impose any new or eliminate any old requirements. Thus, this change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. Does this change involve a significant reduction in a margin of safety?

   The proposed change will not reduce a margin of safety because it has no impact on any safety analyses assumptions. This change is administrative in nature. Therefore, the change does not involve a significant reduction in a margin of safety.

TECHNICAL CHANGES - MORE RESTRICTIVE
("M.x" Labeled Comments/Discussions)

In accordance with the criteria set forth in 10 CFR 50.92, ComEd has evaluated this proposed Technical Specifications change and determined it does not represent a significant hazards consideration. The following is provided in support of this conclusion.

1. Does the change involve a significant increase in the probability or consequences of an accident previously evaluated?

   The proposed change provides more stringent requirements for operation of the facility. These more stringent requirements do not result in operation that will increase the probability of initiating an analyzed event and do not alter assumptions relative to mitigation of an accident or transient event. The more restrictive requirements continue to ensure process variables, structures, systems, and components are maintained consistent with the safety analyses and licensing basis. Therefore, this change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. Does the change create the possibility of a new or different kind of accident from any accident previously evaluated?

   The proposed change does not involve a physical alteration of the plant (no new or different type of equipment will be installed) or changes in the methods governing normal plant operation. The proposed change does impose different requirements. However, these changes are consistent with the assumptions in the safety analyses and licensing basis. Thus, this change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. Does this change involve a significant reduction in a margin of safety?

   The imposition of more restrictive requirements either has no impact on or increases the margin of plant safety. As provided in the discussion of the change, each change in this category is by definition, providing additional restrictions to enhance plant safety. The change maintains requirements within the safety analyses and licensing basis. Therefore, this change does not involve a significant reduction in a margin of safety.

GENERIC NO SIGNIFICANT HAZARDS CONSIDERATION
ITS: SECTION 3.5 - ECCS AND IC SYSTEM

"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.

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 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 - IC SYSTEM

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

ENVIRONMENTAL ASSESSMENT
ITS: SECTION 3.5 - ECCS AND IC 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.