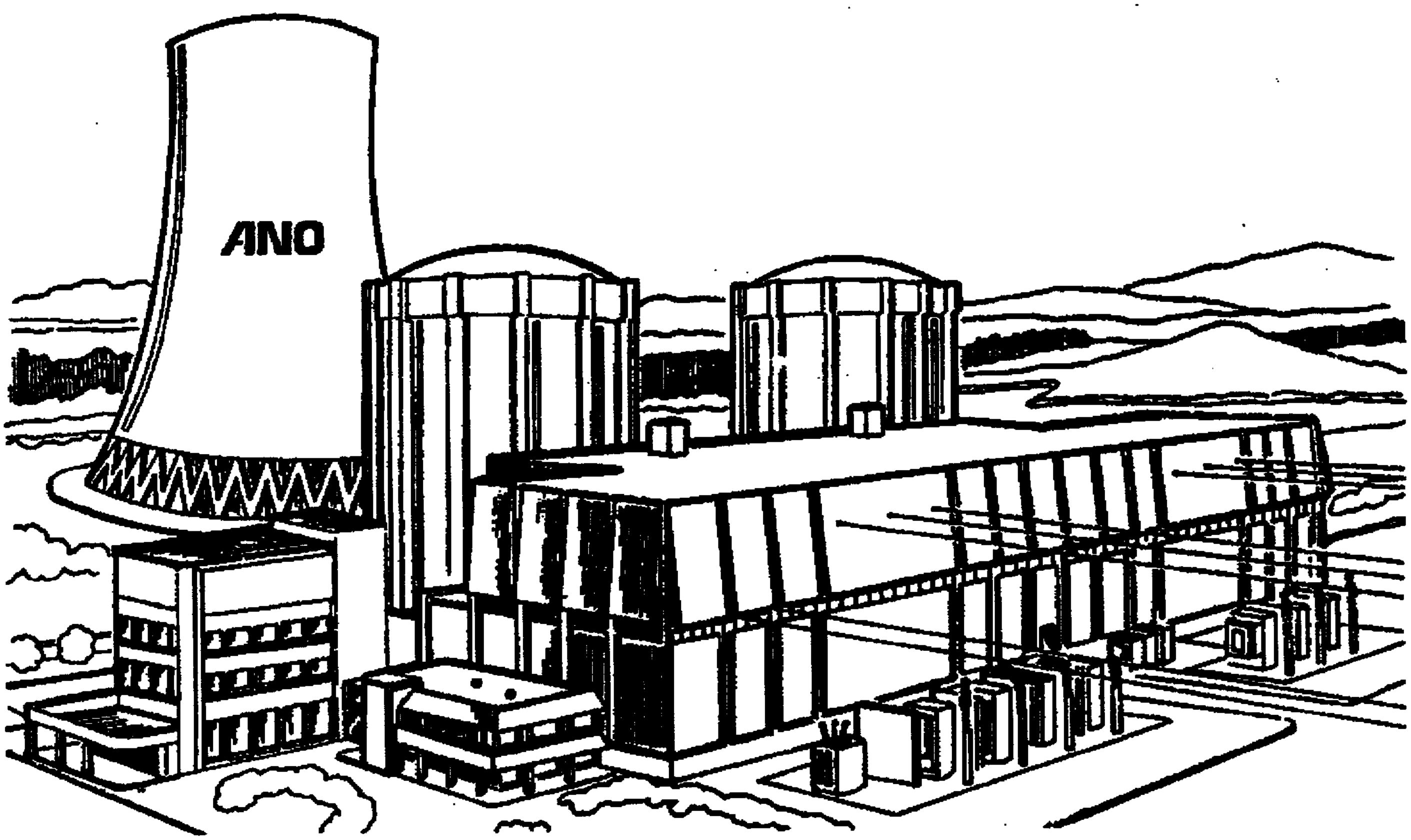
ARKANSAS NUCLEAR ONE - UNIT 1

IMPROVED TECHNICAL SPECIFICATIONS SUBMITTAL



03/19/01 Supplement
Volume 1 of 2
(Sections 3.4A, 3.4B, and 3.5)

Entergy

March 19, 2001

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.1 RCS Pressure, Temperature, and Flow Departure from Nucleate Boiling (DNB) Limits

LCO 3.4.1 RCS DNB parameters (loop pressure, hot leg temperature, and RCS total flow rate) shall be within the limits specified in the COLR.

APPLICABILITY: MODE 1.

---NOTE---
RCS loop pressure limit does not apply during pressure transients due to a THERMAL POWER change > 5% RTP per minute.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more RCS DNB parameters not within limits.	A.1 Restore RCS DNB parameter(s) to within limit.	2 hours
B. Required Action and associated Completion Time not met.	B.1 Be in MODE 2.	6 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.4.1.1	---NOTE--- With three RCPs operating, the limits are applied to the loop with two RCPs in operation. --- Verify RCS loop pressure is within the limit specified in the COLR.	12 hours

	SURVEILLANCE	FREQUENCY
SR 3.4.1.2	-----------------------------NOTE----------------------------- With three RCPs operating, the limits are applied to the loop with two RCPs in operation. --- Verify RCS hot leg temperature is within the limit specified in the COLR.	12 hours
SR 3.4.1.3	Verify RCS total flow is within the limit specified in the COLR.	12 hours
SR 3.4.1.4	-----------------------------NOTE----------------------------- Only required to be performed when stable thermal conditions are established at ≥ 90% RTP. --- Verify RCS total flow rate is within the limit specified in the COLR by measurement.	18 months

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.2 RCS Minimum Temperature for Criticality

LCO 3.4.2 The RCS average temperature (T_{avg}) shall be $\geq$ 525°F.

APPLICABILITY: MODE 1 and 2.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. T_{avg} not within limit.	A.1 Be in MODE 3.	30 minutes

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
SR 3.4.2.1 Verify RCS $T_{avg} \geq$ 525°F.	12 hours

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.3 RCS Pressure and Temperature (P/T) Limits

LCO 3.4.3 RCS pressure, RCS temperature, and RCS heatup and cooldown rates shall be maintained within limits specified in Figures 3.4.3-1, 3.4.3-2, and 3.4.3-3.

-----------------------------NOTE-----------------------------
Not applicable to the pressurizer.

APPLICABILITY: At all times.

ACTIONS

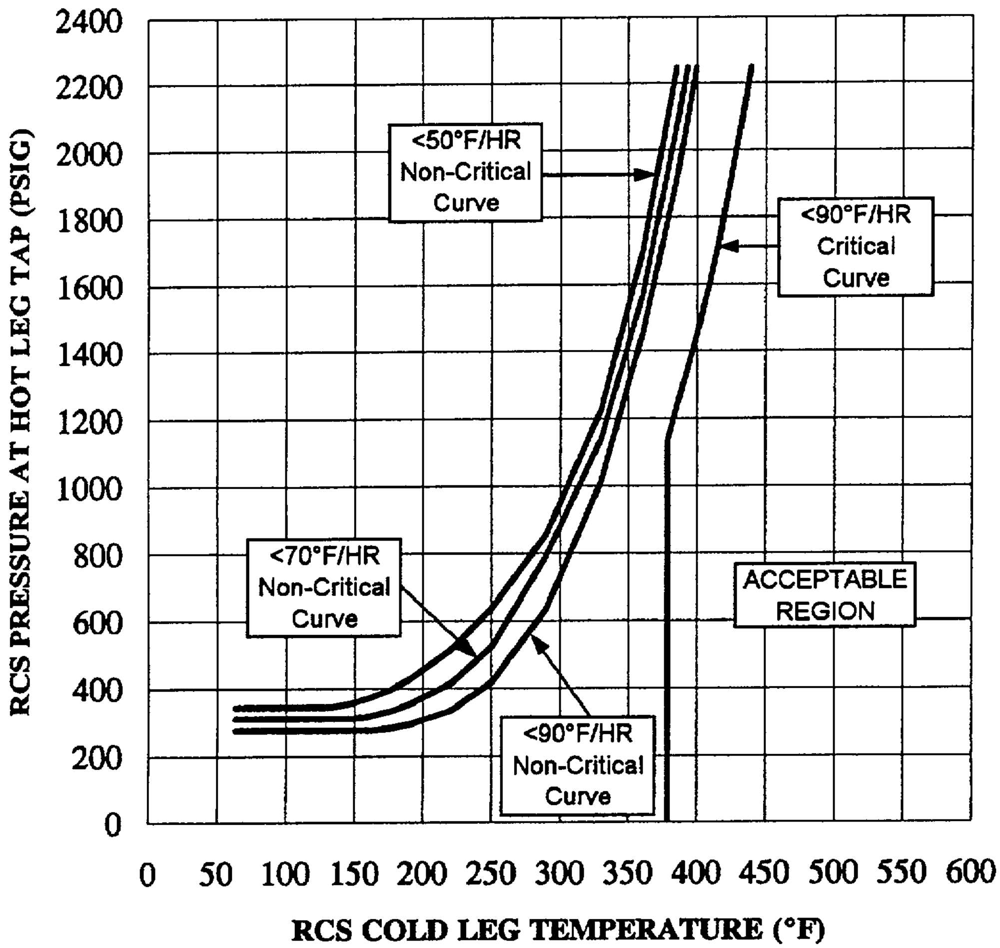
CONDITION	REQUIRED ACTION	COMPLETION TIME
A. RCS Pressure and Temperature not within criticality limit of Figure 3.4.3-1 during PHYSICS TESTS with RCS temperature $\leq$ 525°F.	A.1 Be in MODE 3.	30 minutes
B. --------------NOTE------------ Required Action B.2 shall be completed whenever this Condition is entered. ---------------------------------- Requirements of LCO not met in MODE 1, 2, 3, or 4.	B.1 Restore parameter(s) to within limits. <u>AND</u> B.2 Determine RCS is acceptable for continued operation.	30 minutes 72 hours
C. Required Action and associated Completion Time of Condition B not met.	C.1 Be in MODE 3. <u>AND</u> C.2 Be in MODE 5.	6 hours 36 hours

CONDITION	REQUIRED ACTION		COMPLETION TIME
D. --------------NOTE------------ Required Action D.2 shall be completed whenever this Condition is entered. ---------------------------------- Requirements of LCO not met in other than MODE 1, 2, 3, or 4.	D.1	Initiate action to restore parameter(s) to within limit.	Immediately
	<u>AND</u>		
	D.2	Determine RCS is acceptable for continued operation.	Prior to entering MODE 4

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.4.3.1	------------------------------NOTE------------------------------ Only required to be performed during RCS heatup operations with fuel in the reactor vessel. --- Verify RCS pressure, RCS temperature, and RCS heatup rates are within the limits specified in Figure 3.4.3-1.	30 minutes
SR 3.4.3.2	------------------------------NOTE------------------------------ Only required to be performed during RCS cooldown operations with fuel in the reactor vessel. --- Verify RCS pressure, RCS temperature, and RCS cooldown rates are within the limits specified in Figure 3.4.3-2.	30 minutes
SR 3.4.3.3	------------------------------NOTE------------------------------ Only required to be performed during RCS heatup and cooldown operations with no fuel in the reactor vessel. --- Verify RCS pressure, RCS temperature, and RCS cooldown rates are within the limits specified in Figure 3.4.3-3.	30 minutes

SURVEILLANCE		FREQUENCY
SR 3.4.3.4	------------------------------NOTE------------------------------ Only required to be performed during PHYSICS TESTS with RCS temperature $\leq$ 525°F. --- Verify RCS pressure and RCS temperature are within the criticality limits specified in Figure 3.4.3-1.	 30 minutes



Notes:

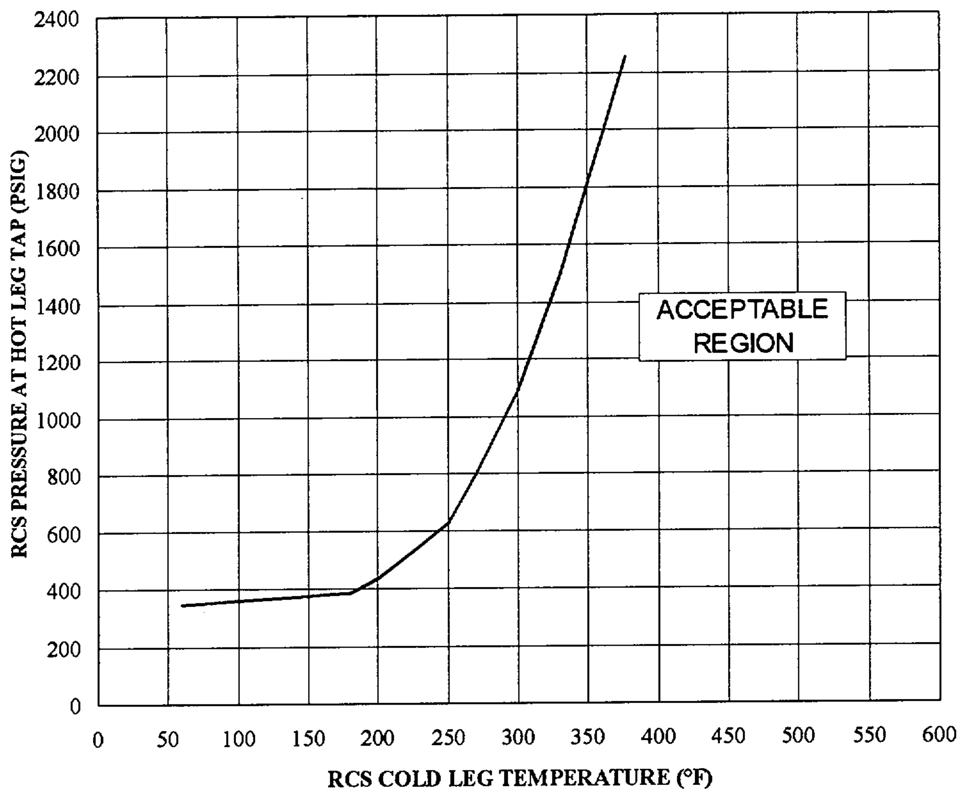
1. These curves are not adjusted for instrument error and shall not be used for operation.
2. When DHR is in operation with no RCPs operating, the DHR system return temperature shall be used.
3. RCP Operating Restrictions:

RCS TEMP	RCP RESTRICTIONS
T > 300°F	None
300°F ≥ T ≥ 225°F	≤ 3
225°F > T ≥ 84°F	≤ 2
T < 84°F	No RCPs operating

4. Allowable Heatup Rates:

RCS TEMP	H/U RATE
60°F < T ≤ 84°F	≤ 15°F/HR
T > 84°F	As allowed by applicable curve

FIGURE 3.4.3-1 (page 1 of 1)
RCS Heatup Limitations to 31 EFPY



Notes:

1. This curve is not adjusted for instrument error and shall not be used for operation.
2. A maximum step temperature change of 25°F is allowable when securing all RCPs with the DHR system in operation. This change is defined as the RCS temperature prior to securing all the RCPs minus the DHR return temperature after the RCPs are secured. When DHR is in operation with no RCPs operating, the DHR system return temperature shall be used.
3. RCP Operating Restrictions:

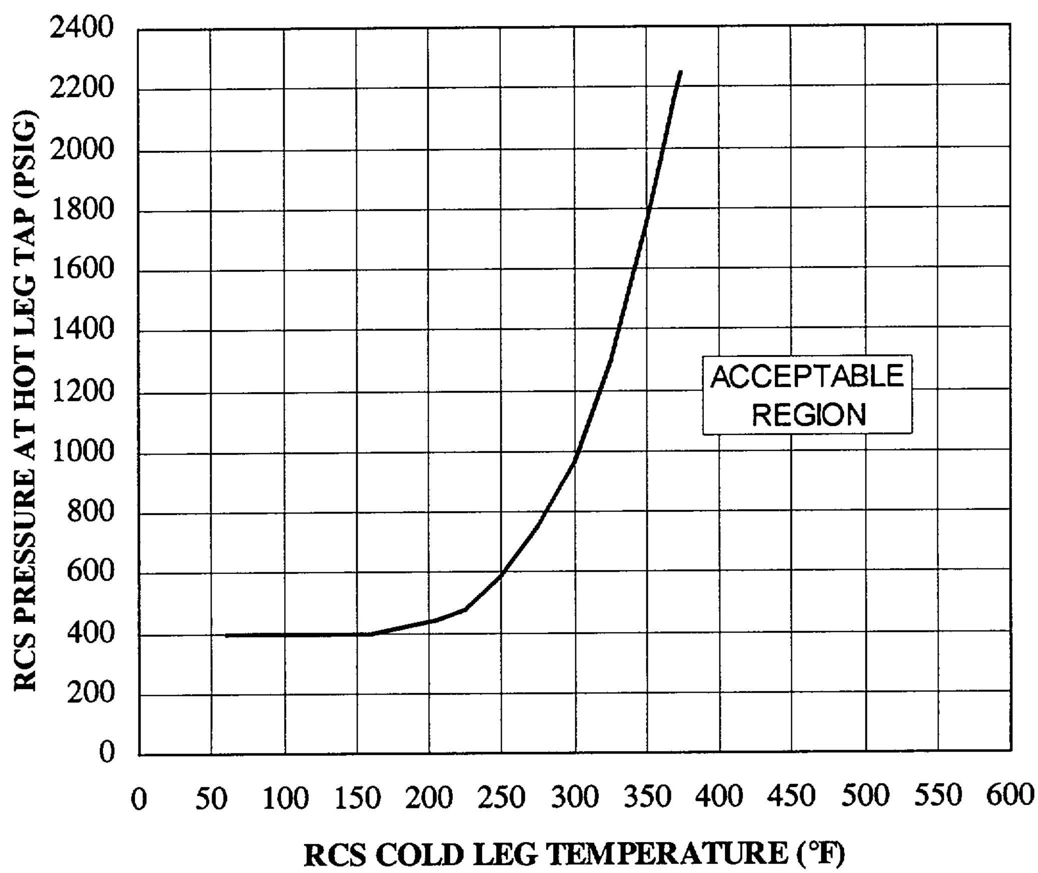
RCS TEMP	RCP RESTRICTIONS
T > 255°F	None
150°F ≤ T ≤ 255°F	≤ 2 (See Note 5)
T < 150°F	No RCPs operating

4. Allowable Cooldown Rates:

RCS TEMP	C/D RATE	STEP CHANGE
T ≥ 280°F	100°F/HR	≤ 50°F in any 1/2 HR
280°F > T ≥ 150°F	50°F/HR (See Note 5)	≤ 25°F in any 1/2 HR
T < 150°F	25°F/HR	≤ 25°F in any 1 HR

5. If RCPs are operated < 200°F, then the RCS cooldown rate from 150°F ≤ T ≤ 180°F is reduced to 30°F in 15 hours.

FIGURE 3.4.3-2 (page 1 of 1)
RCS Cooldown Limits to 31 EFPY



Notes:

1. This curve is not adjusted for instrument error and shall not be used for operation.

2. All Notes on Figure 3.4.3-1 are applicable for heatups. This curve is based on a heatup rate of < 90°F/HR.

3. All Notes on Figure 3.4.3-2 are applicable for cooldowns.

FIGURE 3.4.3-3 (page 1 of 1)
RCS Inservice Hydrostatic Test H/U & C/D Limits to 31 EFPY

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.4 RCS Loops - MODES 1 and 2

LCO 3.4.4 Two RCS Loops shall be in operation, with:

a. Four reactor coolant pumps (RCPs) operating; or

b. Three RCPs operating and THERMAL POWER restricted as specified in the COLR.

APPLICABILITY: MODES 1 and 2.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One RCP not in operation in each loop.	A.1 Restore one non-operating RCP to operation.	18 hours
B. Required Action and associated Completion Time of Condition A not met. OR LCO not met for reasons other than Condition A.	B.1 Be in MODE 3.	6 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.4.4.1	Verify required RCS loops are in operation.	12 hours

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.5 RCS Loops - MODE 3

LCO 3.4.5 Two RCS loops shall be OPERABLE and one OPERABLE RCS loop shall be in operation.

--------------------NOTES--------------------
All reactor coolant pumps (RCPs) may be removed from operation for $\leq$ 8 hours per 24 hour period for the transition to or from the Decay Heat Removal System, and all RCPs may be removed from operation for $\leq$ 1 hour per 8 hour period for any other reason, provided:

a. No operations are permitted that would cause introduction into the RCS, coolant with boron concentration less than required to meet the SDM of LCO 3.1.1; and

b. Core outlet temperature is maintained at least 10°F below saturation temperature.

APPLICABILITY: MODE 3.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One RCS loop inoperable.	A.1 Restore RCS loop to OPERABLE status.	72 hours
B. Required Action and associated Completion Time of Condition A not met.	B.1 Be in MODE 4.	12 hours

CONDITION	REQUIRED ACTION		COMPLETION TIME
C. Two RCS loops inoperable. <u>OR</u> Required RCS loop not in operation.	C.1	Suspend operations that would cause introduction into the RCS, coolant with boron concentration less than required to meet SDM of LCO 3.1.1.	Immediately
	<u>AND</u>		
	C.2	Initiate action to restore one RCS loop to OPERABLE status and operation.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.4.5.1	Verify required RCS loop is in operation.	12 hours
SR 3.4.5.2	-------------------------------NOTE------------------------------- Not required to be performed until 24 hours after a required pump is not in operation. -- Verify correct breaker alignment and indicated power available to each required pump.	7 days

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.6 RCS Loops - MODE 4

LCO 3.4.6 Two loops consisting of any combination of RCS loops and decay heat removal (DHR) loops shall be OPERABLE and one OPERABLE loop shall be in operation.

--NOTE--
All reactor coolant pumps (RCPs) and DHR pumps may be removed from operation for ≤ 1 hour provided:

a. No operations are permitted that would cause introduction into the RCS, coolant with boron concentration less than required to meet the SDM of LCO 3.1.1; and

b. Core outlet temperature is maintained at less than or equal to a temperature which is 10°F below saturation temperature.

APPLICABILITY: MODE 4.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One required loop inoperable.	A.1 Initiate action to restore a second loop to OPERABLE status.	Immediately
	<u>AND</u>	
	A.2 --------------NOTE-------------- Only required if DHR loop is OPERABLE. -------------------------------------- Be in MODE 5.	24 hours

CONDITION	REQUIRED ACTION		COMPLETION TIME
B. Two required loops inoperable. OR Required loop not in operation.	B.1	Suspend operations that would cause introduction into the RCS, coolant with boron concentration less than required to meet SDM of LCO 3.1.1.	Immediately
	AND		
	B.2	Initiate action to restore one loop to OPERABLE status and operation.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.4.6.1	Verify required DHR or RCS loop is in operation.	12 hours
SR 3.4.6.2	------------------------------NOTE------------------------------ Not required to be performed until 24 hours after a required pump is not in operation. -- Verify correct breaker alignment and indicated power available to each required pump.	7 days

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.7 RCS Loops - MODE 5, Loops Filled

LCO 3.4.7 One decay heat removal (DHR) loop shall be OPERABLE and in operation, and either:

a. One additional DHR loop shall be OPERABLE; or

b. The secondary side of each steam generator (SG) shall be $\geq$ 20 inches.

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

1. The DHR pump of the loop in operation may be removed from operation for $\leq$ 1 hour provided:

    a. No operations are permitted that would cause introduction into the RCS, coolant with boron concentration less than required to meet the SDM of LCO 3.1.1; and

    b. Core outlet temperature is maintained at less than or equal to a temperature which is 10°F below saturation temperature.

2. One required DHR loop may be inoperable for $\leq$ 2 hours for surveillance testing provided that the other DHR loop is OPERABLE and in operation.

3. All DHR loops may be removed from operation during planned heatup to MODE 4 when at least one RCS loop is in operation.

APPLICABILITY: MODE 5 with RCS loops filled.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One required DHR loop inoperable. AND One DHR loop OPERABLE.	A.1 Initiate action to restore a second DHR loop to OPERABLE status.	Immediately
	OR	
	A.2 Initiate action to restore required SGs secondary side water level to within limit.	Immediately
B. One or more required SGs with secondary side water level not within limit AND One DHR loop OPERABLE.	B.1 Initiate action to restore a second DHR loop to OPERABLE status.	Immediately
	OR	
	B.2 Initiate action to restore required SGs secondary side water level to within limit.	Immediately
C. No required DHR loop OPERABLE. OR Required DHR loop not in operation.	C.1 Suspend operations that would cause introduction into the RCS, coolant with boron concentration less than required to meet SDM of LCO 3.1.1.	Immediately
	AND	
	C.2 Initiate action to restore one DHR loop to OPERABLE status and operation.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.4.7.1	Verify required DHR loop is in operation.	12 hours
SR 3.4.7.2	Verify required SG secondary side water levels are ≥ 20 inches.	12 hours
SR 3.4.7.3	------------------------------NOTE------------------------------ Not required to be performed until 24 hours after a required pump is not in operation. -- Verify correct breaker alignment and indicated power available to each required DHR pump.	7 days

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.8 RCS Loops - MODE 5, Loops Not Filled

LCO 3.4.8 Two decay heat removal (DHR) loops shall be OPERABLE and one OPERABLE DHR loop shall be in operation.

-----NOTES-----

1. All DHR pumps may be removed from operation for ≤ 1 hour provided:
   a. No operations are permitted that would cause introduction into the RCS, coolant with boron concentration less than required to meet the SDM of LCO 3.1.1; and
   b. No draining operations to further reduce the RCS water volume are permitted.
2. One DHR loop may be inoperable for ≤ 2 hours for surveillance testing provided that the other DHR loop is OPERABLE and in operation.

APPLICABILITY: MODE 5 with RCS loops not filled.

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. One DHR loop inoperable.	A.1	Initiate action to restore DHR loop to OPERABLE status.	Immediately

CONDITION	REQUIRED ACTION		COMPLETION TIME
B. No required DHR loop OPERABLE. <u>OR</u> Required DHR loop not in operation.	B.1	Suspend operations that would cause introduction into the RCS, coolant with boron concentration less than required to meet SDM of LCO 3.1.1.	Immediately
	<u>AND</u>		
	B.2	Suspend all operations involving reduction in RCS water volume.	Immediately
	<u>AND</u>		
	B.3	Initiate action to restore one DHR loop to OPERABLE status and operation.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.4.8.1	Verify required DHR loop is in operation.	12 hours
SR 3.4.8.2	-----------------------------NOTE----------------------------- Not required to be performed until 24 hours after a required pump is not in operation. --- Verify correct breaker alignment and indicated power available to each required DHR pump.	7 days

B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.1 RCS Pressure, Temperature, and Flow Departure from Nucleate Boiling (DNB) Limits

BASES

BACKGROUND

These Bases address requirements for maintaining RCS pressure, temperature, and flow rate within limits assumed in the safety analyses. The safety analyses (Ref. 1) of normal operating conditions and abnormalities assume initial conditions within the normal steady state envelope. The limits placed on DNB related parameters ensure that these parameters will not be less conservative than were assumed in the analyses and thereby provide assurance that the minimum departure from nucleate boiling ratio (DNBR) will meet the required criteria for each of the transients analyzed.

The LCO for minimum RCS pressure is consistent with that used as the initial pressure in the analyses. Considering only pressure, a pressure greater than the minimum specified will produce a higher DNBR; and a pressure lower than the minimum specified will produce a lower DNBR.

The LCO for maximum RCS coolant hot leg temperature is consistent with the initial hot leg temperature in the analyses. Considering only temperature, a hot leg temperature lower than that specified will produce a higher DNBR; and a temperature higher than that specified will produce a lower DNBR.

The RCS flow rate is not expected to vary during operation with all pumps running. The LCO for the minimum RCS flow rate corresponds to that assumed for the DNBR analyses. Considering only flow rate, a higher RCS flow rate than that specified will produce a higher DNBR; and a lower RCS flow rate will produce a lower DNBR.

APPLICABLE SAFETY ANALYSES

The requirements of LCO 3.4.1 represent the initial conditions for DNB limited transients analyzed in the plant safety analyses (Refs. 1 and 2). The safety analyses have shown that transients initiated from the limits of this LCO will meet the DNBR criteria of ≥ 1.30 or ≥ 1.18, for the BAW-2 or the BWC critical heat flux correlation, respectively. For the locked rotor accident, the minimum DNB ratio is not less than applicable critical heat flux correlation limit, or fuel cladding is shown to experience no significant temperature excursions. These are the acceptance criteria for the RCS DNBR parameters. Changes to the facility that could impact these parameters must be assessed for their impact on the DNBR criterion. The transients analyzed include loss of coolant flow events and dropped or stuck control rod events. A key assumption for the analysis of these events is that the core

power distribution is within the limits of LCO 3.2.1, "Regulating Rod Insertion Limits," LCO 3.2.3, "AXIAL POWER IMBALANCE Operating Limits," LCO 3.2.4, "QUADRANT POWER TILT (QPT)," LCO 3.1.4, "CONTROL ROD Group Alignment Limits," and LCO 3.1.6, "AXIAL POWER SHAPING ROD (APSR) Alignment Limits."

The safety analyses to establish reload operating limits are performed using nominal values for RCS coolant average temperature, core outlet pressure, and RCS flow rate and core power level with appropriate application of associated uncertainty. Consistent with Statistical Core Design (SCD) methodology, applicable random parametric uncertainties are combined statistically. As necessary, bias parameters are included deterministically. The RCS temperature and pressure are measured in the hot leg. The surveillance criteria specified in the COLR include adjustment for measurement location. The COLR specified hot leg temperature is the maximum allowed so that the analysis value is not exceeded. The COLR specified hot leg pressure and flow are the minimum allowed so that the analysis values are not exceeded.

Analyses have been performed to establish the pressure, temperature, and flow rate requirements for two pump, three pump and four pump operation. The flow limits for two pump and three pump operation are substantially lower than for four pump operation. To meet the DNBR criterion, a corresponding maximum power limit is required (see Bases for LCO 3.4.4, "RCS Loops-MODES 1 and 2").

The steady state limits on DNBR related parameters are provided in Safety Limit (SL) 2.1.1, "Reactor Core SLs." Those limits are less restrictive on plant operations than the limits of LCO 3.4.1, but violation of an SL merits a stricter, more severe Required Action. Should a violation of LCO 3.4.1 occur, a check must be performed to determine whether an SL may have been exceeded.

The RCS DNB limits satisfy Criterion 2 of 10 CFR 50.36 (Ref. 4).

LCO

This LCO specifies limits on the monitored process variables: RCS loop (hot leg) pressure, RCS hot leg temperature, and RCS total flow rate to ensure that the core operates within the limits assumed for the plant safety analyses. Operating within these limits will result in meeting DNBR criteria in the event of a DNB limited transient.

The pressure and temperature limits are to be applied to the loop with two reactor coolant pumps (RCPs) running for the three RCPs operating condition.

The surveillance criteria for pressure, temperature, and flow rate as specified in the COLR have been appropriately adjusted for the measurement location and for instrument error consistent with supporting analysis.

APPLICABILITY

In MODE 1, the limits on RCS pressure, RCS hot leg temperature, and RCS flow rate must be maintained during steady state operation in order to ensure that DNBR criteria will be met in the event of an unplanned loss of forced coolant flow or other DNB limited transient. In all other MODES the power level is low enough so that DNB is not a significant concern.

The Note indicates the limit on RCS pressure may be exceeded during short term operational pressure transients resulting from a THERMAL POWER change > 5% RTP per minute. These conditions represent short term perturbations where actions to control pressure variations might be counterproductive. Also, for transients initiated from power levels less than the Allowable Thermal Power, increased DNBR margin exists to offset the temporary pressure variations.

ACTIONS

A.1

Loop pressure and hot leg coolant temperature are controllable and measurable parameters. With one or both of these parameters not within the LCO limits, action must be taken to restore the parameters. RCS flow rate is not a controllable parameter and is not expected to vary during steady state operation. However, if the flow rate is below the LCO limit, the parameter must be restored to within limits or power must be reduced as required in Required Action B.1, to eliminate the potential for violation of the minimum DNBR limit.

The 2 hour Completion Time for restoration of the parameters provides sufficient time to adjust plant parameters, determine the cause for the off normal condition, and restore the readings within limits. The Completion Time is based on plant operating experience.

B.1

If the Required Action and associated Completion Time are not met, the plant must be brought to a MODE in which the LCO does not apply. To achieve this status, the plant must be brought to at least MODE 2 within 6 hours. In MODE 2, the reduced power condition eliminates the potential for violation of the accident analysis assumptions.

The 6 hour Completion Time is reasonable, based on operating experience, to reach MODE 2 from full power conditions in an orderly manner and without challenging safety systems.

SURVEILLANCE REQUIREMENTS

SR 3.4.1.1

The RCS pressure value specified is dependent on the number of pumps in operation and has been adjusted to account for the pressure difference between the core exit and the measurement location. The 12 hour interval has been shown by operating practice to be sufficient to regularly assess potential degradation and to verify operation is within safety analysis assumptions.

A Note has been added to indicate the pressure limits are to be applied to the loop with two pumps in operation for the three pump operating condition.

SR 3.4.1.2

The 12 hour interval has been shown by operating practice to be sufficient to regularly assess potential degradation and to verify that operation is within safety analysis assumptions.

A Note has been added to indicate the temperature limits are to be applied to the loop with two pumps in operation for the three pump operating condition.

SR 3.4.1.3

The 12 hour Surveillance Frequency for RCS total flow rate is performed using the available flow indications. The 12 hour interval has been shown by operating practice to be sufficient to regularly assess potential degradation and to verify that operation is within safety analysis assumptions.

SR 3.4.1.4

Measurement of RCS total flow rate once every 18 months allows the installed RCS flow instrumentation to be calibrated and verifies that the actual RCS flow is greater than or equal to the minimum required RCS flow rate specified in the COLR.

The Frequency of 18 months reflects the importance of verifying flow after a refueling outage when the core has been altered or RCS flow characteristics may have been modified, which may have caused change of flow.

The Surveillance is modified by a Note that indicates the SR does not need to be performed until stable thermal conditions are established at higher power levels (i.e., ≥ 90% RTP). The Note provides for measurement of the flow rate at normal operating conditions at power in MODE 1. The Surveillance may be performed at low power or in MODE 2 or below. However, at low or zero power conditions, the indications are less accurate and significant penalties for uncertainties may be

necessary. Performance of the calorimetric heat balance at a high power level and normal operating conditions provides for the most accurate flow verification.

REFERENCES

1. SAR, Chapter 14.
2. SAR, Section 3A.6.
3. BAW-10179P-A, 2/96.
4. 10 CFR 50.36.

B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.2 RCS Minimum Temperature for Criticality

BASES

BACKGROUND

Establishing the value for the minimum temperature for reactor criticality is based upon considerations for:

a. Operation within the existing instrumentation ranges;

b. Operation with reactor vessel above its minimum nil ductility reference temperature when the reactor is critical.

The reactor coolant moderator temperature coefficient used in core operating and accident analysis is typically defined for the normal (average) operating temperature range (532°F to 579°F). The Reactor Protection System (RPS) receives inputs from the narrow range hot leg temperature detectors, which have a range of 520°F to 620°F. The integrated control system controls average temperature (Tavg) using inputs of the same range. Nominal Tavg for making the reactor critical is 532°F. Safety and operating analyses for lower temperatures have not been performed for all possible scenarios.

APPLICABLE SAFETY ANALYSES

There are no accident analyses that dictate the minimum temperature for criticality, but all low power safety analyses assume initial temperatures near the 525°F limit (Ref. 1).

The RCS minimum temperature for criticality satisfies Criterion 2 of 10 CFR 50.36 (Ref. 2).

LCO

Compliance with the LCO ensures that the reactor will not be made or maintained critical at a temperature significantly less than the hot zero power (HZP) temperature, which is assumed in the safety analysis. Failure to meet the requirements of this LCO may produce initial conditions inconsistent with the initial conditions assumed in the safety analysis.

The LCO limit of 525°F has been selected to be within the instrument indicating range (520°F to 620°F). This parameter value is considered to be a nominal value. No additional allowances for instrument uncertainty are required in the implementing procedures.

APPLICABILITY

The reactor has been designed and analyzed to be critical in MODES 1 and 2 only with Tavg ≥ 525°F. Criticality is not permitted in any other MODE. Therefore, this LCO is applicable in MODE 1 and MODE 2.

ACTIONS

A.1

With Tavg below 525°F, 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 in 30 minutes. Rapid reactor shutdown can be readily and practically achieved in a 30 minute period. The Completion Time reflects the ability to perform this Action and maintain the plant within the analyzed range. If Tavg can be restored within the 30 minute time period, shutdown is not required.

SURVEILLANCE REQUIREMENTS

SR 3.4.2.1

RCS average temperature is required to be verified at or above 525°F every 12 hours. The SR to verify RCS average temperature every 12 hours takes into account indications that are continuously available to the operator in the control room and is consistent with other routine surveillances which are typically performed once per shift. In addition, Operators are trained to be sensitive to RCS temperature during approach to criticality and will ensure that the minimum temperature for criticality is met as criticality is approached.

RCS T_{avg} is normally calculated as the average of the unit T_{hot} (hot temperature average of loops A and B) and the unit T_{cold} (cold temperature average of loops A and B). During operation with 3 RCPs in operation, T_{avg} is calculated as the average of the loop T_{hot} and loop T_{cold} in the loop that has 2 RCPs running.

REFERENCES

1. SAR, Chapter 14.

2. 10 CFR 50.36.

B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.3 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, and unit transients. This LCO limits the pressure and temperature changes during RCS heatup and cooldown, within the design assumptions and the stress limits for cyclic operation.

Figures 3.4.3-1, 3.4.3-2, and 3.4.3-3 contain P/T limit curves for heatup, cooldown, inservice hydrostatic testing, and physics testing at RCS temperatures ≤ 525°F, and the maximum rate of change of reactor coolant temperature. The methods and criteria employed to establish operating pressure and temperature limits are described in BAW-10046A (Ref. 1). These limit curves are applicable through thirty-one effective full power years (EFPY) of operation. The pressure limit is adjusted for the pressure differential between the point of system pressure measurement and the limiting component for the various operating reactor coolant pump combinations.

Each P/T curve defines an acceptable region for normal operation below and to the right of the limit curve. The curves are used to develop operational guidance for use during heatup or cooldown maneuvering.

The LCO establishes operating limits that provide a margin to brittle failure of the reactor vessel. The vessel is the component most subject to brittle failure due to the fast neutron embrittlement it experiences during power operation, and the LCO limits apply mainly to the vessel. The limits do not apply to the pressurizer, which has different design characteristics and operating functions.

10 CFR 50, Appendix G (Ref. 2), requires the establishment of P/T limits for material fracture toughness requirements of the reactor coolant pressure boundary (RCPB) materials. Reference 2 requires an adequate margin to brittle failure during normal operation, abnormalities, and system hydrostatic tests. It mandates the use of the American Society of Mechanical Engineers (ASME), Boiler and Pressure Vessel Code, Section III, Appendix G (Ref. 3).

Linear elastic fracture mechanics (LEFM) methodology is used to determine the stresses and material toughness at locations within the RCPB. The LEFM methodology follows the guidance given by 10 CFR 50, Appendix G; ASME Code, Section III, Appendix G; and Regulatory Guide 1.99 (Ref. 4).

The major components of the reactor coolant pressure boundary have been analyzed in accordance with Appendix G to 10CFR50. Results of this analysis,

including the actual pressure-temperature limitations of the reactor coolant pressure boundary, are given in FTI Document 77-1258569-01 (Ref. 5). The service period was reduced by one effective full power year from that assumed in Reference 5 to be conservative with respect to independent calculations performed by the NRC staff. The limiting weld material is being irradiated as part of the B&W Owners Group Integrated Reactor Vessel Material Surveillance Program and the identification and locations of the capsules containing the limiting weld material is discussed in the latest revision to B&W report, BAW-1543 (Rev. 6). The chemical composition of the limiting weld material is reported in the B&W report, BAW-2121P (Rev. 7). The effect of neutron irradiation on the nil ductility reference temperature (RT_{NDT}) of the limiting weld material is reported in FTI Calculations 32-1245917-00 and 32-1257716-00 (Rev. 8).

The actual shift in the RT_{NDT} of the vessel beltline region material will be established periodically by removing and evaluating the irradiated reactor vessel material surveillance specimens, in accordance with Appendix H of 10 CFR 50 (Ref. 9). These specimens are installed near the inside wall of this or a similar reactor vessel in the core region. The operating P/T limit curves will be adjusted, as necessary, based on the evaluation findings and the recommendations of Reference 3.

Prior to reaching thirty-one effective full power years of operation, Figures 3.4.3-1, 3.4.3-2, and 3.4.3-3 must be updated for the next service period in accordance with 10 CFR 50, Appendix G. The service period must be of sufficient duration to permit the scheduled evaluation of a portion of the surveillance data scheduled in accordance with the latest revision of Topical Report BAW-1543 (Ref. 6). The highest predicted adjusted reference temperature of all the beltline region materials is used to determine the adjusted reference temperature at the end of the service period. The basis for this prediction is submitted for NRC staff review at least 90 days prior to the end of the service period.

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

The heatup curve represents a different set of restrictions than the cooldown curve because the directions of the thermal gradients through the vessel wall are reversed. The thermal gradient reversal alters the location of the tensile stress between the outer and inner walls.

The calculation to generate the inservice hydrostatic testing curve uses different safety factors (per Ref. 3) than the heatup and cooldown curves. The testing curve also extends to the RCS design pressure of 2500 psia.

The P/T limit curves and associated temperature rate of change limits are developed in conjunction with stress analyses for large numbers of operating cycles and provide conservative margins to nonductile failure. Although created to provide

limits for these specific normal operations, the curves also can be used to determine if an evaluation is necessary for an abnormal transient.

The consequence of violating the LCO limits is that the RCS has been operated under conditions that can result in brittle failure of the RCPB, possibly leading to a nonisolable leak or loss of coolant accident. In the event these limits are exceeded, an evaluation must be performed to determine the effect on the structural integrity of the RCPB components. The ASME Code, Section XI, Appendix E (Ref. 10) 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, an unanalyzed condition. Reference 1 establishes the methodology for determining the P/T limits.

RCS P/T limits satisfy Criterion 2 of 10 CFR 50.36 (Ref. 11).

LCO

The three elements of this LCO are:

a. The limit curves for heatup, cooldown, normal operation, PHYSICS TESTING and inservice hydrostatic testing;

b. Limits on the rate of change of temperature; and

c. Limits on RCP combinations.

The LCO limits apply to all components of the RCS, except the pressurizer (as indicated by the Note). These limits define allowable operating regions and permit a large number of operating cycles while providing a wide margin to nonductile failure.

The limits for the rate of change of temperature control the thermal gradient through the vessel wall and are used as inputs for calculating the 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.

The limit curves include the limiting pressure differential between the point of system pressure measurement and the pressure on the reactor vessel region controlling the limit curve. However, the limit curves are not adjusted for possible instrument error and should not be used for operation.

The heatup and cooldown rates stated are intended as the maximum changes in temperature in one direction in the stated time periods. The actual temperature linear ramp rate may exceed the stated limits for a shorter time period provided that the maximum total temperature difference does not exceed the limit and that a temperature hold is observed to prevent the total temperature difference from exceeding the limit for the stated time period.

The acceptable P/T combinations are below and to the right of the limit curves which are applicable for the first 31 EFPY. The limit curves include the limiting pressure differential between the point of system pressure measurement and the pressure on the reactor vessel region controlling the limit curve. However, the limit curves are not adjusted for possible instrument error and should not be used for operation.

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

a. The magnitude of the departure from the allowable operating P/T regime or the magnitude of the rate of change of temperature;

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

c. The existences, sizes, and orientations of flaws in the vessel material.

APPLICABILITY

The RCS P/T limits Specification provides a definition of acceptable operation for prevention of nonductile failure in accordance with 10 CFR 50, Appendix G (Ref. 2). Although the P/T limits were developed to provide guidance for operation during heatup or cooldown (MODES 3, 4, and 5) or inservice hydrostatic testing, their applicability is at all times in keeping with the concern for nonductile failure. The limits do not apply to the pressurizer.

ACTIONS

A.1 and A.2

With RCS pressure and temperature not within criticality limit of Figure 3.4.3-1 during PHYSICS TESTS with RCS temperature ≤ 525°F, 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 in 30 minutes. Rapid reactor shutdown can be readily and practically achieved in a 30 minute period. The Completion Time reflects the ability

to perform this Action and maintain the plant within the analyzed range. If RCS pressure and temperature can be restored within the 30 minute time period, shutdown is not required.

B.1 and B.2

Operation outside the P/T limits during MODE 1, 2, 3, or 4 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 to within limits, an evaluation is required to determine if RCS operation can continue. The evaluation must verify the RCPB integrity remains acceptable and must be completed before continuing operation beyond the 72 hour Completion Time of Required Action B.2. Several methods may be used, including comparison with pre-analyzed transients in the stress analyses, new analyses, or inspection of the components. The evaluation must be completed, documented, and approved in accordance with established unit procedures and administrative controls.

ASME Code, Section XI, Appendix E (Ref. 10) may be used to support the evaluation. However, its use is restricted to evaluation of the vessel beltline. The evaluation must extend to all components of the RCPB.

The 72 hour Completion Time is reasonable to accomplish the evaluation. The evaluation for a mild violation is possible within this time, but more severe violations may require special, event specific stress analyses or inspections. A favorable evaluation must be completed before continuing to operate beyond the 72 hour Completion Time.

Condition B is modified by a Note requiring Required Action B.2 to 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 B.1 is insufficient because higher than analyzed stresses may have occurred and may have affected the RCPB integrity.

C.1 and C.2

If a Required Action and associated Completion Time of Condition B are not met, the unit must be brought to a lower MODE because: (a) the RCS remained in an unacceptable pressure and temperature region for an extended period of increased stress, or (b) a sufficiently severe event caused entry into an unacceptable region. Either possibility indicates a need for more careful examination of the event. Performing this examination in the required MODES reduces the RCS at reduced pressure and temperature, which decreases the possibility of propagation of undetected flaws.

If the required restoration activity cannot be accomplished within 30 minutes, Required Action C.1 and Required Action C.2 must be initiated to reduce pressure and temperature.

If the required evaluation for continued operation cannot be accomplished within 72 hours, or the results are indeterminate or unfavorable, action must proceed to reduce pressure and temperature as specified in Required Actions C.1 and C.2. A favorable evaluation must be completed and documented before returning to operating pressure and temperature conditions. However, if the favorable evaluation is accomplished while reducing pressure and temperature conditions, a return to power operation may be initiated without completing these Required Actions.

Pressure and temperature are reduced by bringing the unit to MODE 3 within 6 hours and to MODE 5 within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required MODE from full power conditions in an orderly manner and without challenging unit systems.

D.1 and D.2

Actions must be initiated immediately to correct operation outside of the P/T limits at times other than MODE 1, 2, 3, or 4, so that the RCPB is returned to a condition that has been verified acceptable by stress analysis.

The immediate Completion Time reflects the urgency of initiating action to restore the parameters to within the analyzed range. Most violations will not be severe, and the activity can be accomplished promptly in a controlled manner.

In addition to restoring operation to within limits, an evaluation is required to verify that the RCPB integrity remains acceptable. The evaluation must be completed prior to entry into MODE 4. Several methods may be used, including comparison with pre-analyzed transients in the stress analysis, or inspection of the components.

ASME Code, Section XI, Appendix E (Ref. 10), may also be used to support the evaluation. However, its use is restricted to evaluation of the vessel beltline.

Condition D is modified by a Note requiring Required Action D.2 to 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 D.1, is insufficient because higher than analyzed stresses may have occurred and may have affected RCPB integrity.

SURVEILLANCE REQUIREMENTS

SR 3.4.3.1, SR 3.4.3.2, SR 3.4.3.3, and SR 3.4.3.4

Verification that operation is within the limits of the appropriate figure is required every 30 minutes when RCS pressure and temperature conditions are undergoing planned changes.

This Frequency is considered reasonable in view of the control room indication available to monitor RCS status. Also, since temperature rate of change limits are specified in hourly increments, 30 minutes permits assessment and correction for minor deviations within a reasonable time.

Surveillance for heatup, cooldown, or inservice hydrostatic testing may be discontinued when the definition given in the relevant unit procedure for ending the activity is satisfied.

The acceptable P/T combinations are below and to the right of the limit curves which are applicable for the first 31 EFPYs. The limit curves include the limiting pressure differential between the point of system pressure measurement and the pressure on the reactor vessel region controlling the limit curve. However, the limit curves are not adjusted for possible instrument error and should not be used for operation (as identified in Note 1 on each applicable Figure).

SR 3.4.3.1 is modified by a Note that requires this SR to be performed only during system heatup operations with fuel in the reactor vessel. This SR refers to Figure 3.4.3-1 which provides applicable heatup limitations, including reactor coolant pump (RCP) operating restrictions and allowable heatup rates. Figure 3.4.3-1 Note 2 identifies that when the decay heat removal system is operating with no RCPs operating, the indicated DHR system return temperature to the reactor vessel is the appropriate temperature indicator.

SR 3.4.3.2 is modified by a Note that requires this SR to be performed only during system cooldown operations with fuel in the reactor vessel. This SR refers to Figure 3.4.3-2 which provides applicable cooldown limitations, including reactor coolant pump (RCP) operating restrictions and allowable cooldown rates. During system cooldown operations with fuel in the reactor vessel, the RCPs are eventually removed from service. Figure 3.4.3-2 Note 2 identifies that when the decay heat removal system is operating with no RCPs operating, the indicated decay heat removal system return temperature to the reactor vessel is the appropriate temperature indicator. Figure 3.4.3-2 Note 2 also indicates that a maximum step temperature change of 25°F is allowable when removing all RCPs from operation with the decay heat removal system operating. The step temperature change is defined as the reactor coolant temperature (prior to stopping all RCPs) minus the decay heat removal system return temperature to the reactor vessel (after stopping all RCPs). The step change of 25°F is applicable only during transition from RCP operation to DHR. This step change must be included when determining the cooldown rate.

SR 3.4.3.3 is modified by a Note that requires this SR to be performed only during system heatup and cooldown operations with no fuel in the reactor vessel. This SR refers to Figure 3.4.3-2 which provides applicable heatup and cooldown limitations, including reactor coolant pump (RCP) operating restrictions and allowable heatup and cooldown rates. These curves are used during inservice hydrostatic testing that is performed in a defueled condition. The Notes on Figure 3.4.3-1 and Figure 3.4.3-2 are applicable to heatups and cooldowns performed within these limits.

SR 3.4.3.4 is modified by a Note that requires this SR to be performed only during PHYSICS TESTS with the average RCS temperature ≤ 525°F. This SR refers to Figure 3.4.3-1 which provides applicable limitations under which the unit may be critical, including Reactor Coolant Pump (RCP) operating restrictions and allowable heatup rates. This curve is used during PHYSICS TESTING. This is because LCO 3.4.2, "RCS Minimum Temperature for Criticality," normally limits the temperature for criticality to well above this curve. However, an exception to LCO 3.4.2 is provided by LCO 3.1.9, "PHYSICS TEST Exceptions - MODE 2," during PHYSICS TESTS initiated in MODE 2.

When the decay heat removal (DHR) system is operating with no RCPs operating, the indicated DHR system return temperature to the reactor vessel is the appropriate temperature indicator.

REFERENCES

1. BAW-10046A, "Methods of Compliance with Fracture Toughness and Operational Requirements of 10CFR50, Appendix G", Rev. 2, June 1986.
2. 10 CFR 50, Appendix G.
3. ASME, Boiler and Pressure Vessel Code, Section III, Appendix G.
4. Regulatory Guide 1.99, Revision 2, May 1988.
5. FTI Document 77-1258569-01.
6. BAW-1543, Integrated Reactor Vessel Material Surveillance Program (latest revision).
7. BAW-2121P, Irradiation Induced Reduction in Charpy Upper Shelf Energy of Reactor Vessel Welds.
8. FTI Calculations 32-1245917-00 and 32-1257716-00.
9. 10 CFR 50, Appendix H.
10. ASME, Boiler and Pressure Vessel Code, Section XI, Appendix E.
11. 10 CFR 50.36.

B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.4 RCS Loops - MODES 1 and 2

BASES

BACKGROUND

The primary function of the reactor coolant is removal of the heat generated in the fuel due to the fission process, and transfer of this heat, via the steam generators (SGs), to the secondary plant.

The secondary functions of the reactor coolant include:

a. Moderating the neutron energy level to the thermal state, to increase the probability of fission;

b. Improving the neutron economy by acting as a reflector;

c. Carrying the soluble neutron poison, boric acid;

d. Providing a second barrier against fission product release to the environment; and

e. Removing the heat generated in the fuel due to fission product decay following a unit shutdown.

The RCS configuration for heat transport uses two RCS loops. Each RCS loop contains an SG and two reactor coolant pumps (RCPs). An RCP is located in each of the two SG cold legs. The pump flow rate has been sized to provide core heat removal with appropriate margin to departure from nucleate boiling (DNB) during power operation and for anticipated transients originating from power operation. This Specification requires two RCS loops with either three or four pumps to be in operation. With only two or three pumps in operation the reactor power level is restricted to a nominal 49% RTP or 75% RTP, respectively, to preserve the core power to flow relationship, thus maintaining the margin to DNB. The intent of the Specification is to require core heat removal with forced flow during power operation. Specifying the minimum number of pumps is an effective technique for designating the proper forced flow rate for heat transport, and specifying two loops provides for the needed amount of heat removal capability for the allowed power levels. Specifying two RCS loops also provides the minimum necessary paths (two SGs) for heat removal.

The Reactor Protection System (RPS) Nuclear Overpower RCS Flow and Measured AXIAL POWER IMBALANCE trip setpoint is automatically reduced when a pump is taken out of service. Manual resetting is not necessary.

APPLICABLE SAFETY ANALYSES

Safety analyses (Ref. 1) contain various assumptions for the Design Bases Accident (DBA) initial conditions including: RCS pressure, RCS temperature, reactor power level, core parameters, and safety system setpoints. The important aspect for this LCO is the reactor coolant forced flow rate, which is represented by the number of pumps in service.

Both transient and steady state analyses have been performed to establish the effect of RCS flow on DNB. The initial condition DNB protection for the limiting loss of coolant flow event for four, three, and two pump operation is provided by the RCS flow surveillance criteria specified in the COLR for SR 3.4.1.3 and SR 3.4.1.4. The loss of coolant flow event which has been found to produce the limiting DNB is the four-to-two pump coastdown. In addition to the coastdown events, the single pump locked rotor event has been analyzed and shows that either the minimum DNB ratio is not less than the applicable critical heat flux correlation limit, or fuel cladding was shown to experience no significant temperature excursions.

Steady state DNB analysis has been performed for four, three, and two pump combinations. For four pump operation, the steady state DNB analysis, which generates the pressure and temperature SL (i.e., the departure from nucleate boiling ratio (DNBR) limit), assumes a maximum power level of 112% RTP. This is the design overpower condition for four pump operation. The 112% value is the accident analysis limit of the nuclear overpower (high flux) trip and is based on an analysis assumption that bounds possible instrumentation errors. The DNBR limit defines a locus of pressure and temperature points that result in a minimum DNBR that protects the critical heat flux correlation limit.

The three pump pressure temperature limit is tied to the steady state DNB analysis, which is evaluated each cycle. The flow used is the minimum allowed for three pump operation. The actual RCS flow rate will exceed the assumed flow rate. With three pumps operating, overpower protection is automatically provided by the RPS nuclear overpower RCS flow and measured AXIAL POWER IMBALANCE Function. The maximum power level for three pump operation is identified in the COLR and is based on the three pump flow as a fraction of the four pump flow at full power.

Although the Specification limits operation to a minimum of three pumps total, design evaluation (including analyses at steady state, ECCS initial conditions, and DNB conditions) also shows that operation with one pump in each loop (two pumps total) is acceptable when core THERMAL POWER is restricted to be proportionate to the flow. However, continued power operation with two RCPs removed from service is restricted to 24 hours (Ref. 2) since not all transient and accident conditions have been analyzed.

RCS Loops-MODES 1 and 2 satisfy Criterion 2 of 10 CFR 50.36 (Ref. 3).

LCO

The purpose of this LCO is to require adequate forced flow for core heat removal via two RCS loops. An operating loop consists of at least one operating RCP and a SG capable of heat removal. To meet safety analysis acceptance criteria for DNB, four pumps are required at rated power; if fewer pumps are available, power must be reduced as specified in the COLR.

APPLICABILITY

In MODES 1 and 2, the reactor may be critical and has the potential to produce maximum THERMAL POWER. To ensure that the assumptions of the accident analyses remain valid, all RCS loops are required to be OPERABLE and in operation in these MODES to prevent DNB and core damage.

The decay heat production rate is much lower than the full power heat rate. As such, the forced circulation flow and heat sink requirements are reduced for lower, noncritical MODES as indicated by the LCOs for MODES 3, 4, and 5.

Operation in other MODES is covered by:

- LCO 3.4.5, "RCS Loops-MODE 3";
- LCO 3.4.6, "RCS Loops-MODE 4";
- LCO 3.4.7, "RCS Loops-MODE 5, Loops Filled";
- LCO 3.4.8, "RCS Loops-MODE 5, Loops Not Filled";
- LCO 3.9.4, "Decay Heat Removal (DHR) and Coolant Circulation-High Water Level" (MODE 6); and
- LCO 3.9.5, "Decay Heat Removal (DHR) and Coolant Circulation-Low Water Level" (MODE 6).

ACTIONS

A.1

With one RCP not in operation in each loop, the assumptions of the safety analyses are not met, but design evaluation provided in Reference 2 concludes that events initiated during two pump operation would be expected to respond within the acceptance criteria for the ECCS. However, since no analysis was performed, Technical Specifications for two pump operation will only allow operation in MODES 1 or 2 for a period not to exceed 24 hours. The Completion Time of 18 hours provides sufficient time to restore operation of an additional RCP, while

allowing time to place the unit in MODE 3 within the 24 hour limitation if restoration of a third RCP is not accomplished.

B.1

If the Required Action and associated Completion Time of Condition A are not met, or if the LCO is not met for any reason other than provided in Condition A, the unit must be placed in a MODE in which the requirements are not applicable. This is accomplished by placing the unit in MODE 3. This reduces the core heat removal needs and minimizes the possibility of violating DNB limits. The Completion Time of 6 hours is reasonable, based on operating experience, to reach MODE 3 from power conditions in an orderly manner and without challenging safety systems.

SURVEILLANCE REQUIREMENTS

SR 3.4.4.1

This SR requires verification every 12 hours of the required number of loops in operation. Verification includes flow rate, temperature, or pump status monitoring, which helps ensure that forced flow is providing heat removal while maintaining the margin to DNB. The 12 hour interval has been shown by operating practice to be sufficient to regularly assess degradation and verify operation within safety analyses assumptions. In addition, control room indication and alarms will normally indicate loop status.

REFERENCES

1. SAR, Chapters 14 and 3A.
2. BAW-10103A, Revision 3, July 1977.
3. 10 CFR 50.36.

B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.5 RCS Loops - MODE 3

BASES

BACKGROUND

The primary function of the reactor coolant in MODE 3 is removal of decay heat and transfer of this heat, via the steam generators (SGs), to the secondary plant fluid. The secondary function of the reactor coolant is to act as a carrier for soluble neutron poison, boric acid.

In MODE 3, reactor coolant pumps (RCPs) are used to provide forced circulation for heat removal during heatup and cooldown. The number of RCPs in operation will vary depending on operational needs, and the intent of this LCO is to provide forced flow from at least one RCP for core heat removal and transport. The flow provided by one RCP is adequate for heat removal and for boron mixing. However, two RCS loops are required to be OPERABLE to provide redundant paths for heat removal.

Reactor coolant natural circulation is not normally used. If entry into natural circulation is required, the reactor coolant at the highest elevation of the hot leg must be maintained subcooled for single phase circulation. When in natural circulation, it is preferable to remove heat using both SGs to avoid idle loop stagnation that might occur if only one SG were in service. One generator will provide adequate heat removal. Boron reduction in natural circulation is prohibited because mixing to obtain a homogeneous concentration in all portions of the RCS cannot be ensured.

APPLICABLE SAFETY ANALYSES

No safety analyses are performed with initial conditions in MODE 3.

Failure to provide heat removal may result in challenges to a fission product barrier. The RCS loops are part of the primary success path that functions or actuates to prevent or mitigate a Design Basis Accident or transient that either assumes the failure of, or presents a challenge to, the integrity of a fission product barrier.

RCS Loops-MODE 3 satisfy Criterion 4 of 10 CFR 50.36 (Ref. 1).

LCO

The purpose of this LCO is to require two loops to be available for heat removal thus providing redundancy. The LCO requires the two loops to be OPERABLE with the intent of requiring both SGs to be capable of transferring heat from the reactor

coolant at a controlled rate. Forced reactor coolant flow is the preferred way to transport heat, although natural circulation flow is also acceptable under certain conditions. A minimum of one running RCP meets the LCO requirement for one loop in operation.

The Note permits a limited period of operation without RCPs. All RCPs may be removed from operation of ≤ 8 hours per 24 hour period for the transition to or from the Decay Heat Removal (DHR) System, and otherwise may be removed from operation for ≤ 1 hour per 8 hour period. During this condition, boron reduction with coolant at boron concentrations less than required to assure the SDM of LCO 3.1.1, is prohibited because an even concentration distribution throughout the RCS cannot be ensured. Core outlet temperature is to be maintained at least 10°F below the saturation temperature so that: a) no vapor bubble may form and possibly cause a natural circulation flow obstruction; and b) pump restart criteria (which vary with pressure) are met.

In MODES 3, 4, and 5, it is sometimes necessary to stop all RCP or DHR pump forced circulation (e.g., change operation from one DHR train to the other, to perform surveillance or startup testing, to perform the transition to and from DHR System cooling, or to avoid operation below the RCP minimum net positive suction head limit). This is acceptable because the reactor coolant temperature can be maintained subcooled and boron stratification affecting reactivity control is not expected.

An OPERABLE RCS loop consists of at least one OPERABLE RCP and an SG that is OPERABLE. To be considered OPERABLE, an RCP must be capable of being powered and able to provide forced flow if required. Similarly, an SG must be capable of transferring heat from the reactor coolant at a controlled rate and be in compliance with the Steam Generator Tube Surveillance Program.

APPLICABILITY

In MODE 3, the heat load is lower than at power; therefore, one RCS loop in operation is adequate for transport and heat removal. A second RCS loop is required to be OPERABLE but not in operation for redundant heat removal capability.

- Operation in other MODES is covered by:
- LCO 3.4.4, "RCS Loops-MODES 1 and 2";
- LCO 3.4.6, "RCS Loops-MODE 4";
- LCO 3.4.7, "RCS Loops-MODE 5, Loops Filled";
- LCO 3.4.8, "RCS Loops-MODE 5, Loops Not Filled";

- LCO 3.9.4, "Decay Heat Removal (DHR) and Coolant Circulation-High Water Level" (MODE 6); and
- LCO 3.9.5, "Decay Heat Removal (DHR) and Coolant Circulation-Low Water Level" (MODE 6).

ACTIONS

A.1

If one RCS loop is inoperable, redundancy for forced flow heat removal is lost. The Required Action is restoration of the RCS loop to OPERABLE status within a Completion Time of 72 hours. This time allowance is a justified period to be without the redundant nonoperating loop because a single loop in operation has a heat transfer capability greater than that needed to remove the decay heat produced in the reactor core.

B.1

If the Required Action and associated Completion Time of Condition A are not met, the unit must be brought to MODE 4. In MODE 4, the unit may be placed on the DHR System. The allowed Completion Time of 12 hours is reasonable, based on operating experience, to achieve cooldown and depressurization from the existing unit conditions and without challenging unit systems.

C.1 and C.2

If no RCS loop is OPERABLE or a required RCS loop is not in operation, (no RCS loop is required to be in operation provided the conditions of the Note in the LCO section are met), all operations involving introduction of coolant into the RCS with boron concentration less than required to meet the minimum SDM of LCO 3.1.1 must be immediately suspended. Action to restore one RCS loop to operation shall be immediately initiated and continued until one RCS loop is restored to operation and to OPERABLE status. Suspending the introduction of coolant into the RCS with boron concentration less than required to meet the minimum SDM of LCO 3.1.1 is required to assure continued safe operation. With coolant added without forced circulation, unmixed coolant could be introduced to the core, however coolant added with boron concentration meeting the minimum SDM maintains acceptable margin to subcritical operations. The immediate Completion Time reflects the importance of maintaining operation for decay heat removal.

SURVEILLANCE REQUIREMENTS

SR 3.4.5.1

This SR requires verification every 12 hours that the required loop (and pump) is in operation. Verification includes flow rate, temperature, or pump status monitoring, which help ensure that forced flow is providing heat removal. The 12 hour interval has been shown by operating practice to be sufficient to regularly assess RCS loop status. In addition, control room indication and alarms will normally indicate loop status.

SR 3.4.5.2

Verification that each required RCP is OPERABLE ensures that an RCS loop can be placed in operation, if needed, to maintain decay heat removal and reactor coolant circulation. Verification is performed by verifying proper breaker alignment and power availability to each required pump that is not in operation. Alternatively, verification that a pump is in operation also verifies proper breaker alignment and power availability. The Frequency of 7 days is considered reasonable in view of other administrative controls available and has been shown to be acceptable by operating experience.

This SR is modified by a Note that states the SR is not required to be performed until 24 hours after a required pump is not in operation.

REFERENCES

1. 10 CFR 50.36.

B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.6 RCS Loops - MODE 4

BASES

BACKGROUND

In MODE 4, the primary function of the reactor coolant is the removal of decay heat and transfer of this heat to the steam generators (SGs) or decay heat removal (DHR) heat exchangers. The secondary function of the reactor coolant is to act as a carrier for soluble neutron poison, boric acid.

In MODE 4, either reactor coolant pumps (RCPs) or DHR pumps can be used for coolant circulation. The number of pumps in operation can vary to suit the operational needs. The intent of this LCO is to provide forced flow from at least one RCP or one DHR pump for decay heat removal and transport. The flow provided by one RCP or one DHR pump is adequate for heat removal. The other intent of this LCO is to require that two paths (loops) be available to provide redundancy for heat removal.

APPLICABLE SAFETY ANALYSES

No safety analyses are performed with initial condition in MODE 4.

RCS Loops-MODE 4 satisfies Criterion 4 of 10 CFR 50.36 (Ref. 1).

LCO

The purpose of this LCO is to require that two loops, RCS or DHR, be OPERABLE in MODE 4 and one of these loops be in operation. The LCO allows the two loops that are required to be OPERABLE to consist of any combination of RCS or DHR System loops. Any one loop in operation provides enough flow to remove the decay heat from the core with forced circulation. The second loop that is required to be OPERABLE provides redundant paths for heat removal.

The Note permits a limited period of operation with the normally required RCP or DHR pump removed from operation. The Note prohibits boron dilution with coolant at boron concentrations less than required to assure the SDM of LCO 3.1.1 is maintained when forced flow is stopped because an even concentration distribution cannot be ensured. Core outlet temperature is to be maintained below saturation temperature by ≥ 10°F so that no vapor bubble may form and possibly cause a natural circulation flow obstruction.

When the DHR pumps are stopped, no alternate heat removal path exists, unless the RCS and SGs have been placed in service in forced or natural circulation. The

response of the RCS without heat removal through the DHR System or the SGs depends on the core decay heat load and the length of time that the DHR pumps are stopped. As decay heat diminishes, the effects on RCS temperature and pressure diminish. Without cooling by DHR, if the SGs are not capable of removing heat, higher heat loads will cause the reactor coolant temperature and pressure to increase at a rate proportional to the decay heat load. Because pressure can increase, the applicable system pressure limits (pressure and temperature (P/T) or low temperature overpressure protection (LTOP) limits) must be observed and forced DHR flow or heat removal via the SGs must be re-established prior to reaching the pressure limit.

An OPERABLE RCS loop consists of at least one OPERABLE RCP and an OPERABLE SG. To be considered OPERABLE, an SG must be capable of transferring heat from the reactor coolant at a controlled rate and be in compliance with the Steam Generator Tube Surveillance Program.

Similarly for the DHR System, an OPERABLE DHR loop is comprised of the OPERABLE DHR pump(s) capable of circulating RCS fluid through the DHR heat exchanger(s) and back to the RCS. To be considered OPERABLE, a DHR pump must be capable of being powered and able to provide flow if required, and a DHR heat exchanger must be capable of transferring heat from the reactor coolant at a controlled rate.

A DHR loop may be considered OPERABLE during alignment and when aligned for low pressure injection if it is capable of being manually (locally or remotely) realigned to the DHR mode of operation and is not otherwise inoperable. This provision arises because of the dual requirements of the components that comprise the low pressure injection/decay heat removal system.

APPLICABILITY

In MODE 4, this LCO applies because it is possible to remove core decay heat and to provide proper boron mixing with either the RCS loops and SGs or the DHR System.

Operation in other MODES is covered by:

- LCO 3.4.4, "RCS Loops-MODES 1 and 2";
- LCO 3.4.5, "RCS Loops-MODE 3";
- LCO 3.4.7, "RCS Loops-MODE 5, Loops Filled";
- LCO 3.4.8, "RCS Loops-MODE 5, Loops Not Filled";
- LCO 3.9.4, "Decay Heat Removal (DHR) and Coolant Circulation-High Water Level" (MODE 6); and

- LCO 3.9.5, "Decay Heat Removal (DHR) and Coolant Circulation-Low Water Level" (MODE 6).

ACTIONS

A.1

If only one required RCS loop or DHR loop is OPERABLE and in operation, redundancy for heat removal is lost. Action must be initiated to restore a second loop to OPERABLE status. The immediate Completion Time reflects the importance of maintaining the availability of two paths for heat removal.

A.2

If restoration is not accomplished and a DHR loop is OPERABLE, the unit must be brought to MODE 5 within the following 24 hours. Bringing the unit to MODE 5 is a conservative action with regard to decay heat removal. With only one DHR loop OPERABLE, redundancy for decay heat removal is lost and, in the event of a loss of the remaining DHR loop, it would be safer to initiate that loss from MODE 5 rather than MODE 4. The Completion Time of 24 hours is reasonable, based on operating experience, to reach MODE 5 in an orderly manner and without challenging unit systems.

This Required Action is modified by a Note which indicates that the unit must be placed in MODE 5 only if a DHR loop is OPERABLE. With no DHR loop OPERABLE, the unit is in a condition with only limited cooldown capabilities. Therefore, the actions are to be concentrated on restoration of a DHR loop, rather than a cooldown of extended duration.

B.1 and B.2

If no RCS or DHR loops are OPERABLE or a required loop is not in operation (no loop is required to be in operation provided the conditions of the Note in the LCO section are met), all operations involving introduction of coolant into the RCS with boron concentration less than required to meet the minimum SDM of LCO 3.1.1 must be suspended and action to restore one RCS or DHR loop to OPERABLE status and operation must be initiated. The required margin to criticality must not be reduced in this type of operation. Suspending the introduction of coolant into the RCS with boron concentration less than required to meet the minimum SDM of LCO 3.1.1 is required to assure continued safe operation. With coolant added without forced circulation, unmixed coolant could be introduced to the core, however coolant added with boron concentration meeting the minimum SDM maintains acceptable margin to subcritical operations. The immediate Completion Times reflect the importance of maintaining operation for decay heat removal. The action to restore must continue until one loop is restored to operation.

SURVEILLANCE REQUIREMENTS

SR 3.4.6.1

This Surveillance requires verification every 12 hours of the required DHR or RCS loop in operation to ensure forced flow is providing decay heat removal. Verification includes flow rate, temperature, or pump status monitoring. The 12 hour interval has been shown by operating practice to be sufficient to regularly assess RCS loop status. In addition, control room indication and alarms will normally indicate loop status.

SR 3.4.6.2

Verification that each required pump is OPERABLE ensures that an RCS or DHR loop can be placed in operation if needed to maintain decay heat removal and reactor coolant circulation. Verification is performed by verifying proper breaker alignment and power available to each required pump. Alternatively, verification that a pump is in operation also verifies proper breaker alignment and power availability. The Frequency of 7 days is considered reasonable in view of other administrative controls and has been shown to be acceptable by operating experience.

This SR is modified by a Note that states the SR is not required to be performed until 24 hours after a required pump is not in operation.

REFERENCES

1. 10 CFR 50.36.

B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.7 RCS Loops - MODE 5, Loops Filled

BASES

BACKGROUND

In MODE 5 with RCS loops filled, the primary function of the reactor coolant is the removal of decay heat and transfer of this heat either to the steam generator (SG) secondary side coolant or the service water via the decay heat removal (DHR) heat exchangers. While the principal means for decay heat removal is via the DHR System, the SGs are specified as a backup means for redundancy. Although the SGs do not typically remove heat unless steaming occurs, they are available as a temporary heat sink and can be used by allowing the RCS to heat up into the temperature region of MODE 4 where steaming can be effective for heat removal. The secondary function of the reactor coolant is to act as a carrier for soluble neutron poison, boric acid.

In MODE 5 with RCS loops filled, DHR loops are the principal means for heat removal. The number of loops in operation can vary to suit the operational needs. The intent of this LCO is to provide forced flow from at least one DHR loop for decay heat removal and transport. The flow provided by one DHR loop is adequate for decay heat removal. The other intent of this LCO is to require that a second path be available to provide a backup method for heat removal.

APPLICABLE SAFETY ANALYSES

No safety analyses are performed with initial conditions in MODE 5.

RCS Loops-MODE 5 (Loops Filled) satisfies Criterion 4 of 10 CFR 50.36 (Ref. 1).

LCO

The purpose of this LCO is to require that at least one of the DHR loops be OPERABLE and in operation with an additional DHR loop OPERABLE or both SGs with secondary side water level ≥ 20 inches. One DHR loop provides sufficient forced circulation to perform the safety functions of the reactor coolant under these conditions. The second DHR loop is normally maintained as a backup to the operating DHR loop to provide redundancy for decay heat removal. However, if the standby DHR loop is not OPERABLE, a sufficient alternate method of providing redundant heat removal paths is to provide both SGs with their secondary side water levels ≥ 20 inches. Should the operating DHR loop fail, the SGs could be used to remove the decay heat.

The LCO provides for either SG heat removal or DHR System heat removal. In this MODE, reactor coolant pump (RCP) operation may be restricted because of net positive suction head (NPSH) limitations, and the SG will not be able to provide steam for the turbine driven feed pumps. However, to ensure that the SG(s) can be used as a heat sink, a motor driven feedwater pump is needed, because it is independent of steam. Condensate pumps, the auxiliary feedwater pump, or the motor driven emergency feedwater pump can be used. If RCPs are available, the steam generator level need not be adjusted. If RCPs are not available, the water level must be adjusted for natural circulation. The high entry point in the generator should be accessible from the feedwater pumps so that natural circulation can be stimulated. The SGs are primarily a backup to the DHR pumps, which are used for forced flow. By requiring the SGs to be a backup heat removal path, the option to increase RCS pressure and temperature for heat removal in MODE 4 is provided.

Note 1 permits the DHR pumps to be stopped for up to 1 hour. The circumstances for stopping both DHR trains are to be limited to situations where: (a) Pressure and temperature increases can be maintained well within the allowable pressure (P/T and low temperature overpressure protection) and 10°F subcooling limits; and (b) no operations are in process that would cause reduction of the RCS boron concentration.

The Note prohibits boron dilution with coolant at boron concentrations less than required to assure the SDM of LCO 3.1.1 is maintained when DHR forced flow is stopped because an even concentration distribution cannot be ensured. Core outlet temperature is to be maintained below saturation temperature by ≥ 10°F so that no vapor bubble would form and possibly cause a natural circulation flow obstruction. In this MODE, the steam generators are used as a backup for decay heat removal and, to ensure their availability, the RCS loop flow path is to be maintained with subcooled liquid.

In MODE 5, it is sometimes necessary to stop all RCP or DHR pump forced circulation. For example, this may be necessary to change operation from one DHR train to the other, perform surveillance or startup testing, perform the transition to and from the DHR System, or to avoid operation below the RCP minimum NPSH limit. The time period is acceptable because the reactor coolant temperature can be maintained subcooled, and boron stratification affecting reactivity control is not expected.

Note 2 allows one required DHR loop to be inoperable for a period of ≤ 2 hours provided that the other loop is OPERABLE and in operation. This permits periodic surveillance tests to be performed on the inoperable loop during the only time when such testing is safe and possible.

Note 3 provides for an orderly transition from MODE 5 to MODE 4 during a planned heatup by permitting removal of DHR loops from operation when at least one RCP is in operation. This Note provides for the transition to MODE 4 where an RCP is permitted to be in operation and replaces the RCS circulation function provided by the DHR loops.

A DHR loop may be considered OPERABLE during alignment and when aligned for low pressure injection if it is capable of being manually (locally or remotely) realigned to the DHR mode of operation and is not otherwise inoperable. This provision arises because of the dual requirements of the components that comprise the low pressure injection/decay heat removal system.

An OPERABLE DHR loop is composed of an OPERABLE DHR pump and an OPERABLE DHR heat exchanger.

To be considered OPERABLE, DHR pumps must be capable of being powered and are able to provide flow if required. During performance of SR 3.8.1.7 or SR 3.8.1.8, the affected DHR pump may be considered OPERABLE even with the breaker "racked down" since placing this second pump in operation is a manual action. Similarly, an OPERABLE SG can perform as a heat sink when it has an adequate water level and is in compliance with the Steam Generator Tube Surveillance Program.

APPLICABILITY

In MODE 5 with loops filled, forced circulation is provided by this LCO to remove decay heat from the core and to provide proper boron mixing. One loop of DHR provides sufficient circulation for these purposes.

Operation in other MODES is covered by:

- LCO 3.4.4, "RCS Loops-MODES 1 and 2";
- LCO 3.4.5, "RCS Loops-MODE 3";
- LCO 3.4.6, "RCS Loops-MODE 4";
- LCO 3.4.8, "RCS Loops-MODE 5, Loops Not Filled";
- LCO 3.9.4, "Decay Heat Removal (DHR) and Coolant Circulation-High Water Level" (MODE 6); and
- LCO 3.9.5, "Decay Heat Removal (DHR) and Coolant Circulation-Low Water Level" (MODE 6).

ACTIONS

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

If one required DHR loop is inoperable and any required SG has secondary side water level < 20 inches, redundancy for heat removal is lost. Action must be initiated to restore a second DHR loop to OPERABLE status or initiate action to

restore the secondary side water level in the SG(s), and action must be taken immediately. Either Required Action will restore redundant decay heat removal paths. The immediate Completion Time reflects the importance of maintaining the availability of two paths for heat removal.

C.1 and C.2

If no required DHR loop is in operation, except as provided in Note 1, or no required DHR loop is OPERABLE, all operations involving introduction of coolant into the RCS with boron concentration less than required to meet the minimum SDM of LCO 3.1.1 must be suspended and action to restore a DHR loop to OPERABLE status and operation must be initiated. The margin to criticality must not be reduced in this type of operation. Suspending the introduction of coolant into the RCS with boron concentration less than required to meet the minimum SDM of LCO 3.1.1 is required to assure continued safe operation. With coolant added without forced circulation, unmixed coolant could be introduced to the core, however coolant added with boron concentration meeting the minimum SDM maintains acceptable margin to subcritical operations. The immediate Completion Time reflects the importance of maintaining operation for decay heat removal.

SURVEILLANCE REQUIREMENTS

SR 3.4.7.1

This SR requires verification every 12 hours that the required DHR loop is in operation. Verification includes flow rate, temperature, or pump status monitoring, which help ensure that forced flow is providing heat removal. The 12 hour Frequency has been shown by operating practice to be sufficient to regularly assess degradation. In addition, control room indication and alarms will normally indicate loop status.

SR 3.4.7.2

Verifying the SGs are OPERABLE by ensuring their secondary side water levels are $\geq$ 20 inches ensures that redundant heat removal paths are available if the second DHR loop is not OPERABLE. If both DHR loops are OPERABLE, this Surveillance is not needed. The 12 hour Frequency has been shown by operating practice to be sufficient to regularly assess RCS loop status.

SR 3.4.7.3

Verification that each required DHR pump is OPERABLE ensures that a DHR loop can be placed in operation if needed to maintain decay heat removal and reactor coolant circulation. If the secondary side water level is $\geq$ 20 inches in both SGs, this Surveillance is not needed. Verification is performed by verifying proper breaker

alignment and power available to each required pump. Alternatively, verification that a pump is in operation also verifies proper breaker alignment and power availability. The Frequency of 7 days is considered reasonable in view of other administrative controls available and has been shown to be acceptable by operating experience.

This SR is modified by a Note that states the SR is not required to be performed until 24 hours after a required pump is not in operation.

REFERENCES

1. 10 CFR 50.36.

B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.8 RCS Loops - MODE 5, Loops Not Filled

BASES

BACKGROUND

In MODE 5 with loops not filled, the primary function of the reactor coolant is the removal of decay heat and transfer of this heat to the decay heat removal (DHR) heat exchangers. The steam generators (SGs) are not available as a heat sink when the loops are not filled. The secondary function of the reactor coolant is to act as a carrier for the soluble neutron poison, boric acid.

Loops are considered not filled when the RCS draining is initiated (as might be the case for refueling or maintenance). Additionally, reductions of RCS inventory below el. 375 ft. are termed reduced inventory operations. GL 88-17 (Ref. 1) expresses concerns for loss of decay heat removal for this operating condition. With water at this low level, the margin above the decay heat suction piping connection to the hot leg is small. The possibility of loss of level or inlet vortexing exists and if it were to occur, the operating DHR pump could become air bound and fail resulting in a loss of forced flow for heat removal. As a consequence the water in the core will heat up and could boil with the possibility of core uncovering due to boil off.

In MODE 5 with loops not filled, only DHR pumps can be used for coolant circulation. The number of pumps in operation can vary to suit the operational needs. The intent of this LCO is to require forced flow from at least one DHR pump for decay heat removal and transport, and to require that two paths be available to provide redundancy for heat removal.

APPLICABLE SAFETY ANALYSES

No safety analyses are performed with initial conditions in MODE 5 with loops not filled.

RCS Loops-MODE 5 (Loops Not Filled) satisfies Criterion 4 of 10 CFR 50.36 (Ref. 2).

LCO

The purpose of this LCO is to require that a minimum of two DHR loops be OPERABLE and that one of these loops be in operation. An OPERABLE loop is one that has the capability of transferring heat from the reactor coolant at a controlled rate. Heat cannot be removed via the DHR system unless forced flow is used. A minimum of one running decay heat removal pump meets the LCO

requirement for one loop in operation. An additional DHR loop is required to be OPERABLE to provide redundancy for heat removal.

Note 1 permits the DHR pumps to be de-energized for ≤ 1 hour. The Note prohibits boron dilution with coolant at boron concentrations less than required to assure the SDM of LCO 3.1.1 is maintained or draining operations when DHR forced flow is stopped.

Note 2 allows one DHR loop to be inoperable for a period of ≤ 2 hours provided that the other loop is OPERABLE and in operation. This permits periodic surveillance tests to be performed on the inoperable loop during MODE 5 when these tests are safe and possible.

A DHR loop may be considered OPERABLE during alignment and when aligned for low pressure injection if it is capable of being manually (locally or remotely) realigned to the DHR mode of operation and is not otherwise inoperable. This provision arises because of the dual requirements of the components that comprise the low pressure injection/decay heat removal system.

An OPERABLE DHR loop is composed of an OPERABLE DHR pump capable of circulating RCS fluid through an OPERABLE DHR heat exchanger and back to the RCS. To be considered OPERABLE, the DHR pumps must be capable of being powered and able to provide flow if required. During performance of SR 3.8.1.7 or SR 3.8.1.8, the affected DHR pump may be considered OPERABLE even with the breaker "racked down" since placing this second pump in operation is a manual action.

APPLICABILITY

In MODE 5 with loops not filled, this LCO requires core heat removal and coolant circulation by the DHR System.

Operation in other MODES is covered by:

- LCO 3.4.4, "RCS Loops-MODES 1 and 2";
- LCO 3.4.5, "RCS Loops-MODE 3";
- LCO 3.4.6, "RCS Loops-MODE 4";
- LCO 3.4.7, "RCS Loops-MODE 5, Loops Filled";
- LCO 3.9.4, "Decay Heat Removal (DHR) and Coolant Circulation-High Water Level" (MODE 6); and
- LCO 3.9.5, "Decay Heat Removal (DHR) and Coolant Circulation-Low Water Level" (MODE 6).

ACTIONS

A.1

If one required DHR loop is inoperable, redundancy for heat removal is lost. Required Action A.1 is to immediately initiate activities to restore a second loop to OPERABLE status. The immediate Completion Time reflects the importance of maintaining the availability of two paths for heat removal.

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

If no required loop is OPERABLE or the required loop is not in operation, except as provided by Note 1 in the LCO, the Required Action requires immediate suspension of all operations involving introduction of coolant into the RCS with boron concentration less than required to meet the minimum SDM of LCO 3.1.1 or reduction of RCS water inventory and requires initiation of action to immediately restore one DHR loop to OPERABLE status and operation. The Required Action for restoration does not apply to the condition of both loops not in operation when the exception Note in the LCO is in force. Suspending the introduction of coolant into the RCS with boron concentration less than required to meet the minimum SDM of LCO 3.1.1 is required to assure continued safe operation. With coolant added without forced circulation, unmixed coolant could be introduced to the core, however coolant added with boron concentration meeting the minimum SDM maintains acceptable margin to subcritical operations. The immediate Completion Time reflects the importance of maintaining operations for decay heat removal. The action to restore must continue until one loop is restored.

SURVEILLANCE REQUIREMENTS

SR 3.4.8.1

This Surveillance requires verification every 12 hours that at least one loop is in operation. Verification includes flow rate, temperature, or pump status monitoring, which help ensure that forced flow is providing heat removal. The 12 hour interval has been shown by operating practice to be sufficient to regularly assess RCS loop status.

SR 3.4.8.2

Verification that each required pump is OPERABLE ensures that redundancy for heat removal is provided. The requirement also ensures that a DHR loop can be placed in operation if needed to maintain decay heat removal and reactor coolant circulation. Verification is performed by verifying proper breaker alignment and power available to each required pump. Alternatively, verification that a pump is in operation also verifies proper breaker alignment and power availability. The

Frequency of 7 days is considered reasonable in view of other administrative controls available and has been shown to be acceptable by operating experience.

This SR is modified by a Note that states the SR is not required to be performed until 24 hours after a required pump is not in operation.

REFERENCES

1. Generic Letter 88-17, October 17, 1988.

1. 10 CFR 50.36.

CTS DISCUSSION OF CHANGES

ITS Section 3.4A: Reactor Coolant System

Note: The ITS Section 3.4A package includes the following ITS:

ITS 3.4.1	RCS Pressure, Temperature and Flow DNB Limits
ITS 3.4.2	RCS Minimum Temperature for Criticality
ITS 3.4.3	RCS P/T Limits
ITS 3.4.4	RCS Loops - MODE 1 and 2
ITS 3.4.5	RCS Loops - MODE 3
ITS 3.4.6	RCS Loops - MODE 4
ITS 3.4.7	RCS Loops - MODE 5, Loops Filled
ITS 3.4.8	RCS Loops - MODE 5, Loops Not Filled

which address the corresponding NUREG-1430 RSTS.

ADMINISTRATIVE

A1 The designated change represents a non-technical, non-intent change to the Arkansas Nuclear One, Unit 1 Current Technical Specifications (CTS) made to make the ANO-1 Improved Technical Specifications (ITS) consistent with the Babcock and Wilcox (B&W) revised Standard Technical Specification (RSTS), NUREG-1430, Revision 1. This change does not alter the requirements of the CTS or RSTS. Examples of this type of change include: wording preference; convention adoption; editorial, numbering and formatting changes; and hierarchy structure.

A2 The ANO-1 CTS Bases will be administratively deleted in their entirety in favor of the NUREG-1430 Bases. The CTS Bases will be reviewed for technical content that will be identified for retention in the ITS Bases.

A3 The CTS 3.1.1.1.B requirements for coolant circulation when boron concentration is being reduced are presumed to be "at all times" since no applicable conditions are identified. These requirements are fulfilled in ITS LCO 3.4.4 for MODES 1 and 2, LCO 3.4.5 for MODE 3, LCO 3.4.6 for MODES 4 and 5 , and LCO 3.9.1, LCO 3.9.4 and LCO 3.9.5 for MODE 6. However, the Actions identified in CTS 3.1.1.1.B are not considered to be applicable in MODES 1 and 2 (i.e., for LCO 3.4.4) since complete loss of flow will result in a reactor trip and placing the unit in MODE 3. The Actions for MODE 6 are addressed in the ITS Section 3.9 Discussions of Change.

A4 The CTS 3.1.1.5.A requirements for OPERABILITY of RCS loops is identified as applicable "with the reactor coolant average temperature above 280 F." These requirements are fulfilled in ITS LCO 3.4.4 and LCO 3.4.5. However, the Actions identified in CTS 3.1.1.5 are not considered to be applicable in MODES 1 and 2 (i.e., for LCO 3.4.4) since complete loss of flow in one loop will result in a reactor trip and placing the unit in MODE 3.

A5 The CTS 3.1.1.6.A requirement to be in COLD SHUTDOWN in 20 hours is not reflected in ITS 3.4.7 or ITS 3.4.8 since the unit is already in MODE 5.

A6 The CTS 3.1.2.1 statement that "The provisions of Specifications 3.0.3 are not applicable" is not required to be reflected in ITS LCO 3.4.3 since the ACTIONS provided address all possible conditions in MODES 1, 2, 3, and 4, and ITS LCO 3.0.3 is only applicable in these MODES.

A7 The CTS 3.1.2.6 requirement to place the unit in cold shutdown "while maintaining RCS temperature and pressure below the curve" is identified in ITS only as "be in MODE 5." The specifics of meeting the requirements while shutting down are not reflected since these are included in the LCO and are always understood to be required. If the requirements of the LCO can be met, they are required, and if they cannot be met (i.e., compliance is not restored as required by Required Action A.1), the shutdown to MODE 5 is still required. Therefore, this is considered an administrative change due only to application and format consistent with NUREG-1430.

A8 The "above 525°F" requirement for a minimum condition for criticality in CTS 3.1.3.1 has been revised to ≥ 525°F in ITS 3.4.2. These are considered to be essentially equivalent since the parameter can be less than the limit, but be so close as to be imperceptible. This change is consistent with design basis and with NUREG-1430.

A9 The "restore... to within the limit" requirement of CTS 3.1.3.7 is not retained in ITS. Since restoration of compliance is always an option, it is not necessary to specifically identify this action. This is considered an administrative change due only to application and format consistent with NUREG-1430.

A10 Not used.

A11 CTS 3.1.2.2 requires compliance with requirements which are already in effect and otherwise applicable. . Such duplication is unnecessary and results in additional administrative burden to revise the duplicate TS when these regulations are revised. Since removal of the duplication results in no actual change in the requirements, removal of the duplicative information is considered an administrative change. This change is consistent with NUREG-1430.

A12 CTS 3.1.1.6 provides requirements for "with the reactor coolant average temperature at or below 280°F, but the reactor above the refueling shutdown condition." In ITS, these operating conditions are presented as MODES 4 and 5, and are split into three Specifications for MODE 4, MODE 5 with the loops filled, and MODE 5 with the loops not filled. This change is consistent with NUREG-1430.

A13 Not used

A14 The allowance of CTS 3.1.1.6 Note * to "de-energize" the reactor coolant pump(s) and decay heat removal pump(s) is revised to allow the pumps to be "removed from operation." This allowance more closely matches the requirement for the pump(s) to be "in operation" and is consistent with the wording of a similar Note in NUREG-1430 LCO 3.9.4. Since there is no change in intent or application, this change is considered administrative.

A15 An Applicability of "at all times" is included in ITS 3.4.3. CTS 3.1.2 provides similar requirements but does not clearly specify the Applicability except as during heatup, during cooldown, or during hydro tests. Since the ITS SR Notes provide the same limitations for each of the various limits, this addition of the Applicability is considered an administrative change to accommodate format.

A16 The CTS 3.1.2.3 and 3.1.3.2 limitation for the RCS temperature to be to the right of the criticality curve is revised to be applicable only during the physics testing allowed under CTS 3.1.3.1 (ITS 3.1.9). If not performing physics testing, the minimum temperature for criticality (525°F as required by ITS LCO 3.4.2) is well above the required temperatures on the pressure/temperature limits curve. Therefore, if above the normal RCS temperature limits of 530°F for performing a frequent (i.e., every 30 minutes) Surveillance of RCS temperature, there is also no need to require the performance of a Surveillance with lower limits.

A17 CTS 3.1.1.1.A does not provide required actions for noncompliance. Therefore, the appropriate actions were provided by CTS 3.0.3 which would require that the unit be placed in a mode for which the requirement does not apply. This is the same action as will be required by ITS Required Action B.1. Therefore, this change is considered to be administrative in nature.

A18 This page is not yet approved in its current form. Therefore, this markup is dependent on the expected NRC approval of the January 27, 2000, (0CAN010004) LAR related to the Q Condensate Storage Tank volume.

CTS DISCUSSION OF CHANGES

TECHNICAL CHANGE -- MORE RESTRICTIVE

M1 The CTS 3.1.1.1.A and Table 2.3-1, Note (d), limitation of 24 hours with only two operating reactor coolant pumps is converted to a Required Action with an explicit time frame to restore a third operating pump. Also, a default Required Action is included to clarify the specific action required if Condition A is not met. The proposed Completion Time for restoration of a third pump (Required Action A.1) and exiting the applicable conditions (Required Action B.1) provide for appropriate and prompt compensatory actions, while allowing sufficient time to accomplish the activities required in an orderly manner and without challenging safety systems. Further, the combined Completion Times (18 hrs + 6 hrs) are consistent with CTS allowance for continued critical operation limited to 24 hours. However, the additional detail and intermediate requirements are an additional restriction on unit operation.

Additionally, the CTS applicable conditions of "when the reactor is critical" are revised to include ITS MODES 1 and 2. These Applicability's are essentially the same except that ITS MODE 2 also includes a condition of $k_{eff} < 1.0$ but ≥ 0.99. This addition results in no practical change since the conditions are not readily differentiated in the control room. This is considered to be a minor additional restriction on unit operation consistent with NUREG-1430.

M2 CTS 3.1.2.2 provides a cross reference to identify that when the leak tests required by CTS 4.3 are conducted, they must be conducted under the provision of CTS 3.1.2.3, and identifies that the provisions of CTS 3.0.3 are not applicable. In the ITS, this exception to LCO 3.0.3 is not retained since it is not expected to be needed and would probably be moot for most situations that would cause failure of the leak test. Regardless, the allowance is removed, and is considered to be a minor additional restriction on unit operation consistent with NUREG-1430.

M3 Appropriate Surveillance Requirements are included with ITS LCO 3.4.4 and LCO 3.4.5. These SRs require verification that the required RCS loops are in operation in MODE 1 and 2 (SR 3.4.4.1) and verification that the required RCS loop is in operation in MODE 3 (SR 3.4.5.1). These SRs are an additional restriction on unit operation consistent with NUREG-1430.

3.4A-12 M4 The CTS 3.1.1.6 requirements allow for any two of the four identified heat removal loops to be used in MODES 4 and 5. ITS 3.4.7 will require that both steam generators be OPERABLE if only one DHR system is OPERABLE. Requiring both SGs to be OPERABLE when only one DHR system is OPERABLE is an additional restriction on unit operation consistent with NUREG-1430.

M5 A specific Completion Time is provided for completing the evaluation of the impact of the out-of-limit condition on the fracture toughness properties of the RCS and determining that the RCS remains acceptable for continued operation. CTS 3.1.2.6 contains no such Completion Time but requires only that the evaluation be done. The proposed Completion Time of 72 hours is considered reasonable for operation in MODES 1, 2, 3, and 4, because the limits represent controls on long term vessel

fatigue and usage factors, and short periods (i.e., ≤ 30 minutes) of noncompliance with the limits are not expected to present an immediate threat to the RCS integrity. In other conditions (i.e., MODES 5 and 6, and defueled), the proposed Required Action and associated Completion Time would prevent entry into MODE 4 which is consistent with CTS LCO 3.0.4. Additionally, Notes are provided in proposed Conditions A and C to require the evaluation to be completed even if compliance with the limits is restored. Therefore, this change is an additional restriction on unit operation consistent with NUREG-1430.

M6 The CTS 3.1.2.6 and CTS 3.1.6.7 requirements that the unit be placed in HOT STANDBY within the next 6 hours (if the evaluation does not determine the RCS to be acceptable) is revised to require the unit to be placed in ITS MODE 3. Since the CTS HOT STANDBY requires the unit to be ≤ 2% RTP and ITS MODE 3 is a subcritical condition, this change is an additional restriction on unit operation. The activity to reduce the unit by an additional 2% RTP is a minimal change in operation which provides consistency within the ITS for shutdown applications. The change is of little consequence since the unit evaluation will generally require a significant effort prior to restart and the unit must be placed in COLD SHUTDOWN (ITS MODE 5) within an additional 30 hours. This change is consistent with NUREG-1430.

M7 Specific Surveillance Requirements (SR 3.4.3.1, SR 3.4.3.2, SR 3.4.3.3, and SR 3.4.3.4) are provided for verifying the RCS pressure and temperature limits during heatup and cooldown. These requirements provide a specific Frequency which is not included in CTS 3.1.2. This change is an additional restriction on unit operation consistent with NUREG-1430.

M8 CTS 3.1.3.7 is revised to treat the pressure and temperature limits for criticality just as any other pressure and temperature limit in ITS 3.4.3. The revisions include additional Required Actions to perform the evaluation of the RCS to determine that it is acceptable for continued operation and to place the unit in MODE 5 if the evaluation is not acceptable. This change is an additional restriction on unit operation consistent with NUREG-1430.

M9 CTS 3.4.1 requires two steam generators be capable of removing heat for operation above 280°F. CTS 3.4.2 provides the Actions if Specification 3.4.1 is not met and actually allows the steam generators to be removed from service for up to 24 hours before requiring the unit to be in hot shutdown within the next 12 hours. ITS does not allow operation in MODES 1, 2, and 3 without both steam generators OPERABLE. ITS LCO 3.4.4, Condition B, will require the unit, if in MODES 1 or 2, to be in MODE 3 within 6 hours. This is necessary since such operation of the unit would be significantly outside the initial conditions of the safety analysis. This is an additional restriction on unit operation. (See also DOC L6.)

M10 The CTS does not include Reactor Coolant System (RCS) pressure, temperature, or flow departure from nucleate boiling (DNB) limits. The RSTS LCO 3.4.1 requirements for DNB limits are being incorporated into the unit specific ITS. These limits on RCS pressure, temperature, and flow rate are provided "to ensure that the

core operates within the limits assumed for the plant safety analyses." Operating within these limits will result in meeting departure from nucleate boiling ratio (DNBR) criteria in the event of a DNB limited transient. Similar criteria are used to determine the Reactor Protection System (RPS) trip setpoints based on pressure, temperature and flow; however, the RPS trip setpoints are designed to assure the unit does not exceed a safety limit, rather than DNBR criteria. These limits are an additional restriction on the operation of the unit based on NUREG-1430.

M11 The Applicability for ITS 3.4.2 is taken from CTS 3.1.3, Minimum Conditions for Criticality. However, the Applicability is given as including all of MODES 1 and 2, rather than MODE 1 and MODE 2 with $k_{eff} \geq 1.0$. This is consistent with the action requirements of CTS 3.1.3.7 which require the unit to be placed in Hot Shutdown (MODE 3), with past practice, and with the unit control rod ejection analysis which is performed for full power and zero power conditions, and evaluated to bound the event should it occur in MODE 2 with $k_{eff} < 1.0$ (see SAR Section 14.2.2.4.1.1).

M12 An additional restriction is added to the allowance for de-energizing the DHR loops during MODE 5 with the loops not filled, as provided by CTS 3.1.1.6. This additional restriction precludes draining operations to further reduce the RCS water volume with no forced flow from a DHR pump, and significantly reduces the probability of a loss of decay heat removal event. Since this not a CTS restriction for pump de-energization, this is an additional restriction on unit operation consistent with NUREG-1430.

M13 New Surveillance Requirements (ITS SR 3.4.7.3 and SR 3.4.8.2) are added to periodically verify the additional loop is ready to be placed in operation if required. This change is an additional restriction on unit operation consistent with NUREG-1430.

M14 A new Surveillance Requirement (SR 3.4.2.1) is included to periodically verify compliance with the requirements of CTS 3.1.3 (ITS 3.4.2). This SR provides frequent verification of compliance during operation. This change is an additional restriction on unit operation consistent with NUREG-1430 as modified by Generic Traveler TSTF-027, Rev. 1.

3.4A-09

3.4A-10

M15 Not used.

M16 An additional restriction (ITS 3.4.8, Required Action B.2) is incorporated to "suspend all operations involving reduction in RCS water volume" with both DHR loops inoperable or both required DHR pumps are not in operation when they are required to be. This is consistent with the requirements for no reduction in water volume while intentionally removing both DHR pumps from operation as allowed by ITS 3.4.8, Note 1, part b. This change adds a requirement which is not included in either the CTS or NUREG-1430.

CTS DISCUSSION OF CHANGES

TECHNICAL CHANGE -- LESS RESTRICTIVE

L1 The CTS 3.1.1.6 actions for an inoperable coolant loop in MODES 4 and 5 have been revised to allow an additional 4 hours before requiring the unit to be in COLD SHUTDOWN (MODE 5) if a decay heat removal system loop is OPERABLE. The 24 hours is reasonable based on operating experience to reach MODE 5 in an orderly manner and without challenging unit systems. The actions are also revised to omit the requirement to be in MODE 5 in 20 hours if the only OPERABLE coolant loop is an RCS loop. A single RCS loop may not be able to remove sufficient heat to reduce the RCS temperature to MODE 5 conditions, or at best will require an extended duration to reach MODE 5. Therefore, the actions are concentrated on restoration of a DHR loop, rather than attempting to cooldown to MODE 5. These proposed Required Actions are consistent with NUREG-1430.

L2 The CTS 3.1.1.6 requirements allow for OPERABLE RCS loops to provide the required cooling during operation at or below 280°F but above the refueling shutdown condition (i.e., ITS MODES 4 and 5). However, CTS 3.1.1.6 requires the RCS loop to include the steam generator and at least one associated reactor coolant pump. The ANO application of these requirements do not currently provide for use of the RCS loops in MODE 5 since the steam generator is not capable of providing the necessary cooling; therefore, it is not considered OPERABLE. However, with sufficient water available to the SG secondary side (ITS LCO 3.4.7 and SR 3.4.7.2), the steam generator(s) provide an acceptable backup method of decay heat removal without an operating reactor coolant pump. (Also see DOC M4.) This change is consistent with NUREG-1430.

3.4A-13 In addition, When in cold shutdown with loops not filled (ITS Mode 5 with loops not filled) The RCS loop is not currently considered to be OPERABLE because CTS 3.1.1.6 requires a reactor coolant pump in order to credit the RCS loop and in this condition, loops drained, the reactor coolant pump may not have sufficient fluid for net positive suction head. Therefore, ITS 3.4.8 is considered to be a relaxation of the CTS requirements since the ANO interpretation of 3.1.1.6 would require an OPERABLE reactor coolant pump in order to credit an RCS loop. This change is consistent with NUREG-1430.

L3 The CTS 3.1.1.6 requirements for an operating heat removal loop in MODE 5 are revised to allow one of the required decay heat removal loops to be de-energized for ≤ 2 hours for surveillance testing, and both decay heat removal loops to be removed from operation if both loops are filled and one RCS loop is in operation for heatup into MODE 4. These Notes (ITS LCO 3.4.7, Notes 2 & 3, and LCO 3.4.8, Note 2) are acceptable since the additional restrictions on application of the allowance provided by these Notes provide for sufficient decay heat removal. This change is consistent with NUREG-1430.

L4 CTS 3.1.1.6 Note * part (2) requirements for an operating heat removal loop in MODE 5 are not included in ITS 3.4.8 Note 1. The allowance for both of the required decay heat removal loops to be removed from operation for ≤ 1 hour is retained

provided no operations are permitted that would cause a reduction of the RCS boron concentration, and no draining operations to further reduce the RCS water volume are permitted. The CTS Note requires that the core outlet temperature is maintained at least 10°F below saturation temperature. However, as indicated in the Bases for ITS 3.4.7, this restriction is intended to assure the capability for natural circulation which is not available in the conditions for which ITS 3.4.8 is applicable, i.e., MODE 5 with loops not filled. Therefore, this restriction is unnecessary. This change is consistent with NUREG-1430.

L5 The CTS 3.1.3.7 requirements to "restore..." in 15 minutes or be in "at least hot shutdown" within the next 15 minutes when CTS 3.1.3.2 is not met are revised, in ITS 3.4.3, to require the unit to "restore" in 30 minutes or be in MODE 3 within the next 6 hours. These revised Completion Times are considered to be appropriate for the Required Actions, allowing the activity to be accomplished in a controlled, orderly manner without challenging plant systems. The proposed changes are consistent with NUREG-1430.

L6 CTS 3.4.1 requires two steam generators be capable of removing heat for operation above 280°F. CTS 3.4.2 provides the Actions if Specification 3.4.1 is not met and actually allows the steam generators to be removed from service for up to 24 hours before requiring the unit to be in hot shutdown within the next 12 hours. ITS does not allow operation in MODES 1, 2, and 3 without both steam generators OPERABLE. If the unit is in MODE 3, ITS LCO 3.4.5, Condition A, will allow 72 hours prior to requiring the unit to placed in MODE 4. CTS allowed only 48 hours of operation in hot shutdown (ITS MODE 3) prior to requiring the unit to be placed in cold shutdown (ITS MODE 5). Further, ITS LCO 3.4.5, Condition B requires only that the unit be placed in MODE 4 consistent with the Applicability of both the CTS and ITS. (See also DOC M9.)

L7 CTS 3.1.1.5.B requires one reactor coolant loop to be operating during the equivalent of ITS MODE 3 operation, and if not met, that immediate corrective action be initiated to return the required loop to operation. This CTS requirement is revised for ITS 3.4.5 to allow both reactor coolant loops to be removed from operation provided specific conditions are met, i.e., no operations are permitted that would cause introduction into the RCS, coolant with a boron concentration less than required to meet the SDM of LCO 3.1.1, and core outlet temperature is maintained 10°F below saturation temperature to assure subcooling capability. In addition, this allowance may be used only for limited periods of time. All RCPs may be removed from operation during transition to and from the DHR system for up to 8 hours in any 24 hour period, or otherwise for up to 1 hour during any 8 hour period for any other reason. The allowance is acceptable since the allowance is for a limited time and additional restrictions on application of the allowance provided by the Note provides for sufficient decay heat removal and SDM.

3.4A-09

3.4A-10

L8 CTS 4.27.3 requires steam generator OPERABILITY to be based on secondary side water level for each required steam generator. The CTS requires steam generators to be OPERABLE "whenever the reactor coolant average temperature is above 280°F"

(CTS 3.1.1.2.A), and allows "whenever the reactor coolant average temperature is at or below 280°F, but the reactor above the refueling shutdown condition," a steam generator to be used to fulfill the requirement for decay heat removal. CTS 4.27.3 is applicable for either condition and requires the steam generator secondary side water level to be ≥ 20 inches on the startup range. In MODES 1, 2, 3, and 4, the capability for circulation is typically provided by either the reactor coolant pumps or the decay heat removal pumps, and adequate heat removal can be accomplished with < 20 inches of secondary side water level. Further, the minimum level is not required for decay heat removal via the steam generators in MODES 1, 2, 3, and 4, as long as emergency feedwater (EFW) is provided by the motor driven EFW pump. LCO 3.7.5 requires that the EFW System be OPERABLE to provide this feedwater in MODES 1, 2, and 3, and in MODE 4 when the steam generator is relied upon for heat removal. Therefore, there is no need to require a minimum secondary side water level in the steam generators in MODES 1, 2, 3, or 4.

L9 The shutdown actions in CTS 3.1.1.7.A and 3.1.1.7.B are proposed for deletion. CTS 3.1.1.7 established requirements for operable reactor coolant system vent valves. These requirements are proposed for relocation to the TRM because they do not satisfy any of the 10 CFR 50.36 Criteria for retention in the ITS. The vent valves are intended to provide a means of venting noncondensible gases from the reactor coolant system which could inhibit natural circulation. ANO-1 proposes to administratively control these valves in accordance with the requirements of the Maintenance Rule, 10 CFR 50.65 and 10 CFR 50.59. The deletion of these actions is consistent with NUREG-1430 in that the NUREG established no requirements pertaining to reactor coolant system vent valves.

L10 The shutdown actions in CTS 3.1.5.2, 3.1.5.3 and 3.1.5.4 are proposed for deletion. CTS 3.1.5.1 established requirements for reactor coolant system chemistry control. These requirements are proposed for relocation to the TRM because they do not satisfy any of the 10 CFR 50.36 Criteria for retention in the ITS. As stated in the CTS Bases, the chemistry specifications function to protect the integrity of the reactor coolant system pressure boundary. But also stated in the CTS Bases, the limits chosen are a decade below those which could result in damage to the materials found in the RCS pressure boundary even if maintained for an extended period of time. Therefore, ANO-1 proposes to administratively control the actions for out of specification chemistry parameters. The removal of the shutdown actions provides an increased opportunity to correct the non-compliance condition without inducing system upset and reduces the potential for unplanned transients as a result of the unit shutdown. The deletion of these actions is consistent with NUREG-1430 in that the NUREG established no requirements pertaining to reactor coolant chemistry control.

L11 The requirements of CTS 3.1.1.1.B, 3.1.1.5.B, 3.1.1.6.B and the footnote associated with 3.1.1.6 are revised to allow operations that may result in a limited addition of positive reactivity in the event forced coolant flow is not available. During these conditions, various unit operations must be continued. RCS inventory must be maintained, and RCS temperature must be controlled. These activities necessarily involve additions to the RCS of cooler water (a positive reactivity effect in most cases)

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and may involve inventory makeup from sources that are at boron concentrations that are less than the RCS boron concentration. The addition of this allowance (LCO 3.4.5 Required Action C.1, LCO 3.4.6 Note a, LCO 3.4.6 Required Action B.1, LCO 3.4.7 Note 1a, LCO 3.4.7 Required Action C.1, LCO 3.4.8 Note 1a, LCO 3.4.8 Required Action B.1) is acceptable, since controls are maintained to provide assurance that the minimum boron concentration, and thus a minimum SDM, is maintained as specified in LCO 3.1.1. This change is consistent with NUREG-1430, as modified by generic change TSTF-286, Rev 2.

LESS RESTRICTIVE -- ADMINISTRATIVE DELETION OF REQUIREMENTS

LA1 This information has been moved to the Bases. This information provides details of design or process which are not directly pertinent to the actual requirement, i.e., Definition, Limiting Condition for Operation or Surveillance Requirement, but rather describe additional unnecessary details such as an acceptable method of compliance. Since these details are not necessary to adequately describe the actual regulatory requirement, they can be moved to a licensee controlled document without a significant impact on safety. Placing these details in controlled documents provides adequate assurance that they will be maintained. The Bases will be controlled by the Bases Control Process in Chapter 5 of the proposed Technical Specifications. This change is consistent with NUREG-1430.

CTS Location	New Location
Table 2.3-1	Bases 3.4.4, BACKGROUND
3.1.1.2.A	Bases 3.4.4, LCO
3.1.1.2.A	Bases 3.4.5, LCO
3.1.1.5.A	Bases 3.4.5, LCO
3.1.1.6	Bases 3.4.6, LCO
3.1.1.6	Bases 3.4.7, LCO
3.1.2.6	Bases 3.4.3, RA A.2
3.1.2.7	Bases 3.4.3, BACKGROUND
3.1.2.8	Bases 3.4.3, BACKGROUND
3.4.1.1	Bases 3.4.4, LCO
3.4.1.1	Bases 3.4.5, LCO

LA2 This information has been moved to the Technical Requirements Manual (TRM). This information provides details of design or process which are not directly pertinent to the actual requirement, i.e., Definition, Limiting Condition for Operation or Surveillance Requirement, but rather describe additional unnecessary details such as an acceptable method of compliance. Since these details are not necessary to adequately describe the actual regulatory requirement, they can be moved to a licensee controlled document without a significant impact on safety. Placing these details in controlled documents provides adequate assurance that they will be maintained. The TRM will be controlled in accordance with the requirements of 10 CFR 50.59. This change is consistent with NUREG-1430.

CTS Location	New Location
3.1.1.7	TRM
3.1.5.1	TRM
Table 4.1-2, Item 16	TRM
Table 4.1-3, Item 1.e	TRM
Table 4.1-3, Item 1.e Note (8)	TRM

LA3 This information has been moved to the Technical Requirements Manual (TRM). CTS 3.1.1.4 provides requirements for reactor internals vent valves. The LCO statement requires the acceptance criteria of CTS 4.1 be applied, which is specifically reflected in Item 15 of Table 4.1-2 as an 18 month surveillance. This Specification contains no actions and excludes application of LCO 3.0.3. As such, the CTS requirement functions solely as a surveillance requirement. The reactor internal vent valves are ASME components, which require inservice inspection to demonstrate that they retain structural integrity. These valves are provided as part of the core support structure to relieve pressure resulting from steam generation in the core following a postulated reactor coolant inlet (cold leg) pipe rupture so that the core will be rapidly recovered by coolant. These eight valves function similarly to a check valve with their normal operating position closed. These valves are therefore passive devices for which testing to demonstrate OPERABILITY is done each refueling outage since testing can only be done with the reactor vessel head off.

Since there is no indication available to the operator of the position of these valves and no testing that can be performed online, this Specification does not reflect requirements of immediate importance to the operator. As such, these details are not necessary to be retained in the ITS to protect the public health and safety. Placing these details in controlled documents provides adequate assurance that they will be maintained. The TRM will be controlled in accordance with the requirements of 10 CFR 50.59. This change is consistent with NUREG-1430.

<Add 3.4.1> M10

<LATER> (2.0)

LATER

2. SAFETY LIMITS AND LIMITING SAFETY SYSTEM SETTINGS

2.1 SAFETY LIMITS, REACTOR CORE

Applicability

Applies to reactor thermal power, reactor power imbalance, reactor coolant system pressure, coolant temperature, and coolant flow when the reactor is critical.

Objective

To maintain the integrity of the fuel cladding.

Specification

2.1.1 The maximum local fuel pin centerline temperature shall be ≤ 5080 - (6.5 x 10^{-3} x (Burnup, MWD/MTU)°F) for TACO2 applications and ≤4642 - (5.8 x 10^{-3} x (Burnup, MWD/MTU)°F for TACO3 applications. Operation within this limit is ensured by compliance with the Axial Power Imbalance protective limits preserved by Table 2.3-1 "Reactor Protection System Trip Setting Limits," as specified in the COLR.

2.1.2 The departure from nucleate boiling ratio shall be maintained greater than the limits of 1.3 for the BAW-2 correlation and 1.18 for the BWC correlation. Operation within this limit is ensured by compliance with Specification 2.1.3 and with the Axial Power Imbalance protective limits preserved by Table 2.3-1 "Reactor Protection System Trip Setting Limits," as specified in the COLR.

2.1.3 Reactor Coolant System (RCS) core outlet temperature and pressure shall be maintained above and to the left of the Variable Low RCS Pressure-Temperature Protective Limits as specified in the COLR.

A2

Bases

To maintain the integrity of the fuel cladding and to prevent fission product release, it is necessary to prevent overheating of the cladding under normal operating conditions. This is accomplished by operating within the nucleate boiling regime of heat transfer, wherein the heat transfer coefficient is large enough so that the clad surface temperature is only slightly greater than the coolant temperature. The upper boundary of the nucleate boiling regime is termed departure from nucleate boiling (DNB). At this point there is a sharp reduction of the heat transfer coefficient, which could result in high cladding temperatures and the possibility of cladding failure. Although DNB is not an observable parameter during reactor operation, the observable parameters of neutron power, reactor coolant flow, temperature, and pressure can be related to DNB through the use of a critical heat flux (CHF) correlation. The BAW-2(1) and BWC(2) correlations have been developed to predict DNB and the location of DNB for axially uniform and non-uniform heat flux distributions. The BAW-2 correlation applies to Mark-B fuel and the BWC correlation applies to Mark-BZ fuel. The local DNB ratio (DNBR), defined as the ratio of the heat flux that would cause DNB at a particular core location to the actual heat flux, is indicative of the margin to DNB. The minimum value of the DNBR, during steady-state operation, normal operational transients, and anticipated transients is limited to 1.30 (BAW-2) and 1.18 (BWC).

A DNBR of 1.30 (BAW-2) or 1.18 (BWC) corresponds to a 95 percent probability at a 95 percent confidence level that DNB will not occur; this is considered a conservative margin to DNB for all operating conditions. The difference between the actual core outlet pressure and the indicated reactor coolant system pressure for the allowable RC pump combination has been considered in determining the Variable Low RCS Pressure-Temperature Protective Limits.

A2

The Variable Low RCS Pressure-Temperature Protective Limits presented in the COLR represent the conditions at which the DNBR is greater than or equal to the minimum allowable DNBR for the limiting combination of thermal power and number of operating reactor coolant pumps which is based on the nuclear power peaking factors (3) as specified in the COLR with potential fuel densification effects.

The Axial Power Imbalance Protective Limits in the COLR are based on the more restrictive of two thermal limits and include the effects of potential fuel densification:

1. The DNBR limit produced by the limiting combination of the radial peak, axial peak, and position of the axial peak.
2. The combination of radial and axial peak that prevents central fuel melting at the hot spot as given in the COLR.

Power peaking is not a directly observable quantity and therefore limits have been established on the basis of the reactor power imbalance produced by the power peaking.

The flow rates for the Variable Low RCS Pressure-Temperature Protective Limits specified in the COLR correspond to the expected minimum flow rates with four pumps, three pumps, and one pump in each loop.

The Variable Low RCS Pressure-Temperature Protective Limit for four reactor coolant pumps operating is the most restrictive of all possible reactor coolant pump maximum thermal power combinations as specified in the COLR. The Variable Low RCS Pressure-Temperature Protective Limits in the COLR represent the conditions at which the DNBR limit is predicted at the maximum possible thermal power for the number of reactor coolant pumps in operation. If the actual pressure/temperature point is below and to the right of the pressure/temperature line, the Variable Low RCS Pressure-Temperature Protective Limit is exceeded. The local quality at the point of minimum DNBR is less than 22 percent (BAW-2)(1) or 26 percent (BWC)(2).

Using a local quality limit of 22 percent (BAW-2) or 26 percent (BWC) at the point of minimum DNBR as a basis for less than four reactor coolant pumps operating of the Variable Low RCS Pressure-Temperature Protective Limits specified in the COLR is a conservative criterion even though the quality at the exit is higher than the quality at the point of minimum DNBR.

The DNBR as calculated by the BAW-2 or the BWC correlation continually increases from point of minimum DNBR, so that the exit DNBR is always higher and is a function of the pressure.

A2

The maximum thermal power, as a function of reactor coolant pump operation is limited by the power level trip produced by the flux-flow ratio (percent flow x flux-flow ratio), plus the appropriate calibration and instrumentation errors.

For each combination of operating reactor coolant pumps of the Variable Low RCS Pressure-Temperature Protective Limits specified in the COLR, a pressure-temperature point above and to the left of the curve would result in a DNBR greater than 1.30 (BAW-2) or 1.18 (BWC) or a local quality at the point of minimum DNBR less than 22 percent (BAW-2) or 26 percent (BWC) for that particular reactor coolant pump combination. The Variable Low RCS Pressure-Temperature Protective Limit for four reactor coolant pumps operating is the most restrictive because any pressure-temperature point above and to the left of this curve will be above and to the left of the other curves.

REFERENCES

(1) Correlation of Critical Heat Flux in a Bundle Cooled by Pressurized Water, BAW-10000A, May, 1976.

(2) BWC Correlation of Critical Heat Flux, BAW-10143P-A, April, 1985.

(3) FSAR, Section 3.2.3.1.1.c.

A1

M1

LA1 Bases

Table 2.3-1
Reactor Protection System Trip Setting Limits

3.4.4 LCO
3.4.4 RA A.1

	Four Reactor Coolant Pumps Operating (Nominal Operating Power - 100%)	Three Reactor Coolant Pumps Operating (Nominal Operating Power, 75%)	One Reactor Coolant Pump Operating in Each Loop[d] (Nominal Operating Power, 49%)	Shutdown Bypass
Nuclear power, % of rated, max	104.9	104.9	104.9	5.0[a]
Nuclear Power based on flow[b] and imbalance, % of rated, max	Protection System Maximum Allowable Setpoints for Axial Power Imbalance envelope in COLR	Protection System Maximum Allowable Setpoints for Axial Power Imbalance envelope in COLR	Protection System Maximum Allowable Setpoints for Axial Power Imbalance envelope in COLR	Bypassed
Nuclear Power based on pump monitors, % of rated, max[c]	NA	NA	55	Bypassed
High RC system pressure, psig, max	2355	2355	2355	1720[a]
Low RC system pressure, psig. min	1800	1800	1800	Bypassed
Variable low RC system pressure, psig, min	Specified in RCS Pressure-Temperature Protective Maximum Allowable Setpoints figure in COLR	Specified in RCS Pressure-Temperature Protective Maximum Allowable Setpoints figure in COLR	Specified in RCS Pressure-Temperature Protective Maximum Allowable Setpoints figure in COLR	Bypassed
RC temp, F, max	618	618	618	618
High reactor building pressure, psig, max	4(18.7 psia)	4(18.7 psia)	4(18.7 psia)	4(18.7 psia)

<LATER> (3.3A)

LATER

(a) Automatically set when other segments of the RPS (as specified) are bypassed.

(b) Reactor coolant system flow, %

(c) The pump monitors also produce a trip on (a) loss of two RC pumps in one RC loop, and (b) loss of one or two RC pumps during two-pump operation.

3.4.4 RA A.1
RA B.1

(d) Operation with one Reactor Coolant Pump operating in each loop is limited to 24 hrs. with the reactor critical.

M1

3.4.4

3.4.4 3.4.7
3.4.5 3.4.8
3.4.6

<Add 3.4.5 LCO Note> L12
<Add SR 3.4.4.1> M3

No operations are permitted that would cause introduction into the RCS, coolant with boron concentration less than required to meet SDM of LCO 3.1.1

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3.1 REACTOR COOLANT SYSTEM

~~**Applicability**~~

~~Applies to the operating status of the reactor coolant system.~~

~~**Objective**~~

~~To specify those limiting conditions for operation of the reactor coolant system which must be met to ensure safe reactor operations.~~

~~3.1.1 Operational Components~~

~~**Specification**~~

A1

3.1.1.1 Reactor Coolant Pumps <Add 3.4.4 RA B.1.> A17

3.4.4 LCO & Appl.
3.4.4 RA A.1

A. Pump combinations permissible for given power levels shall be as shown in Table 2.3-1. Operation with one Reactor Coolant Pump operating in each loop is limited to 24 hours ~~with the reactor critical.~~ MODES 1 & 2 M1

3.4.5 RA C.1
3.4.6 RA B.1
3.4.6/7/8 LCO Note 1a
3.4.7 RA B.1
3.4.8 RA B.1
& <LATER> (3.9)

B. ~~The boron concentration in the reactor coolant system shall not be reduced unless~~ at least one reactor coolant pump or one decay heat removal pump is circulating reactor coolant. With no reactor coolant pumps or decay heat removal pumps running, immediately suspend ~~all~~ operations ~~involving a reduction of boron concentration in the reactor coolant system.~~ that would cause introduction into the RCS, coolant with boron concentration less than required to meet SDM of LCO 3.1.1. L11 A3 L11 LA1 Bases

3.1.1.2 Steam Generator

3.4.4 LCO & Appl.
3.4.5 LCO & Appl.

A. Two steam generators shall be operable ~~whenever the reactor coolant average temperature is above 280°F.~~ in MODES 1, 2 & 3 A1

<LATER> (3.4B) LATER

~~3.1.1.3 Pressurizer Safety Valves~~

~~A. Both pressurizer code safety valves shall be operable when the reactor is critical. With one pressurizer code safety valve inoperable, either restore the valve to operable status within 15 minutes or be in HOT SHUTDOWN within 12 hours.~~

~~B. When the reactor is subcritical, at least one pressurizer code safety valve shall be operable if all reactor coolant system openings are closed, except for hydrostatic tests in accordance with ASME Boiler and Pressure Vessel Code, Section III. The provisions of Specification 3.0.3 are not applicable.~~

~~3.1.1.4 Reactor Internals Vent Valves~~

~~The structural integrity and operability of the reactor internals vent valves shall be maintained at a level consistent with the acceptance criteria in Specification 4.1. The provisions of Specification 3.0.3 are not applicable.~~ LA3 TRM

3.1.1.5 Reactor Coolant Loops

3.4.4 LCO & Appl.
3.4.5 LCO & Appl.

A. ~~With the reactor coolant average temperature above 280°F,~~ MODE 1, 2, 3 the reactor coolant loops listed below shall be operable: A4

3.4.5
3.4.6

3.4A-09
3.4A-10

<Add 3.4.5 LCO Note 6> L7

1. Reactor Coolant Loop (A) ~~and at least one associated reactor coolant pump.~~ LA1 Bases

2. Reactor Coolant Loop (B) ~~and at least one associated reactor coolant pump.~~

3.4.5 RA A.1
RA B.1

Otherwise, restore the required loops to operable status within 72 hours or reduce the reactor coolant average temperature to ~~less than or equal to 280°F~~ MODE 4 within the next 12 hours. A3 A1

3.4.5 LCO

B. ~~With the reactor coolant average temperature above 280°F,~~ In MODE 3 at least one of the reactor coolant loops listed above shall be in operation. L11

3.4.5 RA C.1
RA C.2

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~~Otherwise, suspend all operations involving a reduction in boron concentration of the Reactor Coolant System~~ that would cause introduction into the RCS, coolant with boron concentration less than required to meet SDM of LCO 3.1.1 and immediately initiate corrective action to return the required loop to operation. L7

3.1.1.6 Decay Heat Removal

3.4.6 LCO & Appl.
[3.4.7 LCO]
[See page 16a-2]
[3.4.8 LCO]
[See page 16a-3]

~~With the reactor coolant average temperature at or below 280°F, but the reactor above the refueling shutdown condition,~~ In MODES 4 and 5 at least two of the coolant loops listed below shall be operable, and at least one loop shall be in operation:* A12

1. Reactor Coolant Loop (A) ~~and its associated steam generator and at least one associated reactor coolant pump.~~ LA1 Bases

2. Reactor Coolant Loop (B) ~~and its associated steam generator and at least one associated reactor coolant pump.~~

3. Decay Heat Removal Loop (A)**

4. Decay Heat Removal Loop (B)**

3.4.6 RA A.1
RA B.2
RA A.2

A. With less than the above required coolant loops OPERABLE, immediately initiate corrective action to return the required coolant loops to OPERABLE status as soon as possible; be in COLD SHUTDOWN within 20 hours. L1

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3.4.6 RA B.1
RA B.2

B. With no coolant loop in operation, suspend ~~all~~ operations ~~involving a reduction in boron concentration of the Reactor Coolant System~~ and immediately initiate corrective action to return the required coolant loop to operation. L11

3.4.6
LCO Note

ANO-247

*All reactor coolant pumps and decay heat removal pumps may be ~~de-energized~~ removed from operation for up to 1 hour provided (1) no operations are permitted that would cause ~~dilution of the reactor coolant system boron concentration~~ and (2) core outlet temperature is maintained at least 10°F below saturation temperature. A14 L11

<LATER>
(1.0)

~~**The normal or emergency power source may be inoperable when the reactor is in a cold shutdown condition.~~ LATER

Introduction into the RCS, coolant with boron concentration less than required to meet SDM of LCO 3.1.1

<Add 3.4.7 LCO Notes 2 & 3> L3

[3.4.5] [See page 16a-1]

1. Reactor Coolant Loop (A) and at least one associated reactor coolant pump.

2. Reactor Coolant Loop (B) and at least one associated reactor coolant pump.

Otherwise, restore the required loops to operable status within 72 hours or reduce the reactor coolant average temperature to less than or equal to 280°F within the next 12 hours.

B. With the reactor coolant average temperature above 280°F, at least one of the reactor coolant loops listed above shall be in operation.

Otherwise, suspend all operations involving a reduction in boron concentration of the Reactor Coolant System and immediately initiate corrective action to return the required loop to operation.

3.4A-12

3.4.7 LCO & Appl. [3.4.6 LCO] [See page 16a-1] [3.4.8 LCO] [See page 16a-3]

3.1.1.6 Decay Heat Removal

In MODES 4 & 5 A12

~~With the reactor coolant average temperature at or below 280°F, but the reactor above the refueling shutdown condition,~~ at least two of the coolant loops listed below shall be operable, and at least one loop shall be in operation:* M4

1. ~~Reactor Coolant Loop (A) and its associated~~ steam generator and at least one associated reactor coolant pump. L2

2. ~~Reactor Coolant Loop (B) and its associated~~ steam generator and at least one associated reactor coolant pump. LA1 Bases

3. Decay Heat Removal Loop (A)**

4. Decay Heat Removal Loop (B)**

3.4A-11

3.4.7 RA A.1/A.2 RA B.1/B.2

A. With less than the above required coolant loops OPERABLE, immediately initiate corrective action to return the required coolant loops to OPERABLE status as soon as possible. ~~Be in COLD SHUTDOWN within 20 hours.~~ A5

3.4.7 RA C.1 C.2

B. With no coolant loop in operation, suspend ~~all~~ operations involving ~~a reduction in boron concentration of the Reactor Coolant System~~ and immediately initiate corrective action to return the required coolant loop to operation. L11

AND-247

3.4.7 LCO Note 1

*All reactor coolant pumps and decay heat removal pumps may be ~~de-energized~~ removed from operation for up to 1 hour provided (1) no operations are permitted that would cause ~~dilution of the reactor coolant system boron concentration~~ introduction into the RCS, coolant with boron concentration less than required to meet SDM of LCO 3.1.1 and (2) core outlet temperature is maintained at least 10°F below saturation temperature. A14 L11

(LATER) (1.0)

~~**The normal or emergency power source may be inoperable when the reactor is in a cold shutdown condition.~~ LATER

<Add 3.4.8, RA B.2> M16 3.4.8

<Add 3.4.8 LCO Note 1.b> M12

<Add 3.4.8 LCO Note 2> L3

[3.4.5]

[See page 16a-1]

1. Reactor Coolant Loop (A) and at least one associated reactor coolant pump.

2. Reactor Coolant Loop (B) and at least one associated reactor coolant pump.

Otherwise, restore the required loops to operable status within 72 hours or reduce the reactor coolant average temperature to less than or equal to 280°F within the next 12 hours.

B. With the reactor coolant average temperature above 280°F, at least one of the reactor coolant loops listed above shall be in operation.

Otherwise, suspend all operations involving a reduction in boron concentration of the Reactor Coolant System and immediately initiate corrective action to return the required loop to operation.

3.1.1.6 Decay Heat Removal

3.4.8 LCO & Appl.

3.4.6 LCO [See page 16a-1]

3.4.7 LCO [See page 16a-2]

In MODES 4 & 5 A12

~~With the reactor coolant average temperature at or below 280°F, but the reactor above the refueling shutdown condition,~~ at least two of the coolant loops listed below shall be operable, and at least one loop shall be in operation:*

L2

1. Reactor Coolant Loop (A) and its associated steam generator and at least one associated reactor coolant pump.

2. Reactor Coolant Loop (B) and its associated steam generator and at least one associated reactor coolant pump.

3. Decay Heat Removal Loop (A)**

4. Decay Heat Removal Loop (B)**

3.4.8 RA A.1 RA B.3

A. With less than the above required coolant loops OPERABLE, immediately initiate corrective action to return the required coolant loops to OPERABLE status as soon as possible; ~~be in COLD SHUTDOWN within 20 hours.~~ A5

3.4.8 RA B.1/B.3

B. With no coolant loop in operation, suspend ~~all~~ operations involving ~~a reduction in boron concentration of the Reactor Coolant System~~ and immediately initiate corrective action to return the required coolant loop to operation. L11

AND-247

3.4.8 LCO Note 1.a

removed from operation A14

*All reactor coolant pumps and decay heat removal pumps may be ~~de-energized~~ for up to 1 hour provided (1) no operations are permitted that would cause ~~dilution of the reactor coolant system boron concentration,~~ and (2) ~~core outlet temperature is maintained at least 10°F below saturation temperature.~~ L11 L4

<LATER> (1.0)

~~**The normal or emergency power sources may be inoperable when the reactor is in a cold shutdown condition.~~ LATER

introduction into the RCS, coolant with boron concentration less than required to meet SDM of LCO 3.1.1

3.1.1.7 Reactor Coolant System Vents

At least one reactor coolant system vent path consisting of at least two valves in series shall be operable at each of the following locations whenever the Reactor Coolant average temperature is above 280F.

1. Reactor vessel head

2. Pressurizer steam space

3. Reactor coolant system Hot Leg high point (2 locations)

LA2

TRM

A. With one of the above vent paths inoperable, startup and/or power operation may continue provided the inoperable vent path is maintained closed; restore the inoperable vent path to operable status within 30 days, or be in hot standby within 6 hours and in cold shutdown within the following 30 hours.

L9

B. With two or more of the above vent paths inoperable, maintain the inoperable vent paths closed and restore at least two vent paths to operable status within 72 hours or be in hot standby within 6 hours and in cold shutdown within the following 30 hours.

3.4.4
3.4.5
3.4.6
3.4.7
3.4.8

BASES:

The plant is designed to operate with both reactor coolant loops and at least one reactor coolant pump per loop in operation, and maintain DNBR above 1.30 (for the BAW-2 correlation) and 1.18 (for the BWC correlation) during all normal operations and anticipated transients.(1)

A2

Whenever the reactor coolant average temperature is above 280°F, single failure considerations require that two loops be operable.

The decay heat removal system suction piping is designed for 300°F thus, the system can remove decay heat when the reactor coolant system is below this temperature. (2,3)

One pressurizer code safety valve is capable of preventing overpressurization when the reactor is not critical since its relieving capacity is greater than that required by the sum of the available heat sources which are pump energy, pressurizer heaters, and reactor decay heat. (4) Both pressurizer code safety valves are required to be in service prior to criticality to conform to the system design relief capabilities. The code safety valves prevent overpressure for a rod withdrawal accident. (5) The pressurizer code safety valve lift setpoint shall be 2,500 psig ±1 percent allowance for error and each valve shall be capable of relieving 300,000 lb/h of saturated steam at a pressure not greater than 3 percent above the set pressure. When testing the pressurizer code safety valves, the "as found" lift setpoint may be 2500 psig +1, -3 percent. However, if found outside the ±1 percent tolerance band, they shall be reset to 2500 psig ±1 percent.

The internals vent valves are provided to relieve the pressure generated by steaming in the core following a LOCA so that the core remains sufficiently covered. Inspection and manual actuation of the internal vent valves (1) ensure operability, (2) ensure that the valves are not open during normal operation, and (3) demonstrate that the valves begin to open and are fully open at the forces equivalent to the differential pressures assumed in the safety analysis.

The reactor coolant vents are provided to exhaust noncondensible gases and/or steam from the primary system that could inhibit natural circulation core cooling. The operability of at least one reactor coolant system vent path from the reactor vessel head, the reactor coolant system highpoints, and the pressurizer steam space ensures the capability exists to perform this function. The valve redundancy of the vent paths serves to minimize the probability of inadvertent actuation and breach of reactor coolant pressure boundary while ensuring that a single failure of a vent valve, power supply, or control system does not prevent isolation of the vent path. Testing requirements are covered in Section 4.0 for the class 2 valves and Table 4.1-2 for the vent paths. These are consistent with ASME Section XI and Item II.B.1 of NUREG-0737, "Clarification of TMI Action Plan Requirements," 11/80.

REFERENCES

(1) FSAR, Tables 9-10 and 4-3 through 4-7
(2) FSAR, Section 4.2.5.1 and 9.5.2.3
(3) FSAR, Section 4.2.5.4
(4) FSAR, Section 4.3.10.4 and 4.2.4
(5) FSAR, Section 4.3.7

< Add 3.4.3 Appl. > A15

3.1.2 Pressurization, Heatup, and Cooldown Limitations

Specification

3.4.3 LCO 3.1.2.1 Hydro Tests

SR 3.4.3.1 SR 3.4.3.2 SR 3.4.3.3 & Note

For thermal steady state system hydro tests, the system may be pressurized to the limits set forth in Specification 2.2 when there are fuel assemblies in the core, under the provisions of 3.1.2.3, and to ASME Code limits when no fuel assemblies are present provide the reactor coolant system limits are to the right of and below the limit line in Figure 3.1.2-1. ~~The provisions of Specifications 3.0.3 are not applicable.~~ M7 A6

~~3.1.2.2 Leak Tests~~

~~Leak tests required by Specification 4.3 shall be conducted under the provision of 3.1.2.3. The provisions of Specification 3.0.3 ar not applicable.~~ A11 M2

3.4.3 LCO & Note 3.1.2.3 The reactor coolant pressure and the system heatup and cooldown rates (with the exception of the pressurizer) shall be limited in accordance with Figure 3.1.2-2 and Figure 3.1.2-3, and are as follows:

Heatup:

SR 3.4.3.1 & Note SR 3.4.3.4 & Note

Allowable combinations of pressure and temperature shall be to the right of and below the limit line in Figure 3.1.2-2. The heatup rates shall not exceed those shown in Figure 3.1.2-2. A16

SR 3.4.3.2 & Note

Cooldown:

Allowable combinations of pressure and temperature for a specific cooldown shall be to the right of and below the limit line in Figure 3.1.2-3. Cooldown rates shall not exceed those shown in Figure 3.1.2-3. M7

~~3.1.2.4 The secondary side of the steam generator shall not be pressurized above 200 psig if the temperature of the steam generator shell is below 100F.~~ R TRM

~~3.1.2.5 The pressurizer heatup and cooldown rates shall not exceed 100F/hr. The spray shall not be used if the temperature difference between the pressurizer and the spray fluid is greater than 430F.~~ R TRM

3.4-07

3.4.3 RA B.1
3.4.3 RA B.2
3.4.3 RA C.1
3.4.3 RA C.2

3.1.2.6 With the limits of Specifications 3.1.2.3 ~~or 3.1.2.4 or 3.1.2.5~~ exceeded, restore the temperature and/or pressure to within the limit within 30 minutes; perform an engineering evaluation to determine the effects of the out-of-limit condition on the fracture toughness properties of the Reactor Coolant System; determine that the Reactor Coolant System remains acceptable for continued operations or be in at least HOT STANDBY MODE 3 within the next 6 hours and ~~reduce the RCS Tavg to less than 200F,~~ Be in MODE 5 while maintaining RCS temperature and pressure below the curve, within the following 30 hours. R TRM LA1 Bases M5 M6 A1 A7

3.1.2.7 ~~Prior to reaching thirty one effective full power years of operation, Figures 3.1.2-1, 3.1.2-2 and 3.1.2-3 shall be updated for the next service period in accordance with 10CFR50, Appendix G. The service period shall be of sufficient duration to permit the scheduled evaluation of a portion of the surveillance data scheduled in accordance with the latest revision of Topical Report BAW-1543(5). The highest predicted adjusted reference temperature of all the beltline region materials shall be used to determine the adjusted reference temperature at the end of the service period. The basis for this prediction shall be submitted for NRC staff review in accordance with Specification 3.1.2.8. The provisions of Specification 3.0.3 are not applicable.~~ LA1 Bases

3.1.2.8 ~~The updated proposed technical specifications referred to in 3.1.2.7 shall be submitted for NRC review at least 90 days prior to the end of the service period.~~ LA1 Bases

<LATER> (3.4B)

3.1.2.9 ~~With the exception of ASME Section XI testing and when the core flood tank is depressurized, during a plant cooldown the core flood tank discharge valves shall be closed and the circuit breakers for the motor operators opened before depressurizing the reactor coolant system below 600 psig.~~ LATER

3.1.2.10 ~~With the exception of ASME Section XI testing, fill and vent of the reactor coolant system, emergency RCS makeup and to allow maintenance of the valves, when the reactor coolant temperature is less than 262°F, the High Pressure Injection motor operated valves shall be closed with their opening control circuits for the motor operators disabled.~~

3.1.2.11 ~~The plant shall not be operated in a water solid condition when the RCS pressure boundary is intact except as allowed by Emergency Operating Procedures and during System Hydrotest.~~

<Add 3.4.3 Condition A Note> M5

<Add 3.4.3 Condition C with Note> M5

BASES

All reactor coolant system components are designed to withstand the effects of cyclic loads due to system temperature and pressure changes.[1] These cyclic loads are introduced by unit load transients, reactor trips, and unit heatup and cooldown operations. The number of thermal and loading cycles used for design purposes are shown in Table 4-8 of the FSAR. The maximum unit heatup and cooldown rates satisfy stress limits for cyclic operation.[2] The 200 psig pressure limit for the secondary side of the steam generator at a temperature less than 100°F satisfies stress levels for temperatures below the DTT.[3]

R TRM

The major components of the reactor coolant pressure boundary have been analyzed in accordance with Appendix G to 10CFR50. Results of this analysis, including the actual pressure-temperature limitations of the reactor coolant pressure boundary, are given in FTI Document 77-1258569-01 [4]. The limiting weld material is being irradiated as part of the B&W Owners Group Integrated Reactor Vessel Material Surveillance Program and the identification and locations of the capsules containing the limiting weld material is discussed in the latest revision to B&W report, BAW-1543.[5] The chemical composition of the limiting weld material is reported in the B&W Report, BAW-2121P [6]. The effect of neutron irradiation on the RT_{NDT} of the limiting weld material is reported in FTI Calculations 32-1245917-00 and 32-1257716-00 [7].

Figures 3.1.2-1, 3.1.2-2, and 3.1.2-3 present the pressure-temperature limit curves for hydrostatic test, normal heatup, and normal cooldown respectively. The limit curves are applicable through the thirty first effective full power year of operation. The service period was reduced by one effective full power year from that assumed in FTI Document 77-1258569-01 to be conservative with respect to independent calculations performed by the NRC staff. The pressure limit is also adjusted for the pressure differential between the point of system pressure measurement and the limiting component for all allowed operating reactor coolant pump combinations.

The pressure-temperature limit lines shown on Figure 3.1.2-2 for reactor criticality and on Figure 3.1.2-1 for hydrostatic testing have been provided to assure compliance with the minimum temperature requirements of Appendix G to 10CFR50 for reactor criticality and for inservice hydrostatic testing.

The actual shift in RT_{NDT} of the beltline region material will be established periodically during operation by removing and evaluating, in accordance with Appendix H to 10CFR50, reactor vessel material irradiation surveillance specimens which are installed near the inside wall of this or a similar reactor vessel in the core region.

A2

The spray temperature difference restriction based on a stress analysis of the spray line nozzle is imposed to maintain the thermal stresses at the pressurizer spray line nozzle below the design limit. Temperature requirements for the steam generator correspond with the measured NDTT for the shell.

R TRM

The heatup and cooldown rates stated in this specification are intended as the maximum changes in temperature in one direction in a one hour period. The actual temperature linear ramp rate may exceed the stated limits for a time period provided that the maximum total temperature difference does not exceed the limit and that a temperature hold is observed to prevent the total temperature difference from exceeding the limit for the one hour period.

Specification 3.1.2.9 is to ensure that the core flood tanks are not the source for pressurizing the reactor coolant system when in cold shutdown.

Specification 3.1.2.10 is to ensure that high pressure injection is not the source of pressurizing the reactor coolant system when in cold shutdown. The LTOP enable temperature has been calculated in accordance with Code Case N-514. Instrument error is not included in the reactor coolant temperature of 262°F.

Specification 3.1.2.11 is to ensure that the reactor coolant system is not operated in a manner which would allow overpressurization due to a temperature transient.

A2

REFERENCES

(1) FSAR, Section 4.1.2.4

(2) ASME Boiler and Pressure Code, Section III, N-415

(3) FSAR, Section 4.3.11.5

R
TRM

(4) FTI Document Number 77-1258569-01

(5) BAW-1543, latest revision

(6) BAW-2121P

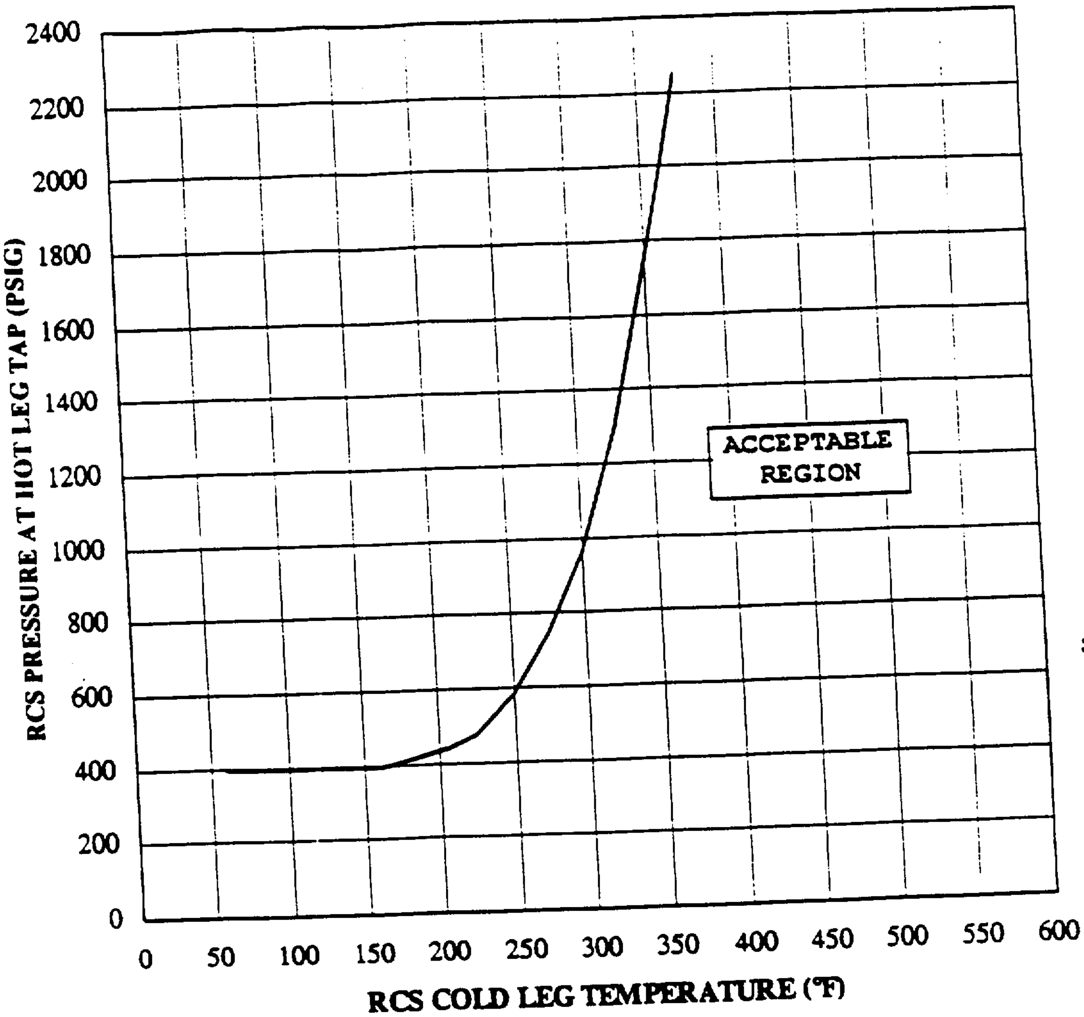
(7) FTI Calculation Numbers 32-1245917-00 and 32-1257716-00

A2

Fig 3.4.3 -3

FIGURE 3.1.2-1

RCS INSERVICE HYDROSTATIC TEST H/U & C/D LIMITS TO 31 EFPY

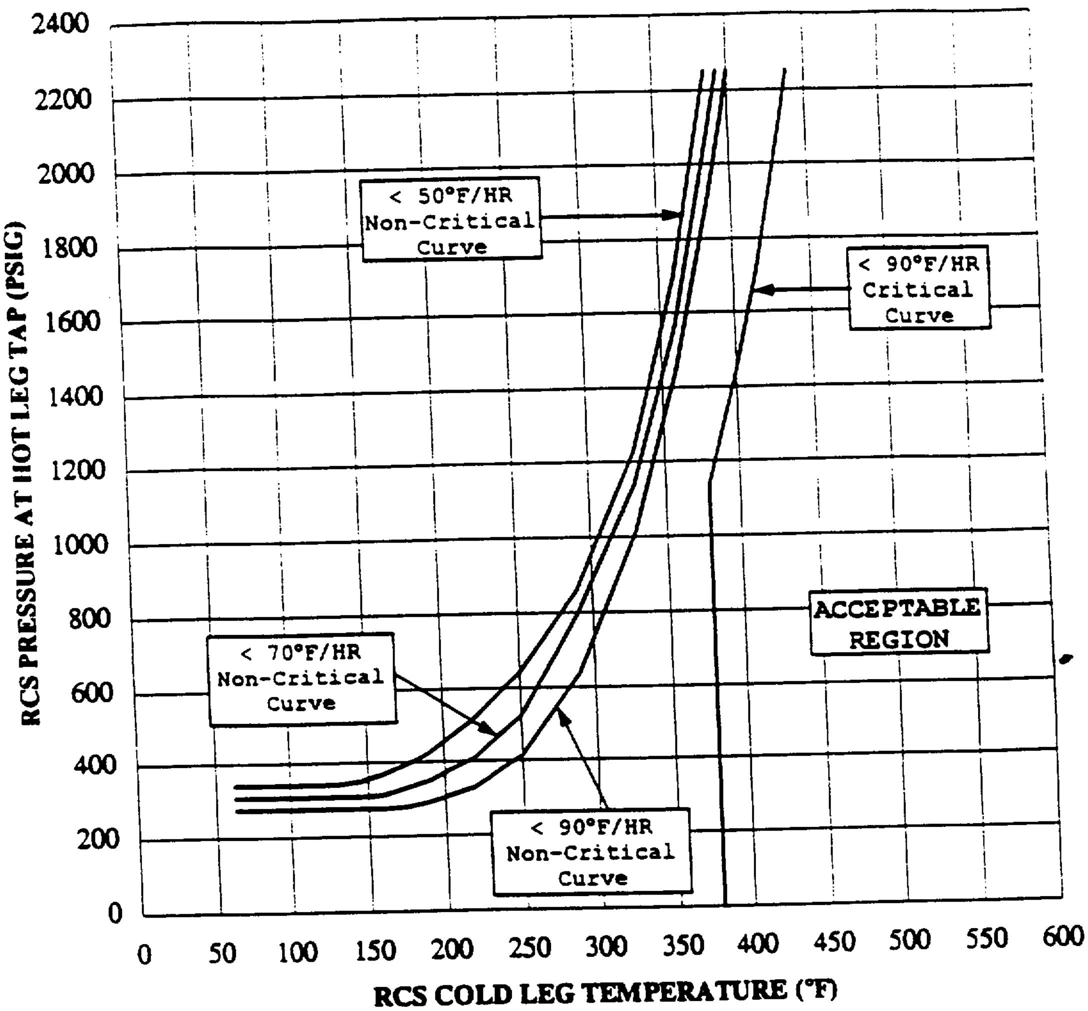


Notes:

1. This curve is not adjusted for instrument error and shall not be used for operation.
2. All Notes on Figure 3.1.2-2 are applicable for heatups. This curve is based on a heatup rate of < 90°F/HR.
3. All Notes on Figure 3.1.2-3 are applicable for cooldowns.

Fig. 3.4.3-1

FIGURE 3.1.2-2
RCS HEATUP LIMITATIONS TO 31 EFPY



Notes:

1. These curves are not adjusted for instrument error and shall not be used for operation.
2. When DHR is in operation with no RCPs operating, the DHR system return temperature shall be used.
3. RCP Operating Restrictions:

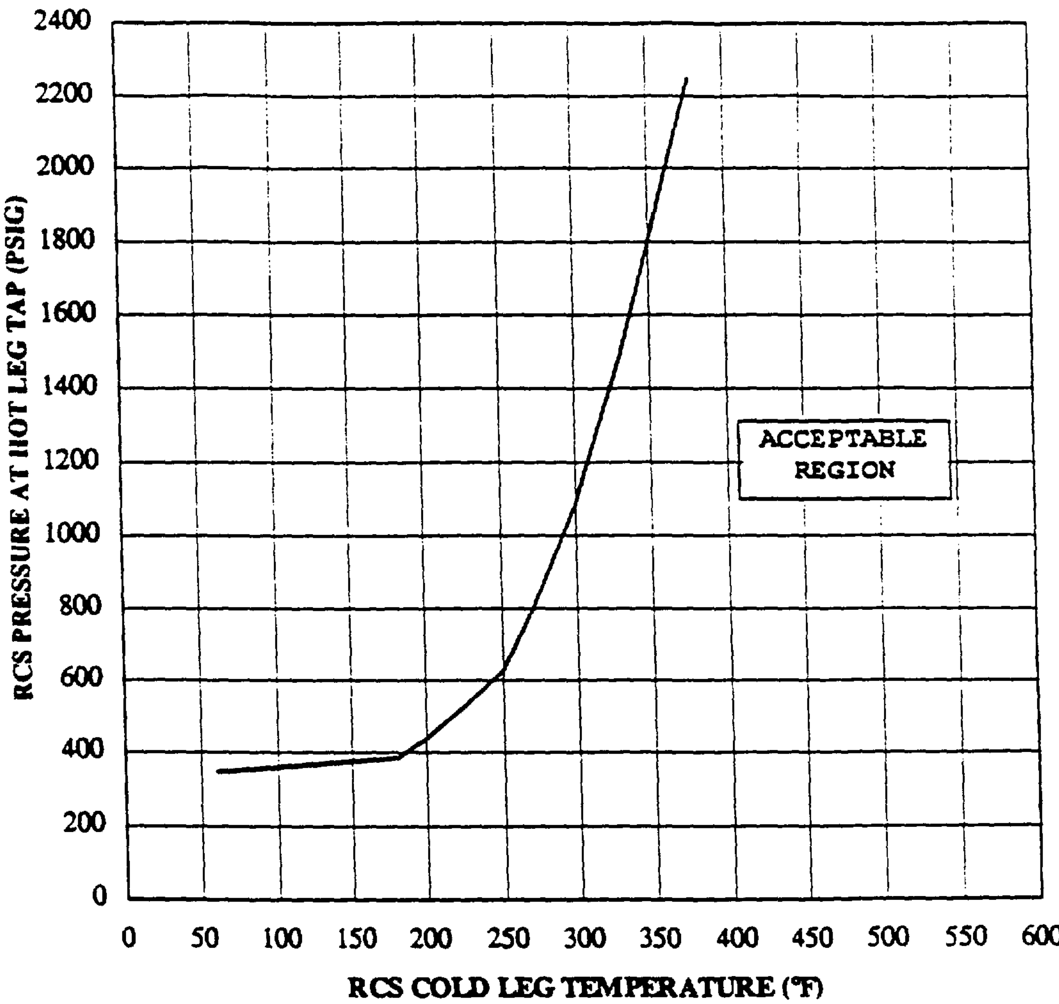
RCS TEMP	RCP RESTRICTIONS
T > 300°F	None
300°F ≥ T ≥ 225°F	≤ 3
225°F > T ≥ 84°F	≤ 2
T < 84°F	No RCPs operating

4. Allowable Heatup Rates:

RCS TEMP	H/U RATE
60°F < T ≤ 84°F	≤ 15°F/HR
T > 84°F	As allowed by applicable curve

Fig. 3.4.3-2

FIGURE 3.1.2-3
RCS COOLDOWN LIMITS TO 31 EFPY



Notes:

1. This curve is not adjusted for instrument error and shall not be used for operation.
2. A maximum step temperature change of 25°F is allowable when securing all RCPs with the DHR system in operation. This change is defined as the RCS temperature prior to securing all the RCPs minus the DHR return temperature after the RCPs are secured. When DHR is in operation with no RCPs operating, the DHR system return temperature shall be used.
3. RCP Operating Restrictions:

RCS TEMP	RCP RESTRICTIONS
T > 255°F	None
150°F ≤ T ≤ 255°F	≤ 2 (See Note 5)
T < 150°F	No RCPs operating

4. Allowable Cooldown Rates:

RCS TEMP	C/D RATE	STEP CHANGE
T ≥ 280°F	100°F/HR	≤ 50°F in any 1/2 HR
280°F > T ≥ 150°F	50°F/HR (See Note 5)	≤ 25°F in any 1/2 HR
T < 150°F	25°F/HR	≤ 25°F in any 1 HR

5. If RCPs are operated < 200°F, then the RCS cooldown rate from 150°F ≤ T ≤ 180°F is reduced to 30°F in 15 hours.

<ADD SR 3.4.2.1> M14

3.4.2 APPL <LATER> (3.1) 3.1.3 ~~Minimum Conditions for Criticality~~ MODES 1+2 M11

Specification

3.4.2 LCO 3.1.3.1 The reactor coolant temperature shall be ~~above~~ ≥ 525F except for portions of low power physics testing when the requirements of Specification 3.1.8 shall apply. A8 <LATER> (3.1) LATER

[See page 21-2] 3.1.3.2 Reactor coolant temperature shall be to the right of the criticality limit of Figure 3.1.2-2.

<LATER> (3.1) 3.1.3.3 ~~When the reactor coolant temperature is below the minimum temperature specified in 3.1.3.1 above, except for portions of low power physics testing when the requirements of Specification 3.1.8 shall apply, the reactor shall be subcritical by an amount equal to or greater than the calculated reactivity insertion due to depressurization.~~ LATER

<LATER> (3.4B) 3.1.3.4 ~~The reactor shall be maintained subcritical by at least 1 percent Δk/k until a steam bubble is formed and an indicated water level between 45 and 305 inches is established in the pressurizer.~~ LATER

<LATER> (3.1, 3.2) 3.1.3.5 ~~Except for physics tests and as limited by 3.5.2.1, safety rod groups shall be fully withdrawn and the regulating rods shall be positioned within their position limits as defined by Specification 3.5.2.5 prior to any other reduction in shutdown margin by deboration or regulating rod withdrawal during the approach to criticality.~~ LATER

<LATER> (3.4B) 3.1.3.6 ~~The reactor shall not be made critical until at least 2 of the 3 emergency-powered pressurizer heater groups are operable. With less than 2 of the 3 required heater groups operable, restore the required heater groups to operable status within 72 hours. If the required heater groups are not restored to operable status within 72 hours, be in hot shutdown within the following 12 hours.~~ LATER

3.4A-07

3.4.3 RA A.1 3.4.2 RA A.1 + <LATER> (3.1 3.4B) 3.1.3.7 With any of the above limits violated, restore the reactor to within the limit in 15 minutes or be in ~~at least Hot Shutdown~~ MODE 3 within the next 15 minutes. A9 +LATER A1

~~Bases~~

~~At the beginning of life of the initial fuel cycle, the moderator temperature coefficient is expected to be slightly positive at operating temperatures with the operating configuration of control rods. (1) Calculations show that above 525F the positive moderator coefficient is acceptable.~~

~~Since the moderator temperature coefficient at lower temperatures will be less negative or more positive than at operating temperature, (2) startup and operation of the reactor when reactor coolant temperature is less than 525F is prohibited except where necessary for low power physics tests.~~

~~The potential reactivity insertion due to the moderator pressure coefficient (2) that could result from depressurizing the coolant from 2100 psia to saturation pressure of 900 psia is approximately 0.1 percent Δk/k.~~

~~During physics tests, special operating precautions will be taken. In addition, the strong negative Doppler coefficient (1) and the small integrated Δk/k would limit the magnitude of power excursion resulting from a reduction of moderator density.~~ A2

[See Page 21-1]

3.1.3 Minimum Conditions for Criticality

Specification

3.1.3.1 The reactor coolant temperature shall be above 525F except for portions of low power physics testing when the requirements of Specification 3.1.8 shall apply.

SR 3.4.3.4 Note

A16

3.4.3 LCO SR 3.4.3.4

3.1.3.2 Reactor coolant temperature shall be to the right of the criticality limit of Figure ~~3.1.2-2~~ 3.4.3-1

3.1.3.3 When the reactor coolant temperature is below the minimum temperature specified in 3.1.3.1 above, except for portions of low power physics testing when the requirements of Specification 3.1.8 shall apply, the reactor shall be subcritical by an amount equal to or greater than the calculated reactivity insertion due to depressurization.

[See page 21-1]

3.1.3.4 The reactor shall be maintained subcritical by at least 1 percent Δk/k until a steam bubble is formed and an indicated water level between 45 and 305 inches is established in the pressurizer.

3.1.3.5 Except for physics tests and as limited by 3.5.2.1, safety rod groups shall be fully withdrawn and the regulating rods shall be positioned within their position limits as defined by Specification 3.5.2.5 prior to any other reduction in shutdown margin by deboration or regulating rod withdrawal during the approach to criticality.

3.1.3.6 The reactor shall not be made critical until at least 2 of the 3 emergency-powered pressurizer heater groups are operable. With less than 2 of the 3 required heater groups operable, restore the required heater groups to operable status within 72 hours. If the required heater groups are not restored to operable status within 72 hours, be in hot shutdown within the following 12 hours.

3.4A-07

3.4.3 R.A. B.1/C.1

3.1.3.7 With any of the above limits violated, restore the reactor to within the limit in ~~15~~ 30 minutes or be in ~~at least Hot Shutdown~~ MODE 3 within the next ~~15 minutes~~ 6 hours

A1

L5

Bases

<ADD 3.4.3 RA B.2 + C.2>

M8

~~At the beginning of life of the initial fuel cycle, the moderator temperature coefficient is expected to be slightly positive at operating temperatures with the operating configuration of control rods. (1) Calculations show that above 525F the positive moderator coefficient is acceptable.~~

~~Since the moderator temperature coefficient at lower temperatures will be less negative or more positive than at operating temperature, (2) startup and operation of the reactor when reactor coolant temperature is less than 525F is prohibited except where necessary for low power physics tests.~~

~~The potential reactivity insertion due to the moderator pressure coefficient (2) that could result from depressurizing the coolant from 2100 psia to saturation pressure of 900 psia is approximately 0.1 percent Δk/k.~~

~~During physics tests, special operating precautions will be taken. In addition, the strong negative Doppler coefficient (1) and the small integrated Δk/k would limit the magnitude of power excursion resulting from a reduction of moderator density.~~

A2

A2

The requirement that the reactor is not to be made critical below the limits of Figure 3.1.2-2 provides increased assurances that the proper relationship between primary coolant pressure and temperatures will be maintained relative to the NDTT of the primary coolant system. Heatup to this temperature will be accomplished by operating the reactor coolant pumps.

If the shutdown margin required by Specification 3.5.2 is maintained, there is no possibility of an accidental criticality as a result of a decrease of coolant pressure.

The requirement for pressurizer bubble formation and specified water level when the reactor is less than one (1) percent subcritical will assure that the reactor coolant system cannot become solid in the event of a rod withdrawal accident or a start-up accident and that the water level is above the minimum detectable level.

The requirement that 2 of the 3 emergency-powered pressurizer heaters be operable provides assurance that sufficient heater capacity (≥126 kw) is available to provide reactor coolant system pressure control during a loss of off-site power.

The requirement that the safety rod groups be fully withdrawn before criticality ensures shutdown capability during startup. This does not prohibit rod latch confirmation, i.e., withdrawal by group to a maximum of 3 inches withdrawn of all seven groups prior to safety rod withdrawal.

The requirement for regulating rods being within their rod position limits ensures that the shutdown margin and ejected rod criteria at hot zero power are not violated.

REFERENCES

(1) FSAR, Section 3

(2) FSAR, Section 3.2.2.1.5

3.4A

3.1.5 Chemistry

Applicability

Applies to the limiting conditions of reactor coolant chemistry for continuous operation of the reactor.

Objective

To protect the reactor coolant system from the effects of impurities in the reactor coolant.

Specification

3.1.5.1 The following limits shall not be exceeded for the listed reactor coolant conditions.

Contaminant	Specification	Reactor Coolant Conditions
Oxygen as O_2	0.10 ppm max	above 250°F
Chloride as Cl^-	0.15 ppm max	above cold shutdown conditions
Fluoride as F^-	0.15 ppm max	above cold shutdown conditions

LA2

TRM

3.1.5.2 During operation above 250°F, if any of the specifications in 3.1.5.1 is exceeded, corrective action shall be initiated within 8 hours. If the concentration limit is not restored within 24 hours after initiation of corrective action, the reactor shall be placed in a cold shutdown condition using normal procedures.

L10

3.1.5.3 During operations between 250°F and cold shutdown conditions, if the chloride or fluoride specification in 3.1.5.1 are exceeded, corrective action shall be initiated within 8 hours to restore the normal operating limits. If the specifications are not restored within 24 hours after initiation of corrective action, the reactor shall be placed in a cold shutdown condition using normal procedures.

3.1.5.4 If the oxygen concentration and either the chloride or fluoride concentration of the primary coolant system exceed 1.0 ppm, the reactor shall be immediately brought to the hot shutdown condition using normal shutdown procedures, and action is to be taken immediately to return the system to within normal operation specifications. If specifications given in 3.1.5.1 have not been reached in 12 hours, the reactor shall be brought to a cold shutdown condition using normal procedures.

Bases

By maintaining the chloride, fluoride, and oxygen concentration in the reactor coolant within the specifications, the integrity of the reactor coolant system is protected against potential stress corrosion attack ([1,2]).

A2

Bases - (Continued)

The oxygen concentration in the reactor coolant system is normally expected to be below detectable limits since dissolved hydrogen is used when the reactor is critical and a residual of hydrazine is used when the reactor is subcritical to control the oxygen. The requirement that the oxygen concentration not exceed 0.1 ppm is added assurance that stress corrosion cracking will not occur (3).

If the oxygen, chloride, or fluoride limits are exceeded, measures can be taken to correct the condition (e.g., switch to the spare demineralizer, replace the ion exchanger resin, increase the hydrogen concentration in the makeup tank, etc.) and further because of the time dependent nature of any adverse effects arising from halogen or oxygen concentrations in excess of the limits, it is unnecessary to shutdown immediately. A2

The oxygen and halogen limits specified are at least an order of magnitude below concentrations which could result in damage to materials found in the reactor coolant system even if maintained for an extended period of time. (3) Thus, the period of eight hours to initiate corrective action and the period of 24 hours thereafter to perform corrective action to restore the concentration within the limits have been established. The eight hour period to initiate corrective action allows time to ascertain that the chemical analyses are correct and to locate the source of contamination. If corrective action has not been effective at the end of 24 hours, then the reactor coolant system will be brought to the cold shutdown condition using normal procedures and corrective action will continue.

The maximum limit of 1 ppm for the oxygen and halogen concentration that will not be exceeded was selected as the hot shutdown limit because these values have been shown to be safe at 500°F. (4)

References

(1) FSAR Section 4.1.2.7
(2) FSAR Section 9.2.2
(3) Corrosion and Wear Handbook, O.J. DePaul, Editor
(4) Stress Corrosion of Metals, Logan

3.4.4
3.4.5

3.4 STEAM AND POWER CONVERSION SYSTEM

Applicability

Applies to the turbine cycle components for removal of reactor decay heat.

Objective

To specify minimum conditions of the turbine cycle equipment necessary to assure the capability to remove decay heat from the reactor core.

Specifications

MODES 1,2,3 — A1 & LATER

3.4.4 APPL
3.4.5 APPL
& <LATER> (3.7)

3.4.1 ~~The reactor shall not be heated above 280°F~~ unless the following conditions are met:

3.4.4 LCO
3.4.5 LCO

1. Capability to remove decay heat by use of two steam generators. — LA1 BASES

<LATER> (3.7) — LATER

2. ~~Fourteen of the steam system safety valves are operable.~~
3. ~~A minimum usable volume of 32,300 gallons of water is available in Tank T41B.~~
4. ~~(Deleted)~~
5. ~~Both main steam block valves and both main feedwater isolation valves are operable.~~

3.4.4 RA B.1
3.4.5 RA A.1, B.1
& <LATER> (3.7)

3.4.2 Components required to be operable by Specification 3.4.1 shall not be removed from service for more than 24 consecutive hours. If the system is not restored to meet the requirements of Specification 3.4.1 within 24 hours, the reactor shall be placed in the hot shutdown condition within 12 hours. If the requirements of Specification 3.4.1 are not met within an additional 48 hours, the reactor shall be placed in the cold shutdown condition within 24 hours. — M9, L6 & LATER

<LATER> (3.7) — LATER

3.4.3 ~~Two (2) EFW trains shall be operable as follows:~~

1. The motor driven EFW pump and its associated flow path shall be operable when the RCS is above CSD conditions and any Steam Generator is relied upon for heat removal.
2. The turbine driven EFW pump and its associated flow path shall be operable when the RCS temperature is ≥ 280°F.**

<LATER> (3.7) — LATER

* Except that during hydrotests, with the reactor subcritical, fourteen of the steam system safety valves may be gagged and two (one on each header), may be reset for the duration of the test, to allow the required pressure for the test to be attained.

** Except that the surveillance testing of the turbine driven EFW pump shall be performed at the appropriate plant conditions as specified by Surveillance Requirement 4.8.1.

LAR

A18

Table 4.1-2 (Cont.)

Minimum Equipment Test Frequency

Item	Test	Frequency
⟨LATER⟩ (3.4B) 11. Decay heat removal system isolation valve automatic closure and isolation system	Functioning	Each Refueling Shutdown — LATER
⟨LATER⟩ (5.0) 12. Flow limiting annulus on main feedwater line at reactor building penetration	Verify, at normal operating conditions, that a gap of at least 0.025 inches exists between the pipe and the annulus.	One year, two years, three years, and every five years thereafter measured from date of initial test. — LATER
⟨LATER⟩ (3.7) 13. Main steam isolation valves	a. Exercise through approximately 10% travel b. Cycle	a. Quarterly b. Every 18 months — LATER
14. Main feedwater isolation valves	a. Exercise through approximately 5% travel b. Cycle	a. Quarterly b. Every 18 months
15. Reactor internals vent valves	Demonstrate operability by: a. Conducting a remote visual inspection of visually accessible surfaces of the valve body and disc sealing faces and evaluating any observed surface irregularities. b. Verifying that the valve is not stuck in an open position, and c. Verifying through manual actuation that the valve is fully open with a force of ≤ 400 lbs (applied vertically upward).	Each refueling shutdown. — LA3 TRM

Table 4.1-2 (Cont.)

Minimum Equipment Test Frequency

	Item	Test	Frequency	
	~~16. RCS Vent Paths~~	~~Demonstrate operability by flow verification~~	~~At least once per 18 months during cold shutdown~~	LA2 TRM
⟨LATER⟩ (3.4B)	~~17. PORV~~	~~Exercise~~	~~End of each refueling outage~~	LATER

3.4A

Table 4.1-3

MINIMUM SAMPLING AND ANALYSIS FREQUENCY

Item	Test	Frequency	
1. Reactor Coolant Samples	a. Gamma Isotopic Analysis	a. Bi-weekly (7)	LATER <LATER> (3.4B)
	b. Gross Activity Determination	b. 3 times/week and at least every third day (1)(6)(	
	c. Gross Radioiodine Determination	c. Weekly (3)(6)(7)	
	d. Dissolved Gases	d. Weekly (7)	LATER <LATER> (3.1)
	e. Chemistry (Cl, F, and O_2)	e. 3 times/week (8)	LA2 TRM
	f. Boron Concentration	f. 3 times/week	LATER & R TRM <LATER> (3.9)
	g. Radiochemical Analysis for $\bar{E}$ Determination (2) (4)	g. Monthly (7)	LATER <LATER> (3.4B)
2. Borated Water Storage Tank Water Sample	Boron Concentration	Weekly and after each makeup	LATER <LATER> (3.5)
3. Core Flooding Tank Sample	Boron Concentration	Monthly and after each makeup	
4. Spent Fuel Pool Water Sample	Boron Concentration	Monthly and after each makeup (9)	LATER
5. Secondary Coolant Samples	a. Gross Radioiodine Concentration	a. Weekly (5)(7)(10	R TRM <LATER> (3.7)
	b. Isotopic Radioiodine Concentration (4)	b. Monthly (7)(10)	
6. Sodium Hydroxide Tank Sample	Sodium Hydroxide Concentration	Quarterly and after each makeup	LATER <LATER> (3.6)

Notes:

(1) A gross radioactivity analysis shall consist of the quantitative measurement of the total radioactivity of the primary coolant in units µCi/gm. The total primary coolant activity shall be the sum of the degassed beta-gamma activity and the total of all identified gaseous activities 15 minutes after the primary system is sampled. Whenever t gross radioactivity concentration exceeds 10% of the limit specified i the Specification 3.1.4.1 or increases by 10 µCi/gm from the previous measured level, the frequency of sampling and analyzing shall be increased to a minimum of once/day until a steady activity level is established.

<LATER> (3.4B) — LATER

<LATER> (3.4B)

(2) A radiochemical analysis shall consist of the quantitative measurement of the activity for each radionuclide which is identified in the primary coolant 15 minutes after the primary system is sampled. The activities for the individual isotopes shall be used in the determination of $\bar{E}$. A radiochemical analysis and calculation of $\bar{E}$ and iodine isotopic activity shall be performed if the measured gross activity changes by more than 10 μCi/gm from the previous measured level. The gamma energy per disintegration for those radioisotopes determined to be present shall be as given in "Table of Isotopes" (1967) and beta energy per disintegration shall be as given in USNRDL-TR-802 (Part II) or other references using the equivalent values for the radioisotopes. — LATER

(3) In addition to the weekly measurement, the radioiodine concentration shall be determined if the measured gross radioactivity concentration changes by more than 10 μCi/gm from the previous measured level.

<LATER> (3.4B & 3.7)

(4) Iodine isotopic activities shall be weighted to give I-131 dose equivalent activity. — LATER

(5) In addition to the weekly measurement, the radioiodine concentration shall be determined if there are indications that the primary to secondary coolant leakage rate has increased by a factor of 2. — R TRM

<LATER> (3.4B)

(6) Whenever the steady state radioiodine or gross radioactivity concentration of prior operation is greater than 1 percent but less than 10 percent of Specification 3.1.4.1, a sample of reactor coolant shall be taken within 24 hours of any reactor criticality and analyzed for radioactive iodines of I-131 through I-135 and gross radioactivity as well as the coolant sample and analyses required by the above. — LATER

Whenever the steady state radioiodine or gross radioactivity concentration of prior operation is greater than 10 percent of Specification 3.1.4.1, a sample of reactor coolant shall be taken prior to any reactor criticality and analyzed for radioactive iodines of I-131 through I-135 and gross radioactivity as well as the coolant sample and analyses required by above.

<LATER> (3.1, 3.4B, 3.7)

(7) Not required when plant is in the cold shutdown condition or refueling shutdown condition. — LATER & R TRM

(8) O_2 analysis is not required when plant is in the cold shutdown condition or refueling shutdown condition. — LA2 TRM

<LATER> (3.7)

(9) Required only when fuel is in the pool and prior to transferring fuel to the pool. — LATER

<LATER> (3.7)

(10) Not required when not generating steam in the steam generators. — LATER & R TRM

<LATER> (3.4B)

(11) The following shall be required until the end of Cycle 2 operation: — LATER

a. Gross radioiodine shall be determined at least three times per week during power operation.

3.4.5
3.4.6
3.4.7
3.4.8

4.27 DECAY HEAT REMOVAL

~~APPLICABILITY~~

~~Applies to surveillance of the decay heat removal system and to the reactor coolant loops and associated reactor coolant pumps as needed for decay heat removal.~~

~~OBJECTIVE~~

~~To assure the operability of the decay heat removal system and the reactor coolant loops as needed for decay heat removal.~~

A1

SPECIFICATION

SR 3.4.5.2
SR 3.4.6.2

4.27.1 The required reactor coolant pumps shall be determined operable once per seven (7) days by verifying correct breaker alignments and indicated power availability.

⟨LATER⟩ (5.0)

~~4.27.2 The required decay heat removal loop(s) shall be determined operable per Specification 4.2.2.~~

LATER

SR 3.4.7.2
3.4.7 LCO #b

4.27.3 The required steam generator(s) shall be determined operable ~~by verifying the secondary side water level to be ≥20 inches on the startup range~~ at least once per 12 hours.

L8

SR 3.4.5.1
SR 3.4.6.1

4.27.4 The required reactor coolant loop(s) shall be determined ~~operable by verifying the required loop(s)~~ to be in operation ~~and circulating reactor coolant~~ at least once per 12 hours.

A1

SR 3.4.6.1
SR 3.4.7.1
SR 3.4.8.1
& ⟨LATER⟩ (3.9)

4.27.5 The required decay heat removal loop shall be determined to be in operation at least once per 12 hours.

& LATER

⟨Add SR 3.4.7.3 with Note
& SR 3.4.8.2 with Note⟩

M13

NO SIGNIFICANT HAZARDS CONSIDERATIONS
GENERIC EVALUATIONS

"R" - Relocation of requirements:

Relocating requirements which do not meet the Technical Specification selection criteria to documents with an established control program allows the Technical Specifications to be reserved only for those conditions or limitations upon reactor operation which are necessary to adequately limit the possibility of an abnormal situation or event giving rise to an immediate threat to the public health and safety, thereby focusing the scope of Technical Specifications.

Therefore, requirements which do not meet the Technical Specification selection criteria in 10 CFR 50.36 have been relocated to other controlled license basis documents. This regulation addresses the scope and purpose of Technical Specifications. In doing so, it establishes a specific set of objective criteria for determining which regulatory requirements and operating restrictions should be included in Technical Specifications. These criteria are as follows:

Criterion 1: Installed instrumentation that is used to detect and indicate in the control room a significant abnormal degradation of the reactor coolant pressure boundary.

Criterion 2: A process variable that is an initial condition of a design basis accident (DBA) or transient analysis that either assumes the failure of or presents a challenge to the integrity of a fission product barrier.

Criterion 3: A structure, system or component that is part of the primary success path and which functions or actuates to mitigate a design basis accident or transient that either assumes the failure of or presents a challenge to the integrity of a fission barrier.

Criterion 4: A structure, system or component which operating experience or probabilistic safety assessment has shown to be significant to public health and safety.

The application of these criteria is provided in the "Application of Selection Criteria to the ANO-1 Technical Specifications." Requirements which met the criteria have been included in the proposed improved Technical Specifications. Entergy Operations proposes to remove the requirements which do not meet the criteria from the Technical Specifications and relocate the requirements to a suitable owner controlled document. The requirements in the relocated Specifications will not be affected by this Technical Specification change. Entergy Operations will initially continue to perform the required operation and maintenance to assure that the requirements are satisfied. Relocating specific requirements for systems or variables will have no impact on the system's operability or the variable's maintenance, as applicable.

License basis document control mechanisms, such as 10 CFR 50.59, 10 CFR 50.54(a)(3), and ITS Section 5, "Administrative Controls," will be utilized for the relocated Specifications as they will be placed in other controlled license basis documents. This would allow Entergy Operations to make changes to these requirements, without NRC approval, as allowed by the applicable regulatory requirements. These controls are considered adequate for assuring structures, systems and components in the relocated Specifications are maintained operable and variables in the relocated Specifications are maintained within limits.

Entergy Operations has evaluated this proposed Technical Specification change and has determined that it involves no significant hazards consideration. This determination has been performed in accordance with the criteria set forth in 10 CFR 50.92(c) as indicated below:

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 which did not meet the criteria for inclusion in Technical Specifications as identified in the Application of Selection Criteria to the ANO-1 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 license basis document and maintained pursuant to the applicable regulatory requirements. 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 necessitate a physical alteration of the plant (no new or different type of equipment will be installed) or change in parameters governing normal plant operation. The proposed change will not impose any different requirements and adequate control of 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 affected requirement will be relocated to an owner controlled license basis document for which future changes will be evaluated pursuant to the requirements of the applicable regulatory requirements. Therefore, this change does not involve a significant reduction in a margin of safety.

NO SIGNIFICANT HAZARDS CONSIDERATIONS
GENERIC EVALUATIONS

"A" - Administrative changes to requirements:

Reformatting and rewording the remaining requirements in accordance with the style of the improved Babcock & Wilcox Standard Technical Specifications in NUREG-1430 will make the Technical Specifications more readily understandable to plant operators and other users. Application of the format and style will also assure consistency is achieved between specifications. As a result, the reformatting and rewording of the Technical Specifications has been performed to make them more readily understandable by plant operators and other users. During this reformatting and rewording process, no technical changes (either actual or interpretational) to the Technical Specifications were made unless they were identified and justified.

Entergy Operations has evaluated this proposed Technical Specification change and has determined that it involves no significant hazards consideration. This determination has been performed in accordance with the criteria set forth in 10 CFR 50.92(c) as indicated below:

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

   The proposed change involves reformatting and rewording of the existing Technical Specifications. The reformatting and rewording process involves no technical changes to existing requirements. 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 necessitate a physical alteration of the plant (no new or different type of equipment will be installed) or changes in parameters governing normal plant operation. The proposed change will not impose any different 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 significantly reduce the margin of safety because it has no impact on any safety analysis assumptions. This change is administrative in nature. As such, there is no technical change to the requirements and therefore, there is no significant reduction in the margin of safety.

NO SIGNIFICANT HAZARDS CONSIDERATIONS
GENERIC EVALUATIONS

"LA" - Less restrictive, Administrative deletion of requirements:

Portions of some Specifications provide information that is descriptive in nature regarding the equipment, system(s), actions or surveillances. This information is proposed to be deleted from the specification and relocated to other license basis documents which are under licensee control. These documents include the TS Bases, Safety Analysis Report (SAR), Technical Requirements Manual, and Programs and Manuals identified in ITS Section 5, "Administrative Controls." The removal of descriptive information is permissible, because the documents containing the relocated information will be controlled through the applicable process provided by the regulatory requirements, e.g., 10 CFR 50.59, 10 CFR 50.54(a)(3), and ITS Section 5, "Administrative Controls." This will not impact the actual requirements but may provide some flexibility in how the requirement is conducted. Therefore, the descriptive information that has been moved continues to be maintained in an appropriately controlled manner.

Entergy Operations has evaluated this proposed Technical Specification change and has determined that it involves no significant hazards consideration. This determination has been performed in accordance with the criteria set forth in 10 CFR 50.92(c) as indicated below:

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

   The proposed change relocates requirements from the Technical Specifications to other license basis documents which are under licensee control. The documents containing the relocated requirements will be maintained using the provisions of applicable regulatory requirements. 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 necessitate a physical alteration of the plant (no new or different type of equipment will be installed) or changes in parameters governing normal plant operation. The proposed change will not impose any different 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 requirements to be transposed from the Technical Specifications to other license basis documents, which are under licensee control, are the same as the existing Technical Specifications. The documents containing the relocated requirements will be maintained using the provisions of applicable regulatory requirements. Therefore, this change does not involve a significant reduction in a margin of safety.

NO SIGNIFICANT HAZARDS CONSIDERATIONS
GENERIC EVALUATIONS

"M" - More restrictive changes to requirements:

The ANO-1 Technical Specifications are proposed to be modified in some areas to impose more stringent requirements than previously identified. These more restrictive modifications are being imposed to be consistent with the improved Babcock & Wilcox Standard Technical Specifications. Such changes have been made after ensuring the previously evaluated safety analysis was not affected. Also, other more restrictive technical changes have been made to achieve consistency, correct discrepancies, and remove ambiguities from the specification.

The modification of the ANO-1 Technical Specifications and the changes made to achieve consistency within the specifications have been performed in a manner such that the most stringent requirements are imposed, except in cases which are individually evaluated.

Entergy Operations has evaluated this proposed Technical Specification change and has determined that it involves no significant hazards consideration. This determination has been performed in accordance with the criteria set forth in 10 CFR 50.92(c) as quoted below:

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 the ANO-1 Technical Specifications. These more stringent requirements are not assumed to be initiators of analyzed events and will not alter assumptions relative to mitigation of accident or transient events. The change has been confirmed to ensure no previously evaluated accident has been adversely affected. The more stringent requirements are imposed 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 necessitate a physical alteration of the plant (no new or different type of equipment will be installed) or changes in parameters governing normal plant operation. The proposed change does impose different requirements. However, these changes do not impact the safety analysis and licensing basis. Thus, this change does not create the possibility of a new or different kind of accident from any accident previously evaluated for ANO-1.

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

The imposition of more stringent requirements prevents a reduction in the margin of plant safety by:

a) Increasing the analytical or safety limit,
b) Increasing the scope of the specification to include additional plant equipment,
c) Increasing the applicability of the specification,
d) Providing additional actions,
e) Decreasing restoration times,
f) Imposing new surveillances, or
g) Decreasing surveillance intervals.

The change is consistent with the safety analysis and licensing basis. Therefore, this change does not involve a reduction in a margin of safety.

NO SIGNIFICANT HAZARDS CONSIDERATIONS STATEMENTS

ITS Section 3.4A: Reactor Coolant System

Entergy Operations has evaluated these proposed Technical Specification changes and has determined that they involve no significant hazards consideration. This determination has been performed in accordance with the criteria set forth in 10CFR 50.92(c) as indicated below:

3.4A L1

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

In MODE 4, the reactor coolant loops and/or decay heat removal loops are required to remove decay heat; however, they are not considered the initiator of any previously analyzed accident. A short extension of the Completion Time for a Required Action does not result in any hardware changes. The Completion Time for performance also does not significantly increase the probability of occurrence for initiation of any analyzed event since the function of the equipment, or limit for the parameter, does not change (and therefore any initiation scenarios are not changed) and the proposed Completion Time extension is short (and therefore does not significantly affect probability). Further, the Completion Time for performance of Required Actions does not significantly increase the consequences of an accident because there is no change in the assumed response of the equipment in performing its specified mitigation functions, or change in the response of the core parameters to assumed scenarios, from that considered during the original Completion Time. Further, a requirement to place the unit in Cold Shutdown is omitted for a condition with no decay heat removal loop available. Since this required action could not be implemented, this change does not involve a significant increase in 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 necessitate a physical alteration of the plant (no new or different type of equipment will be installed) or changes in parameters governing normal plant operation. The proposed change will still ensure prompt restoration of compliance with the limiting condition for operation, or prompt and appropriate compensatory actions are taken. 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?**

This change does not involve a significant reduction in a margin of safety since the OPERABILITY of the equipment and loss of function continue to be evaluated in the same manner. The increase in time allowed for such an evaluation and restoration is minimal and provides additional potential for the preferred action of restoration of the equipment to OPERABLE status, rather than requiring a shutdown transient.

NO SIGNIFICANT HAZARDS CONSIDERATIONS STATEMENTS

3.4A L2

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

In MODE 5, either the reactor coolant loops and/or decay heat removal loops are required to remove decay heat; however, they are not considered the initiator of any previously analyzed accident. The use of the steam generators as a backup to the decay heat removal continues to provide the alternate source of heat removal should one heat removal method be lost. As such the proposed change does not significantly 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 necessitate a physical alteration of the plant (no new or different type of equipment will be installed) or changes in parameters governing normal plant operation. The proposed change will continue to ensure an alternate method of heat removal is available. 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 availability of adequate backup heat removal methodology continues to be assured. Therefore, this change does not involve a significant reduction in a margin of safety.

NO SIGNIFICANT HAZARDS CONSIDERATIONS STATEMENTS

3.4A L3

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

In MODE 5, either the reactor coolant loops and/or decay heat removal loops are required to remove decay heat; however, they are not considered the initiator of any previously analyzed accident. While the principal means of heat removal is the forced flow through the decay heat removal system, the use of the steam generators as a backup to the decay heat removal system continues to provide the alternate source of heat removal should one heat removal method be lost. Further, the additional controls required for de-energizing the pumps and the time available for action are sufficient to assure that the effects of increasing temperature can be identified and acted upon. As such the proposed change does not significantly 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 necessitate a physical alteration of the plant (no new or different type of equipment will be installed) or changes in parameters governing normal plant operation. The proposed change will continue to ensure an alternate method of heat removal is available. 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 availability of adequate backup heat removal methodology continues to be assured. Therefore, this change does not involve a significant reduction in a margin of safety.

NO SIGNIFICANT HAZARDS CONSIDERATIONS STATEMENTS

3.4A L4

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

In MODE 5, either the reactor coolant loops and/or decay heat removal loops are required to remove decay heat; however, they are not considered the initiator of any previously analyzed accident. With the loops not filled, the principal means of heat removal is the forced flow through the decay heat removal system. The removal of a core outlet temperature when the forced flow is allowed to be removed from operation does not affect the consequences of any analyzed event since the core outlet temperature is monitored to assure the capability for natural circulation which is not available when the loops are not filled. Further, the additional controls required for de-energizing the pumps and the time available for action are sufficient to assure that the effects of increasing temperature can be identified and acted upon. As such the proposed change does not significantly 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 necessitate a physical alteration of the plant (no new or different type of equipment will be installed) or changes in parameters governing normal plant operation. The proposed change will continue to ensure an alternate method of heat removal is available. 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 availability of adequate heat removal capability continues to be assured. Therefore, this change does not involve a significant reduction in a margin of safety.

3.4A L5

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

The RCS pressure/temperature limits are provided to maintain the structural integrity of the reactor coolant pressure boundary. However, the limits are not considered the initiator of any previously analyzed accident. Further, a short extension of the Completion Time for a Required Action does not result in any hardware changes. The Completion Time for performance also does not significantly increase the probability of occurrence of any analyzed event since the function of the equipment, or limit for the parameter, does not change (and therefore any initiation scenarios are not changed) and the proposed Completion Time extension is short (and therefore does not significantly affect probability). Further, the Completion Time for performance of Required Actions does not significantly increase the consequences of an accident because there is no change in the assumed response of the equipment in performing its specified mitigation functions, or change in the response of the core parameters to assumed scenarios, from that considered during the original Completion Time.

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 necessitate a physical alteration of the plant (no new or different type of equipment will be installed) or changes in parameters governing normal plant operation. The proposed change will still ensure prompt restoration of compliance with the limiting condition for operation, or prompt and appropriate compensatory actions are taken. 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?

This change does not involve a significant reduction in a margin of safety since the temperature and pressure parameters, and their impact, continue to be evaluated in the same manner. The increase in time allowed for such an evaluation and restoration is minimal and provides additional potential for the preferred action of restoration of the parameters to within their limits, rather than requiring a shutdown transient.

NO SIGNIFICANT HAZARDS CONSIDERATIONS STATEMENTS

3.4A L6

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

In MODE 3, two reactor coolant loops are required to be available to remove decay heat.. A short time is allowed under both the CTS and ITS for the LCO to not be met. However, a short extension of the Completion Time for a Required Action does not result in any hardware changes. The Completion Time for performance also does not significantly increase the probability of occurrence of any analyzed event since the function of the equipment does not change (and therefore any initiation scenarios are not changed) and the proposed Completion Time extension is short (and therefore does not significantly affect probability). Further, the Completion Time for performance of Required Actions does not significantly increase the consequences of an accident because there is no change in the assumed response of the equipment in performing its specified mitigation functions, and no change in the response of the core parameters to assumed scenarios, from that considered during the original Completion Time.

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 necessitate a physical alteration of the plant (no new or different type of equipment will be installed) or changes in parameters governing normal plant operation. The proposed change will still ensure prompt restoration of compliance with the limiting condition for operation, or prompt and appropriate compensatory actions are taken. 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?

This change does not involve a significant reduction in a margin of safety since the OPERABILITY of the equipment and loss of function continue to be evaluated in the same manner. The increase in time allowed for such an evaluation and restoration is minimal and provides additional potential for the preferred action of restoration of the equipment to OPERABLE status, rather than requiring a shutdown transient.

3.4A L7

3.4A-09

3.4A-10

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

In MODE 3, two reactor coolant loops are required to be available to remove decay heat; however, they are not considered the initiator of any previously analyzed accident. This change will allow operation with no RCPs in operation while in MODE 3 for a limited period of time, provided other conditions are met. The change allows operation for ≤ 8 hours in any 24 hour period during the transition to or from the Decay Heat Removal System, and otherwise for ≤ 1 hour in any 8 hour period for any other reason. During the use of this allowance, at least two RCPs must remain Operable. Two compensatory requirements allowing no operations that would cause introduction into the RCS, coolant with boron concentration less than required to meet the SDM of LCO 3.1.1, and maintaining core outlet temperature at least 10°F below saturation temperature are provided to ensure adequate shutdown margin and cooling capacity are maintained. The availability of the reactor coolant loops, the indications available to the operator of rising coolant temperatures, and the time available to the operator to return the loop(s) to operation are sufficient to prevent inadequate cooling. Allowing positive reactivity additions, with no forced coolant flow available, that will not reduce RCS boron concentration below the boron concentration required to meet the minimum SDM of LCO 3.1.1 will not alter the assumed initiation, and hence, will not significantly increase the probability of any evaluated event. The ITS contains provisions that maintain the initial conditions assumed in the analyses. Because these provisions maintain the initial conditions assumed in the safety analyses, prevent intentional actions which may lead to the occurrence of evaluated events, and preserve the mitigatory response mechanisms should an event occur, the consequences of a postulated event from this condition would not be significantly increased.

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

The proposed change does not necessitate a physical alteration of the plant (no new or different type of equipment will be installed) or changes in parameters governing normal plant operation. The proposed change will continue to ensure an alternate method of heat removal is available. 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 allow operation in MODE 3 with no forced RCS flow. The ITS provisions limit the time the unit is allowed to be in this condition and provides controls to ensure adequate SDM and subcooling margin are maintained. Therefore, this change does not result in a significant reduction in a margin of safety.

3.4A L8

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

The steam generators provide a heat transport function to cool the reactor coolant. Some water is necessary in the secondary side of the steam generator for it to be capable of performing this function; however, no specific levels are identified as minimums for this capability, and any level will provide heat transfer. Therefore, as long as water is available to the steam generators, the heat transport function will be provided. Further, there are no safety analyses performed with initial conditions in MODE 5. Therefore, this change does not involve a significant increase in 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 necessitate a physical alteration of the plant (no new or different type of equipment will be installed) but does change a parameter governing normal plant operation, i.e., required steam generator water level. The proposed change will still ensure that heat transfer capability is available by requiring adequate feedwater availability. 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 margin of safety for heat transport via the steam generators is primarily dependent on the availability of water for steaming. While this volume would provide some cooling, it would be insufficient to provide adequate heat removal without an additional source of feedwater. Therefore, the omission of this requirement is not considered to involve a significant reduction in a margin of safety.

NO SIGNIFICANT HAZARDS CONSIDERATIONS STATEMENTS

<u>3.4A L9</u>

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

The deletion of the reactor coolant system (RCS) vent valves shutdown action statements will not alter the requirement for component operability. The deletion of the action statements does not constitute a physical alteration of the plant, nor does it alter the controls governing operation of the components or their associated system. The RCS vent valves are not assumed initiators of any evaluated accident. Therefore, the deletion of the action statements does not involve a significant increase in probability for any previously evaluated accident. The requirements for the RCS vent valves will be relocated from the Technical Specifications to an appropriately controlled license basis document and maintained pursuant to the applicable regulatory requirements. Further, there are no safety analyses which credit the operation of these valves in providing a mitigatory function. Therefore, this change does not involve a significant increase in 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 necessitate a physical alteration of the plant (no new or different type of equipment will be installed) or change in parameters governing normal plant operation. The proposed change will not impose any different requirements and adequate control of 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 affected requirement for the component will be relocated to an licensee controlled license basis document for which future changes will be evaluated pursuant to the requirements of 10 CFR 50.59. Therefore, this change does not involve a significant reduction in a margin of safety.

3.4A L10

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

The deletion of the reactor coolant system (RCS) chemistry shutdown action statements will not alter the requirement for chemistry controls nor the methods of compliance with the chemistry requirements. The deletion of the action statements does not constitute a physical alteration of the plant, nor does it alter the controls governing operation of the components or their associated system. The RCS chemistry parameters are not assumed initiators of any evaluated accident. Therefore, the deletion of the action statements does not involve a significant increase in probability for any previously evaluated accident. The requirements for RCS chemistry control will be relocated from the Technical Specifications to an appropriately controlled license basis document and maintained pursuant to the applicable regulatory requirements. Further, there are no safety analyses which credit the chemistry control parameters in providing a mitigatory function. Therefore, this change does not involve a significant increase in 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 necessitate a physical alteration of the plant (no new or different type of equipment will be installed) or change in parameters governing normal plant operation. The proposed change will not impose any different requirements and adequate control of 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 affected requirement for the parameter will be relocated to an licensee controlled license basis document for which future changes will be evaluated pursuant to the requirements of 10 CFR 50.59. Therefore, this change does not involve a significant reduction in a margin of safety.

3.4A L11

ANO-247

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

Allowing positive reactivity additions, with no forced coolant flow available, that will not reduce RCS boron concentration below the boron concentration required to meet the minimum SDM of LCO 3.1.1 will not alter the assumed initiation, and hence, will not significantly increase the probability of any evaluated event. The ITS contains actions that maintain the initial conditions assumed in the analyses. Because these Required Actions maintain the initial conditions assumed in the safety analyses, prevent intentional actions which may lead to the occurrence of evaluated events, and preserve the mitigatory response mechanisms should an event occur, the consequences of a postulated event from this condition would not be significantly increased.

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

The proposed change does not necessitate a physical alteration to the plant (no new or different type of equipment will be installed) or changes to parameters governing normal plant operation. The proposed change will continue to ensure that adequate boron concentration, and thus adequate SDM, is maintained. Therefore, this change does not result in 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 positive reactivity changes in MODEs 3, 4, and 5 with no forced coolant flow. However, the ITS Required Actions limit such positive reactivity additions to provide assurance that the minimum SDM of LCO 3.1.1 is maintained. Therefore, this change does not result in a significant reduction in a margin of safety.

ITS DISCUSSION OF DIFFERENCES
ITS Section 3.4A: Reactor Coolant System

Note: The ITS Section 3.4A package addresses the following NUREG-1430 RSTS:

NUREG 3.4.1	RCS Pressure, Temperature and Flow DNB Limits
NUREG 3.4.2	RCS Minimum Temperature for Criticality
NUREG 3.4.3	RCS P/T Limits
NUREG 3.4.4	RCS Loops – MODE 1 and 2
NUREG 3.4.5	RCS Loops – MODE 3
NUREG 3.4.6	RCS Loops – MODE 4
NUREG 3.4.7	RCS Loops – MODE 5, Loops Filled
NUREG 3.4.8	RCS Loops – MODE 5, Loops Not Filled

1 NUREG 3.4.1 & 3.4.4 - These Specifications are revised such that the DNB limits are included in the COLR (rather than LCO 3.4.1 and the associated SRs) for each pump combination operating condition; the LCO 3.4.4 THERMAL POWER limits are included in the COLR; The DNB limits and THERMAL POWER limits are currently controlled administratively, and since they are subject to change with fuel design changes, are proposed to be controlled in the COLR. The Bases are also revised to reflect these changes. Relocating these values to the Core Operating Limits Report has previously been approved by the NRC for the Oconee Nuclear Station in their ITS conversion.

3.4A-01

3.4A-08

The NRC approved the relocation of the Variable Low RCS Pressure-Temperature (VLPT) protective limits figure in Amendment 186, dated October 3, 1996. The thermal power limits and RCS flow rates for all allowable RCP operating conditions were included on this figure. In the Safety Evaluation, the staff stated that "*Although there have only been a few previous revisions to the ANO-1 VLPT setpoint, it is anticipated that an increasing number of future changes will be made in order to accommodate advanced core designs ... Therefore, due to these expected frequent changes to the VLPT setpoint for future cycles of ANO-1, the staff considers the VLPT setpoint an appropriate cycle-specific item.*" Although the staff was aware that these parameters would not change every cycle, they are considered to be cycle-specific. This same conclusion can be reached for the RCS loop pressure and hot leg temperature parameters. Although they will not change every cycle, there may be changes based on reload analyses and other needs, such as increasing the SG tube plugging limit. The Core Operating Limits Report is considered to be part of the ANO-1 SAR. Therefore, changes to the Core Operating Limits Report are evaluated in accordance with the requirements of 10 CFR 50.59. This change is consistent with the current license basis as these values are currently controlled outside of the CTS.

3.4A-02

NUREG 3.4.4 - The LCO 3.4.4 ACTIONS are revised to retain the capability to operate with only two RCPs operating, one in each loop, for up to 24 hours. This latter item is an evaluated condition which is discussed in BAW-10103A, Rev. 3, and determined to be acceptable for the identified Completion Time. This change is consistent with the current license basis.

ITS DISCUSSION OF DIFFERENCES

2 3.4A-03 3.4A-04 NUREG 3.4.1 - The LCO 3.4.1 Applicability Note is revised to omit the criteria of "THERMAL POWER step > 10% RTP" since mathematically any step change will be included in the criteria of "THERMAL POWER change > 5% RTP per minute." The term "ramp" is revised to "change" as "ramp" is not used at this station with this intended meaning. Also, the phrase "pressure transients due to" is incorporated to prevent noncompliance when the ramp is concluded but the associated pressure transient has not yet dampened out. This is necessary since the pressure transient that results from a power step change lags behind the power change. In other words, a pressure transient may still be in progress after the power change is complete. With the NUREG-1430 wording, this could result in entry into a Condition even though the pressure transient was clearly caused by the power transient. This Applicability Note now reads "RCS loop pressure limit does not apply during pressure transients due to a THERMAL POWER change > 5% RTP per minute." The Bases are also revised to reflect these changes.

3 3.4A-05 NUREG 3.4.1 - The SR 3.4.1.4 Note currently requires performance of the SR upon establishing stable conditions in the higher power range. The proposed change removes the ambiguity of "higher power range" by using a specific power level requirement. This measurement is accomplished by performing a calorimetric heat balance. Specifying that the SR is performed at ≥ 90% RTP provides a clear statement for performance of the SR and ensures that the unit is allowed to establish conditions that would provide for the most accurate heat balance calorimetric. The Bases are also revised to reflect these changes.

4 NUREG 3.4.2 - The Applicability of the Specification is revised to be consistent with Required Action A.1 and with the unit specific safety analysis. The control rod ejection analysis is evaluated for full power, zero power, and subcritical conditions. Further, in practice, the LCO and SR are most pertinent as the reactor is approaching criticality. Therefore, the Applicability is revised to include all of MODE 2. This change is not consistent with (but is more restrictive than) generic traveler TSTF-26 which would have revised the Required Action to match the NUREG Applicability. The Bases are also revised to reflect these changes. This change is consistent with current license basis.

In addition, the LCO Bases are revised to provide an improved discussion of why this requirement is necessary. This discussion is similar to the Bases provided in NUREG-1431 for RCS Minimum Temperature for Criticality for a Westinghouse reactor.

5 NUREG 3.4.5, 3.4.6, 3.4.7, & 3.4.8 - Incorporates TSTF-153 with the exception that the wording of each Note is revised from "may not be in operation" to "may be removed from operation." The TSTF-153 allowance more closely matches the requirement for the pump(s) to be "in operation." The proposed wording of "may be removed from operation" provides the intent that the pump is intentionally taken out of the "in operation" condition. An allowance that the pump "may not be in operation" would allow a pump trip to be considered as entry into the Note allowance, and not as entry into the applicable Condition. Use of the Note allowance under such a condition

would be inappropriate. The proposed wording is considered an administrative clarification of intent by wording preference and is not a change in the requirements.

3.4A-06 6 NUREG 3.4.2 - Incorporates TSTF-27, Rev. 3, with the following changes: The references to "loop" average temperature are revised to omit the reference to "loop." ANO-1 normally utilizes an average of the Tcold and Thot averages and not the individual loop Tavg. During 3 pump operation, the Tavg of the loop with the higher flow is monitored, but this is an infrequent operational condition. This change retains unit specific design application. A Discussion of how ANO calcuates Tavg has been added to the Bases for SR 3.4.2.1.

3.4A-18 7 NUREG 3.4.3 and 3.4.12 - The specific pressure/temperature limit curves and RCS heatup and cooldown rate limits are retained in the specification, i.e., not relocated to a Pressure Temperature Limits Report (PTLR). Specifically, this changes the LCO and the SRs to refer to the actual limit curves rather than a PTLR, and adds the limit curves as Figures. Also, a Note is added to LCO 3.4.3 to exclude the pressurizer from the pressure and temperature limits since the separate limits would normally be identified in the PTLR. However, the pressurizer limits are relocated from the CTS as identified in the "split report." The Bases are also revised to reflect these changes, as well as unit specific design, analyses, references, and programs. This change is consistent with current license basis.

3.4A-17 NUREG 3.4.3 Bases background - A statement has been added that the reactor vessel specimens are installed near the inside wall of this or a similar vessel in the core region. In 1976, it was found that surveillance capsule holder tubes in B&W fabricated 177FA plants were experiencing flow-induced vibration and were removed from the vessels (or were destroyed while they were in the vessel like at ANO-1). Three 177FA plants, Crystal River 3, Davis-Besse and TMI-2 were still under construction at that time; improved holder tubes were installed in those plants. They became the "host" plants for surveillance capsules for all the other 177 FA plants, The March 1979 incident at TMI-2 terminated the "host" status of TMI-2.

The program was enlarged by incorporating the material from the Westinghouse-designed, B&W fabricated reactor vessels to the RVIP. The program became known as the Master Integrated Reactor Vessel Surveillance Program (MIRVP). This program meets the requirements of Appendix H of 10CFR50, with regard to integrated surveillance programs (paragraph III.C) and is also an NRC accepted program.

The current in-vessel capsule withdrawal and testing schedule is in BAW-1543, Rev. 4, Supplement 3. The NRC accepted the technical basis for the program with regard to the design and operating conditions through their approval of BAW-1543, Rev. 4, including Supplements 1, 2, and 3 in the NRC SER dated October 26, 1999. This change is consistent with the current license basis.

3.4A-09 8 Not used.

3.4A-11 9 NUREG 3.4.5, 3.4.6, 3.4.7, and 3.4.8 – Incorporated Generic Traveler TSTF-263, Rev 3, except as noted in DOD-12.

10 NUREG 3.4.5, 3.4.6, 3.4.7, and 3.4.8 - NUREG SR 3.4.5.2, SR 3.4.6.2, SR 3.4.7.3 and SR 3.4.8.2 are revised to clarify that the surveillance is applicable to each required pump regardless of its operating status since both pumps may be operating. The Bases are also revised to indicate that if a pump is verified to be in operation, this is also sufficient to verify the correct breaker alignment and indicated power availability. The Bases are also revised to reflect these changes. This change also adds a Note allowing a 24 hour delay in performing the SR. This is consistent with Generic Traveler TSTF-265, Rev 2.

3.4A-15

11 NUREG 3.4.6 - The LCO Note which allows all RCPs and DHR pumps to be de-energized in MODE 4 is revised to be consistent with the CTS 3.1.1.6, Note *, allowances for pump de-energization. The reasons and frequency limits for the de-energization are not considered pertinent to the acceptability of the allowance and are removed. The LCO will continue to require a pump to be operating if conditions jeopardize cooling capability, or if adequate heat removal is not being provided. Therefore, as long as the conditions in the Note are met, a pump should not be required to be in operation. The Bases are also revised to reflect these changes and to clarify the conditions under which the pumps may be removed from operation. This change is consistent with current license basis.

12 NUREG 3.4.6 - The NUREG Conditions A and B are combined and simplified. The revised entry condition is based only on the status of the equipment which is required to be OPERABLE by the LCO, not on the status of all available equipment. An entry condition based on the status of equipment which is not required by the LCO is inconsistent with the remainder of the NUREG and with the Writer's Guide (NUMARC 93-03). The revised Required Actions also provide for clearer direction on when a shutdown to MODE 5 is required; specifically the Note clarifies that MODE 5 is only required if a DHR loop is OPERABLE. Also, the connector between NUREG Required Actions B.1 and B.2 is revised from <u>OR</u> to <u>AND</u> in ITS Condition A. The Bases clearly indicate that NUREG Required Action B.2 is required if restoration (per NUREG Required Action B.1) is not accomplished. With an <u>OR</u> connector, a choice is provided of either NUREG Required Action B.1 or B.2, but if NUREG Required Action B.1 is chosen and fulfilled, i.e., action to restore has been initiated, NUREG Required Action B.2 is not required. Since this is inconsistent with the intent (per the Bases) and with similar requirements in NUREG-1431 and NUREG-1432, the connector is revised to require both actions. The Bases are also revised to reflect these changes. This change is consistent with Generic Traveler TSTF-263, Rev 3.

13 NUREG 3.4.7 - The LCO Note which allows all DHR pumps to be de-energized is revised to be consistent with the CTS 3.1.1.6, Note *, allowances for pump de-energization. The frequency limits for the de-energization are not considered pertinent to the acceptability of the allowance and are removed. Considerable heat removal can be accomplished without a pump operating, and the LCO will continue to require a pump to be operating if conditions jeopardize natural circulation, or if adequate heat removal is not being provided. Therefore, as long as the conditions in the Note are met, a pump should not be required to be operating. The Bases are also revised to reflect these changes. This change is consistent with current license basis.

3.4A-11

14 Not used.

15 NUREG 3.4.8 - The LCO Note which allows all DHR pumps to be de-energized is revised to be consistent with the CTS 3.1.1.6, Note *, allowances for pump de-energization. The reduction in time period and purpose of the de-energization are not considered pertinent to the acceptability of the allowance and the CTS allowed time period is retained. The additional restriction on temperature is also not adopted. The maximum RCS temperature is adequately restricted by the MODE definitions and requirements for changing MODES. Therefore, as long as these requirements are met, and additional conditions in the Note are met, a pump should not be required to be in operation. The Bases are also revised to reflect these changes.

16 NUREG 3.4.5 and 3.4.6 - These LCOs are revised to omit "at least" since it does not affect the requirement. These LCOs provide the minimum acceptable condition and do not prohibit additional operating components. This change provides consistency with similar requirements in LCO 3.4.7 and LCO 3.4.8 which also provide for two OPERABLE loops and one in operation (without specifying "at least" one). This change is also consistent with the (NUMARC 93-03) Writers Guide for RSTS. This change is consistent with Generic Traveler TSTF-261.

3.4A-09

17 Not used.

3.4A-12

18 Not used.

19 NUREG 3.4.6, 3.4.7, and 3.4.8 and Bases – Clarification is added to acknowledge that a DHR loop may be considered OPERABLE during alignment and when aligned for low pressure injection if it is capable of being manually (locally or remotely) realigned to the DHR mode of operation and is not otherwise inoperable. This occurs because of the dual requirements of the components that comprise the low pressure injection/decay heat removal system. This recognition is similar to the NUREG LCO 3.5.3 Note which allows for LPI OPERABILITY if aligned for decay heat removal. Typically, if both decay heat removal loops are OPERABLE, one is maintained aligned for the LPI mode of operation since this is the quickest and easiest way to provide makeup to the reactor coolant system in the shutdown modes. This allowance is consistent the current interpretation of the ANO-1 license basis.

20 NUREG 3.4.1 Bases - The Bases are revised to reflect the unit specific analysis, design, and licensing basis. Additionally, the ACTIONS reference the "accident analysis bounds" is revised to specifically address the single limit which is of concern, and the SR Bases are revised to omit implications that the Frequency for SR 3.4.1.1 and SR 3.4.1.2 are somehow related to the Completion Times for Required Action A.1.
3.4A-16
The NUREG SR 3.4.1.4 Bases state that the RCS flow must be measured by performance of a precision calorimetric heat balance. The terminology of "precision calorimetric heat balance" is not used at ANO. Therefore, this method of measurement has been deleted from the Bases. These changes are consistent with current license basis.

21 NUREG 3.4.4 Bases - The Bases are revised to provide additional information on the minimum acceptable equipment to meet the requirements of the LCO, and to present this information in a method similar to the other Specification Bases. These changes are consistent with current license basis or editorial.

22 NUREG 3.4.8 Bases - The Bases are revised to omit the reference to an open equipment hatch providing a pathway to the outside following core damage due to a loss of decay heat removal. The Temporary Equipment Hatch Cover (TEHC) is in place most of the outage, where the time to closure is much less than the time to steam release. This also true for the time when the equipment hatch is not in place or is being replaced by the TEHC. Since the time to steam release and core damage is tracked and is always less than the time to close containment, the NUREG statement is not appropriate. This change is consistent with current license basis.

23 NUREG 3.4.5, 3.4.6, 3.4.7, and 3.4.8 Bases - The Bases are revised as necessary to remove any implications that natural circulation must be established to stop the pumps. CTS 3.1.1.6 and Note * currently provide the allowance to stop all pumps in MODES 4 and 5. MODE 3 does not have this in CTS, but the NUREG would allow it to be added. However, since the CTS does not require that natural circulation be established to stop the pumps, the Bases are revised to remove language which implies such actions would be taken. This change is consistent with current license basis.

24 Not used.

25 NUREG 3.4.8 – Condition B is revised to include an additional Required Action to "suspend all operations involving reduction in RCS water volume." This is consistent with the requirements for no reduction in water volume while intentionally removing both DHR pumps from operation as allowed by Note 1, part b. This change adds a requirement which is not included in either the CTS or NUREG-1430.

26 NUREG 3.4.3 Bases - The Bases are revised to omit the introduction of a new phrase "acceptance limits" which is not clear defined. For example, it is not clear if this refers to safety analysis acceptance criteria, or some other type of limits. Since the first part of the paragraph adequately describes the Applicable Safety Analysis, the confusing information is unnecessary and omitted. Further, the second paragraph of the Applicability Bases is also omitted as unnecessary cross reference type material which is generally not provided in ITS. These changes are consistent with current license basis or are editorial.

27 NUREG 3.4.7 and 3.4.8 Bases - The Bases for SR 3.4.7.2 and SR 3.4.8.1 are revised to omit reference to safety analysis assumptions. The Applicable Safety Analysis section indicates that there are no safety analyses performed with initial conditions of this operating MODE. Additionally, the statement is revised to be consistent with the Bases for other similar SRs, e.g., SR 3.4.5.1 and SR 3.4.6.1, conducted in MODES for which there are no safety analyses performed with initial conditions of that operating MODE. These changes are consistent with current license basis.

28 NUREG Bases - The Criterion statement at the conclusion of the Applicable Safety Analysis section was modified at each occurrence to refer to 10 CFR 50.36 instead of the NRC Policy Statement. This is an editorial change associated with the implementation of the 10 CFR 50.36 rule changes after NUREG-1430, Revision 1 was issued.

ANO-248

For ITS LCO 3.4.5, the 10 CFR 50.36 Criterion satisfied by the ITS LCO was modified to preserve consistency with the ANO-1 license basis. Specifically, Criterion 4 of 10 CFR 50.36 was cited as the basis for inclusion of these LCOs. This change is consistent with current license basis and 10 CFR 50.36.

For ITS LCOs, 3.4.6, 3.4.7 and 3.4.8, TSTF-367 was incorporated. TSTF-367 has been revised to reflect 10 CFR 50.36, as discussed above.

29 NUREG-1430 - 3.4.2 LCO Bases - Additional information has been added that clarifies the ANO treatment of instrument uncertainty for this parameter value. This change is consistent with the current license basis.

ANO-247

30. Incorporated TSTF-286, Rev 2.

3.4A-07 31. NUREG-1430 - 3.4.3 and Bases - A new Condition A has been inserted. NUREG Conditions A, B,and C have been re-lettered to ITS Condition B, C, and D. The new Condition A provides the Required Actions in the event the RCS pressure/temperature relationship is not within the limits of Figure 3.4.3-1 during the performance of PHYSICS TESTS with RCS temperature ≤ 525°F. This provides an additional restriction on operation not currently contained in the NUREG, and is consistent with the current license basis.

Appropriate changes have been made to the Bases to reflect these changes. The Bases Required Actions discussion has been revised to incorporate a discussion similar to that of NUREG-1430 LCO 3.4.2 Bases Actions A.1 discussion.

CTS

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.1 RCS Pressure, Temperature, and Flow Departure from Nucleate Boiling (DNB) Limits

NA

LCO 3.4.1 RCS DNB parameters ~~for~~ loop pressure, hot leg temperature, and RCS total flow rate shall be within the limits specified ~~below:~~ in the COLR

edit NA

~~a. With four reactor coolant pumps (RCPs) operating:~~

~~RCS loop pressure shall be ≥ [2061.6] psig, RCS hot leg temperature shall be ≤ [604.6]°F, and RCS total flow rate shall be ≥ [139.7 E6] lb/hr; and~~

~~b. With three RCPs operating:~~

~~RCS loop pressure shall be ≥ [2057.2] psig, RCS hot leg temperature shall be ≤ [604.6]°F, and RCS total flow rate shall be ≥ [104.4 E6] lb/hr.~~

1

APPLICABILITY: MODE 1. NA

---NOTE---

RCS loop pressure limit does not apply during:

pressure transients due to a

~~a.~~ THERMAL POWER ~~ramp~~ change > 5% RTP per minute~~; or~~.

~~b. THERMAL POWER step > 10% RTP.~~

NA

2

3.4A-03

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more RCS DNB parameters not within limits.	A.1 Restore RCS DNB parameter(s) to within limit.	2 hours
B. Required Action and associated Completion Time not met.	B.1 Be in MODE 2.	6 hours

NA

NA

CTS

3.4A-04

3.4A-05

SURVEILLANCE REQUIREMENTS

	SURVEILLANCE	FREQUENCY	
SR 3.4.1.1	-------------------NOTE------------------- With three RCPs operating, the limits are applied to the loop with two RCPs in operation. --		NA NA
	Verify RCS loop pressure ~~≥ [2061.6] psig with four RCPs operating or ≥ [2057.2] psig with three RCPs operating.~~ is within the limit specified in the COLR.	12 hours	NA 1
SR 3.4.1.2	-------------------NOTE------------------- With three RCPs operating, the limits are applied to the loop with two RCPs in operation. --		NA
	Verify RCS hot leg temperature ~~≤ [604.6]°F~~ is within the limit specified in the COLR.	12 hours	NA 1
SR 3.4.1.3	Verify RCS total flow ~~≥ [139.7 E6] lb/hr with four RCPs operating or ≥ [104.4 E6] lb/hr with three RCPS operating.~~ is within the limit specified in the COLR.	12 hours	NA 1
SR 3.4.1.4	-------------------NOTE------------------- Only required to be performed when stable thermal conditions are established ~~in the higher power range of MODE 1.~~ at ≥ 90% RTP. --		NA 3
	Verify RCS total flow rate is within the limit specified in the COLR by measurement.	[18] months	NA 1

CTS

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.2 RCS Minimum Temperature for Criticality

LCO 3.4.2 The ~~Each~~ RCS ~~loop~~ average temperature (T_{avg}) shall be ≥ 525°F. 6 3.1.3.1

APPLICABILITY: MODE 1~~,~~ and 2.
~~MODE 2 with k_{eff} ≥ 1.0.~~ 4 3.1.3

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. T_{avg} ~~in one or more RCS loops~~ not within limit.	A.1 Be in MODE 3.	30 minutes

3.1.3.7 6

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
SR 3.4.2.1 Verify RCS T_{avg} ~~in each loop~~ ≥ 525°F.	~~-----NOTE----- Only required if any RCS loop T_{avg} < 530°F~~ ~~30 minutes thereafter~~ 12 hours

NA 6

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.3 RCS Pressure and Temperature (P/T) Limits

specified in Figures 3.4.3-1, 3.4.3-2, and 3.4.3-3.

CTS

LCO 3.4.3 RCS pressure, RCS temperature, and RCS heatup and cooldown rates shall be maintained within ~~the~~ limits ~~specified in the PTLR.~~ 3.1.2.1 3.1.2.3 3.1.3.2

Note
Not applicable to the pressurizer 7

APPLICABILITY: At all times. 3.1.2.3 NA

ACTIONS

<INSERT 3.4-4A>

CONDITION	REQUIRED ACTION	COMPLETION TIME	CTS
B ~~A~~ ---------NOTE--------- Required Action B ~~A~~.2 shall be completed whenever this Condition is entered. --------------------- Requirements of LCO not met in MODE 1, 2, 3, or 4.	B ~~A~~.1 Restore parameter(s) to within limits.	30 minutes	31 NA 3.1.2.6 3.1.3.7
	AND		
	B ~~A~~.2 Determine RCS is acceptable for continued operation.	72 hours	3.1.2.6 NA
C ~~B~~ Required Action and associated Completion Time of Condition B ~~A~~ not met.	C ~~B~~.1 Be in MODE 3.	6 hours	3.1.2.6 3.1.3.7
	AND		
	C ~~B~~.2 Be in MODE 5.	36 hours	3.1.2.6 NA
D ~~C~~ ---------NOTE--------- Required Action D ~~C~~.2 shall be completed whenever this Condition is entered. --------------------- Requirements of LCO not met in other than MODE 1, 2, 3, or 4.	D ~~C~~.1 Initiate action to restore parameter(s) to within limit.	Immediately	NA
	AND		
	D ~~C~~.2 Determine RCS is acceptable for continued operation.	Prior to entering MODE 4	NA

3.4A-07

<INSERT 3.4-4A>

3.4A-07

A. RCS Pressure and Temperature not within criticality limit of Figure 3.4.3-1 during PHYSICS TESTS with RCS temperature ≤ 525°F.	A.1 Be in MODE 3.	30 minutes	CTS 3.1.3.7

CTS

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY	CTS
SR 3.4.3.1	-------------------NOTE------------------- Only required to be performed during RCS heatup ~~and cooldown~~ operations ~~and RCS inservice leak and hydrostatic testing.~~ with fuel in the reactor vessel.		3.1.2.3 (7)
	Verify RCS pressure, RCS temperature, and RCS heatup ~~and cooldown~~ rates are within the limits specified in ~~the PTLR.~~ Figure 3.4.3-1.	30 minutes	3.1.2.1 3.1.2.3 (7)

INSERT 3.4-5A (7)

INSERT 3.4-5B, 3.4-5C, & 3.4-5D (7)

Reviewer's Note: These inserts are revised versions of CTS Figures 3.1.2-2, 3.1.2-3, and 3.1.2-1. They are, respectively, ITS Figures 3.4.3-1, 3.4.3-2, and 3.4.3-3.

<INSERT 3.4-5A>

			CTS
SR 3.4.3.2	------------------NOTE------------------ Only required to be performed during RCS cooldown operations with fuel in the reactor vessel. --		3.1.2.3
	Verify RCS pressure, RCS temperature, and RCS cooldown rates are within the limits specified in Figure 3.4.3-2.	30 minutes	3.1.2.1 3.1.2.3
SR 3.4.3.3	------------------NOTE------------------ Only required to be performed during RCS heatup and cooldown operations with no fuel in the reactor vessel. --		3.1.2.1
	Verify RCS pressure, RCS temperature, and RCS heatup and cooldown rates are within the limits specified in Figure 3.4.3-3.	30 minutes	3.1.2.1
SR 3.4.3.4	------------------NOTE------------------ Only required to be performed during PHYSICS TESTS with RCS temperature ≤ 525°F. --		3.1.2.3 3.1.3.2 3.1.8.3
	Verify RCS pressure and RCS temperature are within the criticality limits specified in Figure 3.4.3-1.	30 minutes	3.1.2.3 3.1.3.2 3.1.8.3

INSERT 3.4-5B

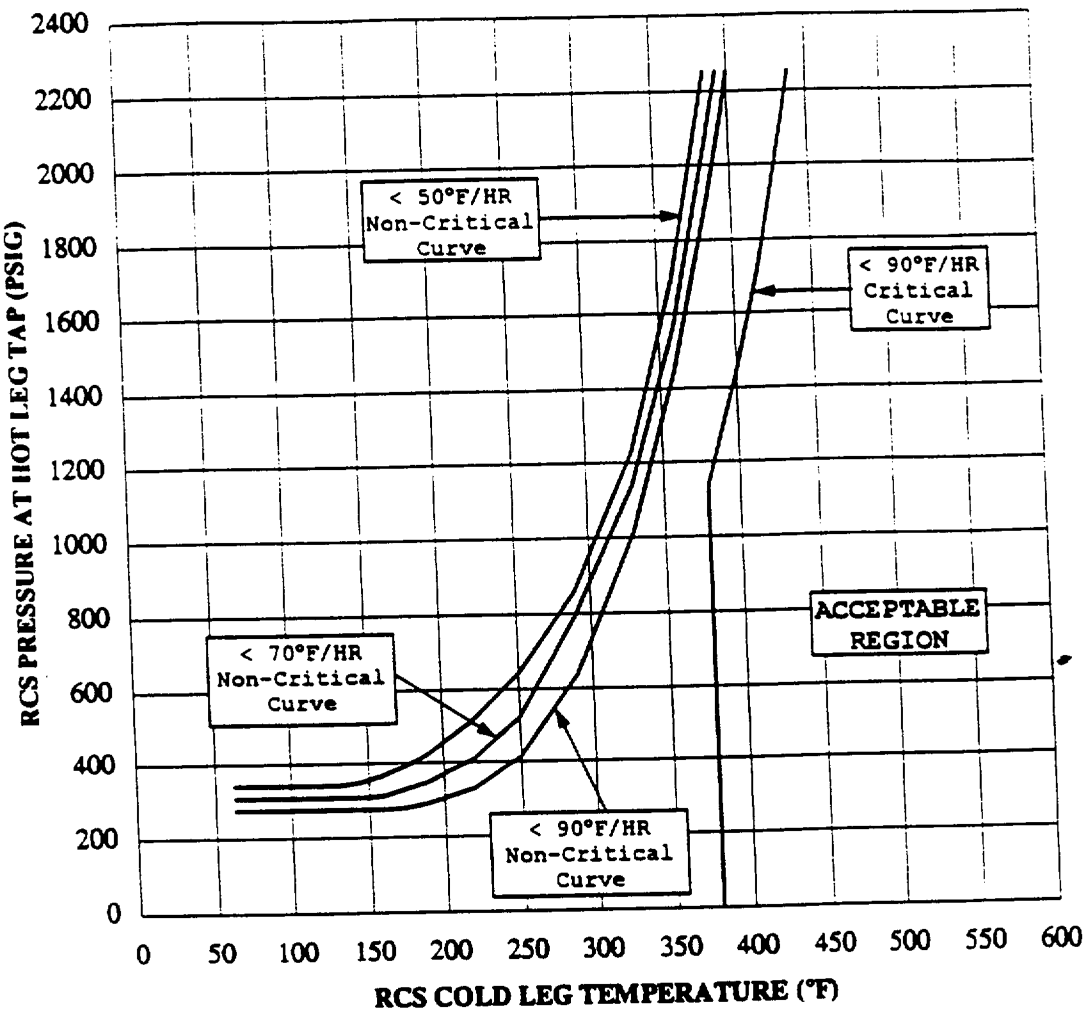
FIGURE 3.4.3-1 (page 1 of 1)

CTS

Fig. 3.1.2-2

~~FIGURE 3.1.2-2~~

RCS HEATUP LIMITATIONS TO 31 EFPY



Notes:

1. These curves are not adjusted for instrument error and shall not be used for operation.
2. When DHR is in operation with no RCPs operating, the DHR system return temperature shall be used.
3. RCP Operating Restrictions:

RCS TEMP	RCP RESTRICTIONS
T > 300°F	None
300°F ≥ T ≥ 225°F	≤ 3
225°F > T ≥ 84°F	≤ 2
T < 84°F	No RCPs operating

4. Allowable Heatup Rates:

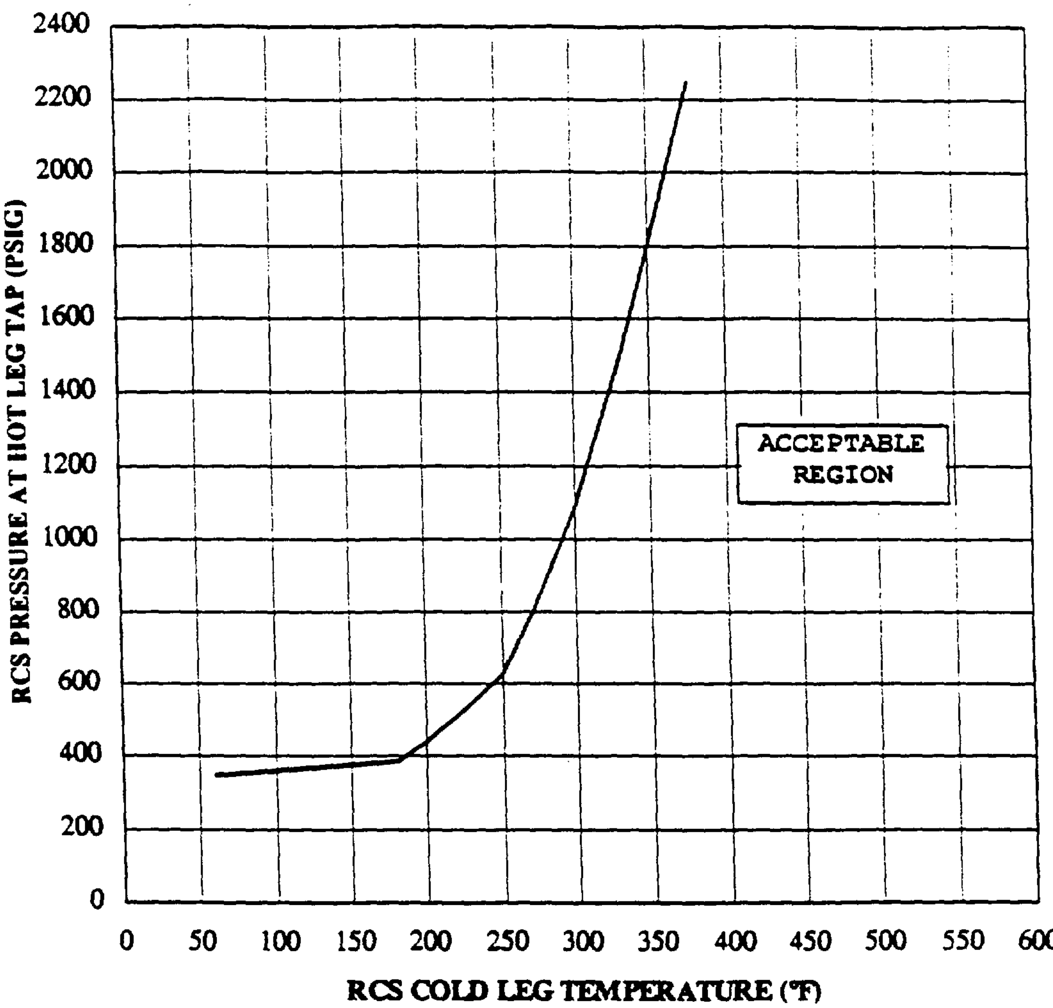
RCS TEMP	H/U RATE
60°F < T ≤ 84°F	≤ 15°F/HR
T > 84°F	As allowed by applicable curve

7

INSERT 34-5C

CTS
Fig. 3.1.2-3

FIGURE ~~3.1.2-3~~ 3.4.3-2 (page 1 of 1)
RCS COOLDOWN LIMITS TO 31 EFPY



Notes:

1. This curve is not adjusted for instrument error and shall not be used for operation.
2. A maximum step temperature change of 25°F is allowable when securing all RCPs with the DHR system in operation. This change is defined as the RCS temperature prior to securing all the RCPs minus the DHR return temperature after the RCPs are secured. When DHR is in operation with no RCPs operating, the DHR system return temperature shall be used.
3. RCP Operating Restrictions:

RCS TEMP	RCP RESTRICTIONS
T > 255°F	None
150°F ≤ T ≤ 255°F	≤ 2 (See Note 5)
T < 150°F	No RCPs operating

4. Allowable Cooldown Rates:

RCS TEMP	C/D RATE	STEP CHANGE
T ≥ 280°F	100°F/HR	≤ 50°F in any 1/2 HR
280°F > T ≥ 150°F	50°F/HR (See Note 5)	≤ 25°F in any 1/2 HR
T < 150°F	25°F/HR	≤ 25°F in any 1 HR

5. If RCPs are operated < 200°F, then the RCS cooldown rate from 150°F ≤ T ≤ 180°F is reduced to 30°F in 15 hours.

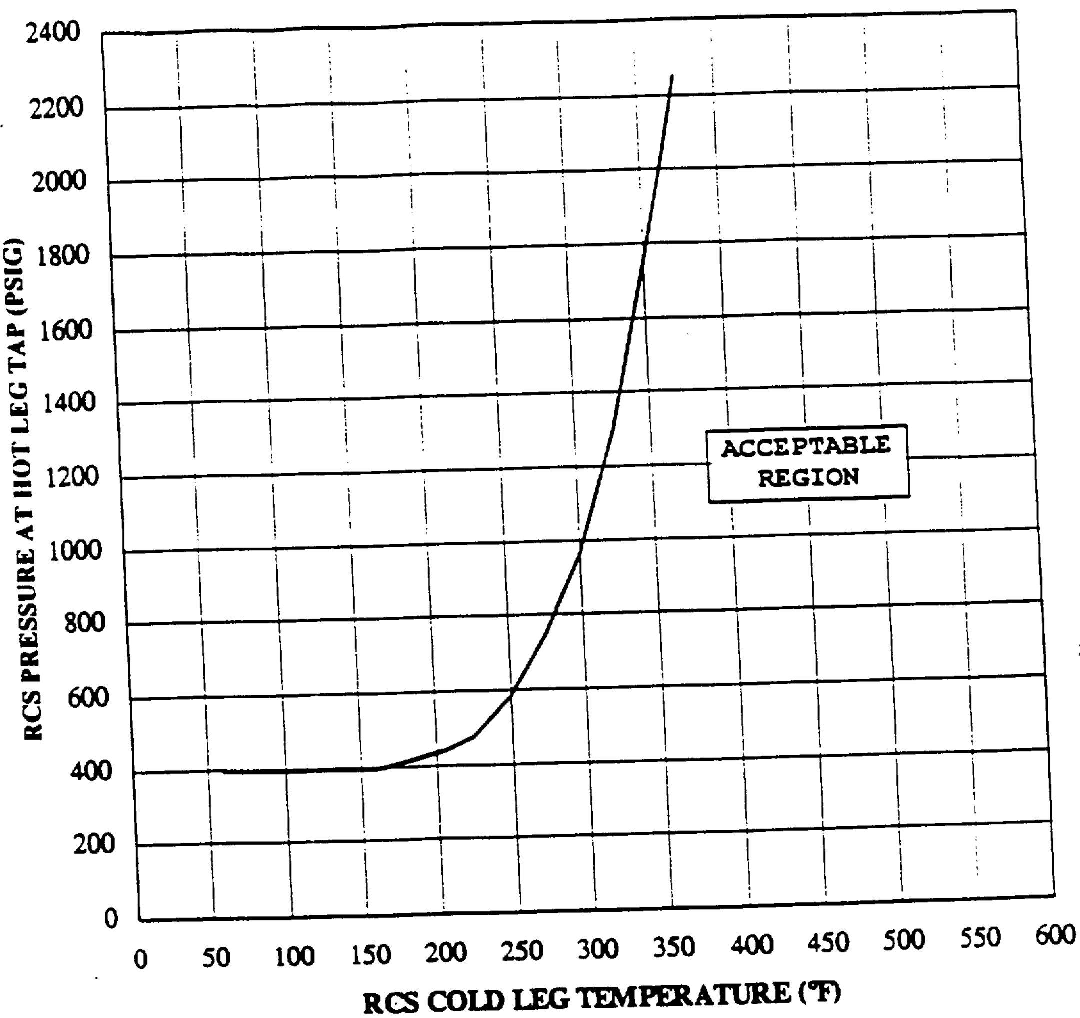
7

INSERT 3.4-5D

CTS
Fig. 3.1.2-1

3.4.3-3
FIGURE ~~3.1.2-1~~ (Page 1 of 1)

RCS INSERVICE HYDROSTATIC TEST H/U & C/D LIMITS TO 31 EFPY



Notes:

1. This curve is not adjusted for instrument error and shall not be used for operation.

2. All Notes on Figure ~~3.1.2-2~~ 3.4.3-1 are applicable for heatups. This curve is based on a heatup rate of < 90°F/HR.

3. All Notes on Figure ~~3.1.2-2~~ 3.4.3-2 are applicable for cooldowns.

7

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.4 RCS Loops—MODES 1 and 2

CTS

LCO 3.4.4 Two RCS Loops shall be in operation, with: 3.1.1.1.A 3.1.1.2.A 3.1.1.5.A 3.4.1.1 Table 2.3-1

a. Four reactor coolant pumps (RCPs) operating; or

b. Three RCPs operating and THERMAL POWER restricted ~~to [78.9]% RTP.~~ as specified in the COLR.

APPLICABILITY: MODES 1 and 2. 3.1.1.1.A 3.1.1.2.A 3.1.1.5.A 3.4.1

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
~~A. Requirements of LCO not met.~~	~~A.1 Be in MODE 3.~~	~~6 hours~~

1

<INSERT 3.4-6A>

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.4.4.1	Verify required RCS loops are in operation.	12 hours

NA

<INSERT 3.4-6A>

			CTS
A. One RCP not in operation in each loop.	A.1 Restore one non-operating RCP to operation.	18 hours	Table 2.3-1 & Note (d) 3.1.1.1.A
B. Required Action and associated Completion Time of Condition A not met. OR LCO not met for reasons other than Condition A.	B.1 Be in MODE 3.	6 hours	Table 2.3-1 & Note (d) 3.1.1.1.A 3.4.2

CTS

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.5 RCS Loops—MODE 3

16

edit

OPERABLE

LCO 3.4.5 Two RCS loops shall be OPERABLE and ~~at least~~ one RCS loop shall be in operation. 3.1.1.2.A 3.1.1.5.A 3.1.1.5.B 3.4.1.1

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removed from operation

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

All reactor coolant pumps (RCPs) may be ~~de-energized~~ for ≤ 8 hours per 24 hour period for the transition to or from the Decay Heat Removal System, and all RCPs may be ~~de-energized~~ for ≤ 1 hour per 8 hour period for any other reason, provided: 5

Introduction into the RCS, coolant with boron concentration less than required to meet the SDM of LCO 3.1.1

a. No operations are permitted that would cause ~~reduction of the RCS boron concentration~~; and 3.1.1.1.B 30

3.4A-10

b. Core outlet temperature is maintained at least ~~[10]~~ 9 °F below saturation temperature. N/A

APPLICABILITY: MODE 3. 3.1.1.2.A 3.1.1.5.A 3.4.1

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One ~~required~~ RCS loop inoperable.	A.1 Restore ~~required~~ RCS loop to OPERABLE status.	72 hours
B. Required Action and associated Completion Time of Condition A not met.	B.1 Be in MODE 4.	12 hours

3.1.1.5.A 3.4.2 9 (Condition A)

3.1.1.5.A 3.4.2 (Condition B)

(continued)

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CTS

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME	CTS
C. ~~No~~ Two RCS loops ~~OPERABLE~~ inoperable, OR ~~No~~ Required RCS loop not in operation.	C.1 ~~Suspend all operations involving a reduction of RCS boron concentration.~~ Suspend operations that would cause introduction into the RCS coolant with boron concentration less than required to meet SDM of LCO 3.1.1.	Immediately	9 3.1.1.1.B 3.1.1.5.B 30
	AND		
	C.2 Initiate action to restore one RCS loop to OPERABLE status and operation.	Immediately	9 3.1.1.5.B

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY	CTS
SR 3.4.5.1 Verify required RCS loop is in operation.	12 hours	4.27.4
SR 3.4.5.2 Verify correct breaker alignment and indicated power available to ~~the~~ each required pumps ~~that is not in operation.~~	7 days	4.27.1 10

INSERT 3.4-8A

<INSERT 3.4-8A>

-----------------------------NOTE-----------------------------
Not required to be performed until 24 hours after a required pump is not in operation.

CTS

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.6 RCS Loops—MODE 4

LCO 3.4.6 Two loops consisting of any combination of RCS loops and decay heat removal (DHR) loops shall be OPERABLE and ~~at least~~ one OPERABLE loop shall be in operation. 16 3.1.1.6

---NOTE---

All reactor coolant pumps (RCPs) ~~may be de-energized for ≤ 8 hours per 24 hour period for the transition to or from the DHR System, and all RCPs and DHR pumps may be de-energized~~ removed from operation for ≤ 1 hour ~~per 8 hour period for any other reason~~, provided: 11 3.1.1.6 Note * edit 5

a. ~~No operations are permitted that would cause reduction of the RCS boron concentration;~~ No operations are permitted that would cause introduction into the RCS, coolant with boron concentration less than required to meet the SDM of LCO 3.1.1 and 30 edit (¢ 3.1.1.1.B) NA

b. Core outlet temperature is maintained at ~~least~~ less than or equal to a temperature which is 10°F below saturation temperature.

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APPLICABILITY: MODE 4. 3.1.1.6

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One required ~~RCS~~ loop inoperable. ~~AND~~ ~~Two DHR loops inoperable.~~	A.1 Initiate action to restore a second loop to OPERABLE status. AND A.2 ---NOTE--- Only required if DHR loop is OPERABLE Be in MODE 5.	Immediately 24 hours

3.1.1.6.A

3.1.1.6.A

(continued) 12

CTS

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME	
~~B. One required DHR loop inoperable. AND Two required RCS loops inoperable.~~	~~B.1 Initiate action to restore a second loop to OPERABLE status.~~ OR ~~B.2 Be in MODE 5.~~	~~Immediately~~ ~~24 hours~~	12
B ~~C.~~ Two Required ~~RCS or DHR~~ loops inoperable. OR Required ~~No RCS or DHR~~ loop not in operation.	B ~~C~~.1 ~~Suspend all operations involving a reduction in RCS boron concentration.~~ Suspend operations that would cause introduction into the RCS, coolant with boron concentration less than required to meet SDM of LCO 3.1.1. AND B ~~C~~.2 Initiate action to restore one loop to OPERABLE status and operation.	Immediately Immediately	30 3.1.1.1.B 3.1.1.6.B 12 9 3.1.1.6.A 3.1.1.6.B

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SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY	
SR 3.4.6.1	Verify required ~~one~~ DHR or RCS loop is in operation.	12 hours	9 4.27.4 4.27.5
SR 3.4.6.2 INSERT 3.4-10A	Verify correct breaker alignment and indicated power available to ~~the~~ each required pump ~~that is not in operation.~~	7 days	4.27.1 10

<INSERT 3.4-10A>

-------------------------NOTE-------------------------
Not required to be performed until 24 hours after a required pump is not in operation.

CTS

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.7 RCS Loops—MODE 5, Loops Filled

3.4A-12

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LCO 3.4.7 One decay heat removal (DHR) loop shall be OPERABLE and in operation, and either: 3.1.1.6

a. One additional DHR loop shall be OPERABLE; or

b. The secondary side water level of each steam generator (SG) shall be ≥ ~~[50]%~~ 20 inches 4.27.3

13

-----NOTES-----

removed from operation

1. The DHR pump of the loop in operation may be ~~de-energized~~ for ≤ 1 hour ~~per 8 hour period~~ provided: 3.1.1.6

5 Note *

(& 3.1.1.1.B)

No operations are permitted that would cause introduction into the RCS, coolant with boron concentration less than required to meet the SDM of LCO 3.1.1

a. ~~No operations are permitted that would cause reduction of the RCS boron concentration~~; and

30

edit

b. Core outlet temperature is maintained at ~~least~~ 10°F below saturation temperature.

less than or equal to a temperature which is

2. One required DHR loop may be inoperable for ~~up to~~ ≤ 2 hours for surveillance testing provided that the other DHR loop is OPERABLE and in operation. edit NA

3. All DHR loops may be removed from operation during planned heatup to MODE 4 when at least one RCS loop is in operation. NA

APPLICABILITY: MODE 5 with RCS loops filled. 3.1.1.6

CTS

ACTIONS

	CONDITION	REQUIRED ACTION		COMPLETION TIME	CTS
3.4A-11 3.4A-12 <INSERT 3.4-12A>	~~A. One DHR loop inoperable. AND Any SG with secondary side water level not within limits.~~	~~A.1~~	~~Initiate action to restore a second DHR loop to OPERABLE status.~~	~~Immediately~~	3.1.1.6.A 9
		~~OR A.2~~	~~Initiate action to restore SG secondary side water levels to within limits.~~	~~Immediately~~	3.1.1.6.A
3.4A-11 AND-247	C ~~B~~. No Required DHR loop ~~inoperable~~. OR Required ~~No~~ DHR loop not in operation. Suspend operations that would cause introduction into the RCS, coolant with boron concentration less than required to meet SDM of LCO 3.1.1.	~~B~~ C.1	~~Suspend all operations involving a reduction in RCS boron concentration.~~	Immediately	9 3.1.1.1.B 3.1.1.6.B 30
		AND ~~B~~ C.2	Initiate action to restore one DHR loop to OPERABLE status and operation.	Immediately	9 3.1.1.6.A 3.1.1.6.B

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY	CTS
SR 3.4.7.1	Verify ~~one~~ required DHR loop is in operation.	12 hours	9 4.27.5
SR 3.4.7.2	Verify required SG secondary side water levels are ≥ ~~[50]%~~ 20 inches	12 hours	3.4A-12 4.27.3

(continued)

3.4A-11

<INSERT 3.4-12A>

A. One required DHR loop inoperable. AND One DHR loop OPERABLE.	A.1	Initiate action to restore a second DHR loop to OPERABLE status.	Immediately
	OR		
	A.2	Initiate action to restore required SG(s) secondary side water level to within limit.	Immediately
B. One or more required SGs with secondary side water level not within limit AND One DHR loop OPERABLE.	B.1	Initiate action to restore a second DHR loop to OPERABLE status.	Immediately
	OR		
	B.2	Initiate action to restore required SGs secondary side water level to within limit.	Immediately

CTS

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.4.7.3	Verify correct breaker alignment and indicated power available to ~~the~~ each required DHR pump ~~that is not in operation~~.	7 days

<INSERT 3.4-13A>

NA

10

<INSERT 3.4-13A>

-----------------------------NOTE-----------------------------
Not required to be performed until 24 hours after a required pump is not in operation.

CTS

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.8 RCS Loops—MODE 5, Loops Not Filled

edit

LCO 3.4.8 Two decay heat removal (DHR) loops shall be OPERABLE and one OPERABLE DHR loop shall be in operation. 3.1.1.6

NOTES

1. All DHR pumps may be ~~de-energized~~ removed from operation for ≤ ~~15 minutes when switching from one loop to another~~ 1 hour provided: 5 15 3.1.1.6 Note *

   ~~[a. The maximum RCS temperature is ≤ [160]°F;]~~

   a ~~b~~. ~~No operations are permitted that would cause a reduction of the RCS boron concentration; and~~ No operations are permitted that would cause introduction into the RCS, coolant with boron concentration less than required to meet the SDM of LCO 3.1.1 30 (* 3.1.1.1.B) NA

   AO-247

   b ~~c~~. No draining operations to further reduce the RCS water volume are permitted.

2. One DHR loop may be inoperable for ≤ 2 hours for surveillance testing provided that the other DHR loop is OPERABLE and in operation. NA

APPLICABILITY: MODE 5 with RCS loops not filled. 3.1.1.6

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One required DHR loop inoperable.	A.1 Initiate action to restore DHR loop to OPERABLE status.	Immediately

9 3.1.1.6.A

(continued)

CTS

3.4A-11

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME	
B. ~~Required~~ No Required DHR loop~~s~~ ~~inoperable~~ operable. OR ~~No~~ Required DHR loop not in operation.	B.1 ~~Suspend all operations involving reduction in RCS boron concentration.~~ Suspend operations that would cause introduction into the RCS, coolant with boron concentration less than required to meet SDM of LCO 3.1.1. AND	Immediately	30 3.1.1.1.B 3.1.1.6.B 9
	B.~~2~~3 Initiate action to restore one DHR loop to OPERABLE status and operation.	Immediately	3.1.1.6.A 3.1.1.6.B
AND-247	AND B.2 Suspend all operations involving reduction in RCS water volume	Immediately	25

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY	
SR 3.4.8.1	Verify ~~one~~ required DHR loop is in operation.	12 hours	9 4.27.5
SR 3.4.8.2 INSERT 3.4-15A	Verify correct breaker alignment and indicated power available to ~~the~~ each required DHR pump~~that is not in operation~~.	7 days	NA 10

<INSERT 3.4-15A>

-----------------------------NOTE-----------------------------
Not required to be performed until 24 hours after a required pump is not in operation.

B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.1 RCS Pressure, Temperature, and Flow Departure from Nucleate Boiling (DNB) Limits

BASES

BACKGROUND

These Bases address requirements for maintaining RCS pressure, temperature, and flow rate within limits assumed in the safety analyses. The safety analyses (Ref. 1) of normal operating conditions and ~~anticipated operational occurrences~~ abnormalities assume initial conditions within the normal steady state envelope. The limits placed on DNB related parameters ensure that these parameters will not be less conservative than were assumed in the analyses and thereby provide assurance that the minimum departure from nucleate boiling ratio (DNBR) will meet the required criteria for each of the transients analyzed. edit

Considering only pressure, a

The LCO for minimum RCS pressure is consistent with ~~operation within the nominal operating envelope and is above~~ that used as the initial pressure in the analyses. A pressure greater than the minimum specified will produce a higher ~~minimum~~ DNBR~~. A~~ ; and a pressure lower than the minimum specified will ~~cause the plant to approach the DNB limit.~~ produce a lower DNBR. edit

Considering only temperature, a

The LCO for maximum RCS coolant hot leg temperature is consistent with ~~full power operation within the nominal operating envelope and is lower than~~ the initial hot leg temperature in the analyses. A hot leg temperature lower than that specified will produce a higher ~~minimum~~ DNBR~~. A~~ ; and a temperature higher than that specified will ~~cause the plant to approach the DNB limit.~~ produce a lower DNBR.

The RCS flow rate is not expected to vary during operation with all pumps running. The LCO for the minimum RCS flow rate corresponds to that assumed for the DNBR analyses. Considering only flow rate, a ~~A~~ higher RCS flow rate than that specified will produce a higher DNBR~~. A~~ , and a lower RCS flow will ~~cause the plant to approach the DNB limit.~~ produce a lower DNBR.

APPLICABLE SAFETY ANALYSES

The requirements of LCO 3.4.1 represent the initial conditions for DNB limited transients analyzed in the plant safety analyses (Refs 1 and 2). The safety analyses have shown that transients initiated from the limits of this LCO will meet the DNBR ~~criterion of ≥ [1.3]. This is~~ INSERT B3.4-1A the acceptance edit 20

(continued)

<INSERT B3.4-1A>

criteria of ≥ 1.30 or ≥ 1.18, for the BAW-2 or the BWC critical heat flux correlation, respectively. For the locked rotor accident, the minimum DNB ratio is not less than applicable critical heat flux correlation limit, or fuel cladding is shown to experience no significant temperature excursions. These are

BASES

APPLICABLE SAFETY ANALYSES (continued)

criteria ~~limit~~ for the RCS DNBR parameters. Changes to the facility that could impact these parameters must be assessed for their impact on the DNBR criterion. The transients analyzed ~~for~~ include loss of coolant flow events and dropped or stuck control rod events. A key assumption for the analysis of these events is that the core power distribution is within the limits of LCO 3.2.1, "Regulating Rod Insertion Limits," LCO 3.2.3, "AXIAL POWER IMBALANCE OPERATING LIMITS," ~~and~~ LCO 3.2.4, "QUADRANT POWER TILT RATIO (QPTR)." edit edit lower case edit

~~The core outlet pressure assumed in the safety analyses is 2135 psia. The minimum pressure specified in LCO 3.4.1 is the limit value in the reactor coolant loop as measured at the hot leg pressure tap.~~

INSERT B 3.4-2A

~~The safety analyses are performed with an assumed RCS coolant average temperature of 581°F (579°F plus 2°F allowance for calculational uncertainty). The corresponding hot leg temperature of 604.6°F is calculated by assuming an RCS core outlet pressure of 2135 psia and an RCS flow rate of 374,880 gpm. The maximum temperature specified is the limit value at the hot leg resistance temperature detector.~~ 20

~~The safety analyses are performed with an assumed RCS flow rate of 374,880 gpm. The minimum flow rate specified in LCO 3.4.1 is the minimum mass flow rate.~~ 1

Analyses have been performed to establish the two pump, pressure, temperature, and flow rate requirements for three pump, and four pump operation. The flow limits for two pump and three pump operation are substantially lower than for four pump operation. To meet the DNBR criterion, a corresponding maximum power limit is required (see Bases for LCO 3.4.4, "RCS Loops—MODES 1 and 2"). 1 1 20

INSERT B3.4-2B from page B 3.4-3

The RCS DNB limits satisfy Criterion 2 of ~~the NRC Policy Statement.~~ 10 CFR 50.36 (Ref. 4). 28

LCO

This LCO specifies limits on the monitored process variables: RCS loop (hot leg) pressure, RCS hot leg temperature, and RCS total flow rate to ensure that the core operates within the limits assumed for the plant safety analyses. Operating within these limits will result in

(continued)

<INSERT B3.4-2A>

LCO 3.1.4, "CONTROL ROD Group Alignment Limits," and LCO 3.1.6, "AXIAL POWER SHAPING ROD (APSR) Alignment Limits."

The safety analyses to establish reload operating limits are performed using nominal values for RCS coolant average temperature, core outlet pressure, and RCS flow rate and core power level with appropriate application of associated uncertainty. Consistent with Statistical Core Design (SCD) methodology, applicable random parametric uncertainties are combined statistically. As necessary, bias parametric uncertainties are included deterministically. The RCS temperature and pressure are measured in the hot leg. The surveillance criteria specified in the COLR include adjustment for measurement location. The COLR specified hot leg temperature is the maximum allowed so that the analysis value is not exceeded. The COLR specified hot leg pressure and flow are the minimum allowed so that the analysis values are not exceeded.

BASES

LCO (continued)

meeting DNBR criteria in the event of a DNB limited transient.

The pressure and temperature limits are to be applied to the loop with two reactor coolant pumps (RCPs) running for the three RCPs operating condition.

surveillance criteria

consistent with supporting analysis.

The ~~LCO numerical values~~ for pressure, temperature, and flow rate ~~are given~~ for the measurement location ~~but have not been adjusted~~ for instrument error. ~~Plant specific limits of instrument error are established by the plant staff to meet the operational requirements of this LCO.~~

as specified in the COLR have been appropriately adjusted

and

1

APPLICABILITY

In MODE 1, the limits on RCS pressure, RCS hot leg temperature, and RCS flow rate must be maintained during steady state ~~with four pump or three pump~~ operation in order to ensure that DNBR criteria will be met in the event of an unplanned loss of forced coolant flow or other DNB limited transient. In all other MODES the power level is low enough so that DNB is not a concern.

Significant

1

The Note indicates the limit on RCS pressure may be exceeded during short term operational transients ~~such as~~ a THERMAL POWER ~~ramp increase~~ > 5% RTP per minute, ~~or a THERMAL POWER step increase > 10% RTP.~~ These conditions represent short term perturbations where actions to control pressure variations might be counterproductive. Also, ~~since they represent~~ transients initiated from power levels ~~≤ 100% RTP,~~ increased DNBR margin exists to offset the temporary pressure variations.

pressure

resulting from

change

for

less than the ALLOWABLE THERMAL POWER,

on plant operations

2

~~Another set of~~ limits on DNBR related parameters ~~is~~ are provided in Safety Limit (SL) 2.1.1, "Reactor Core SLs." Those limits are less restrictive than the limits of LCO 3.4.1, but violation of an SL merits a stricter, more severe Required Action. Should a violation of LCO 3.4.1 occur, ~~the operator must~~ check whether an SL may have been exceeded.

The steady state

a

must be performed to determine

edit

Move to Page B3.4-2 as INSERT B3.4-2B

20

3.4A-03

ACTIONS

A.1

Loop pressure and hot leg coolant temperature are controllable and measurable parameters. With one or both of

(continued)

BASES

ACTIONS

A.1 (continued)

these parameters not within the LCO limits, action must be taken to restore the parameters. RCS flow rate is not a controllable parameter and is not expected to vary during steady state ~~four pump or three pump~~ operation. However, if the flow rate is below the LCO limit, the parameter must be restored to within limits or power must be reduced as required in Required Action B.1, to ~~restore DNBR margin and~~ eliminate the potential for violation of the ~~accident analysis bounds.~~ minimum DNBR limit. ① 20

The 2 hour Completion Time for restoration of the parameters provides sufficient time to adjust plant parameters, determine the cause for the off normal condition, and restore the readings within limits. The Completion Time is based on plant operating experience.

B.1

If the Required Action ~~A.1 is not met within the~~ and associated Completion Time~~;~~ are not met, the plant must be brought to a MODE in which the LCO does not apply. To achieve this status, the plant must be brought to at least MODE 2 within 6 hours. In MODE 2, the reduced power condition eliminates the potential for violation of the accident analysis ~~bounds.~~ assumptions. edit edit

The 6 hour Completion Time is reasonable, based on operating experience, to ~~reduce power~~ reach MODE 2 from full power conditions in an orderly manner ~~in conjunction with even control of steam generator heat removal.~~ and without challenging safety systems. edit

SURVEILLANCE REQUIREMENTS

SR 3.4.1.1

~~Since Required Action A.1 allows a Completion Time of 2 hours to restore parameters that are not within limits the 12 hour Surveillance Frequency for loop (hot leg) pressure is sufficient to ensure that the pressure can be restored to a normal operation, steady state condition following load changes and other expected transient operations.~~ 20 The RCS pressure value specified is dependent on the number of pumps in operation and has been adjusted to account for the pressure ~~loss~~ difference between the core edit

(continued)

3.4A-04

BASES

SURVEILLANCE REQUIREMENTS

SR 3.4.1.1 (continued)

exit and the measurement location. ~~The value used in the plant safety analysis is 2135 psia.~~ The 12 hour interval has been shown by operating practice to be sufficient to regularly assess potential degradation and to verify operation is within safety analysis assumptions. 1-20

A Note has been added to indicate the pressure limits are to be applied to the loop with two pumps in operation for the three pump operating condition.

SR 3.4.1.2

~~Since Required Action A.1 allows a Completion Time of 2 hours to restore parameters that are not within limits, the 12 hour Surveillance Frequency for hot leg temperature is sufficient to ensure that the RCS coolant temperature can be restored to a normal operation, steady state condition following load changes and other expected transient operations.~~ The 12 hour interval has been shown by operating practice to be sufficient to regularly assess potential degradation and to verify that operation is within safety analysis assumptions. 20

A Note has been added to indicate the temperature limits are to be applied to the loop with two pumps in operation for the three pump operating condition.

SR 3.4.1.3

available indications

The 12 hour Surveillance Frequency for RCS total flow rate is performed using the ~~installed~~ flow ~~instrumentation~~. The 12 hour interval has been shown by operating practice to be sufficient to regularly assess potential degradation and to verify that operation is within safety analysis assumptions. 20

SR 3.4.1.4

Measurement of RCS total flow rate ~~by performance of a precision calorimetric heat balance~~ once every [18] months allows the installed RCS flow instrumentation to be 20

(continued)

3.4A-05

BASES

SURVEILLANCE REQUIREMENTS

SR 3.4.1.4 (continued)

calibrated and verifies that the actual RCS flow is greater than or equal to the minimum required RCS flow rate specified in the COLR. (1)

The Frequency of ~~[~~18~~]~~ months reflects the importance of verifying flow after a refueling outage when the core has been altered or RCS flow characteristics may have been modified, which may have caused change of flow.

The Surveillance is modified by a Note that indicates the SR does not need to be performed until stable thermal conditions are established at higher power levels (i.e., $\geq$ 90% RTP). (3) The Note ~~is necessary to allow~~ provides for measurement of the flow rate at normal operating conditions at power in MODE 1. The Surveillance ~~cannot~~ may be performed at low power or in MODE 2 or below. ~~because at low power the ΔT across the core will be too small to provide valid results.~~ (20)

<INSERT B3.4-6A>

REFERENCES

1. ~~U~~SAR, Chapter ~~[15].~~ 14. edit
2. SAR, Section 3A.6.
3. BAW-10179P-A, dated February 1996.
4. 10 CFR 50.36. (28)

<INSERT B3.4-6A>

However, at low or zero power condition, the indications are less accurate and significant penalties for uncertainties may be necessary. Performance of the calorimetric heat balance at a high power level and normal operation conditions provides for the most accurate flow verification.

B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.2 RCS Minimum Temperature for Criticality

BASES

BACKGROUND

Establishing the value for the minimum temperature for reactor criticality is based upon considerations for:

a. Operation within the existing instrumentation ranges ~~and accuracies~~; edit

b. Operation with reactor vessel above its minimum nil ductility reference temperature when the reactor is critical.

The reactor coolant (average) moderator temperature coefficient used in core operating and accident analysis is typically defined for the normal operating temperature range (532°F to 579°F). The Reactor Protection System (RPS) receives inputs from the narrow range hot leg temperature detectors, which have a range of 520°F to 620°F. The integrated control system controls average temperature (T_{avg}) using inputs of the same range. Nominal T_{avg} for making the reactor critical is 532°F. Safety and operating analyses for lower temperatures have not been ~~made.~~ performed for all possible scenarios. edit edit

APPLICABLE SAFETY ANALYSES

There are no accident analyses that dictate the minimum temperature for criticality, but all low power safety analyses assume initial temperatures near the 525°F limit (Ref. 1).

The RCS minimum temperature for criticality satisfies Criterion 2 of ~~the NRC Policy Statement.~~ 10 CFR 50.36 (Ref. 2). 28

LCO

INSERT B3.4-7A

~~The purpose of the LCO is to prevent criticality outside the normal operating regime (532°F to 579°F) and to prevent operation in an unanalyzed condition.~~ 4

The LCO limit of 525°F has been selected to be within the instrument indicating range (520°F to 620°F). ~~The limit is also set slightly below the lowest power range operating temperature (532°F).~~ 4 This parameter value is considered to be a nominal value, No additional allowances for instrument uncertainty are required in the implementing procedures. 29

(continued)

<INSERT B3.4-7A>

Compliance with the LCO ensures that the reactor will not be made or maintained critical at a temperature significantly less than the hot zero power (HZP) temperature, which is assumed in the safety analysis. Failure to meet the requirements of this LCO may produce initial conditions inconsistent with the initial conditions assumed in the safety analysis.

BASES (continued)

APPLICABILITY

with $T_{avg} \geq 525°F$.

The reactor has been designed and analyzed to be critical in MODES 1 and 2 only ~~and in accordance with this Specification.~~ Criticality is not permitted in any other MODE. Therefore, this LCO is applicable in MODE 1 and MODE 2. ~~when $k_{eff} \geq 1.0$.~~

edit

4

ACTIONS

A.1

With T_{avg} below 525°F, 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 in 30 minutes. Rapid reactor shutdown can be readily and practically achieved in a 30 minute period. The Completion Time reflects the ability to perform this Action and maintain the plant within the analyzed range. If T_{avg} can be restored within the 30 minute time period, shutdown is not required.

SURVEILLANCE REQUIREMENTS

SR 3.4.2.1

INSERT B3.4-8A

~~T_{avg} is required to be verified above 525°F every 30 minutes. The 30 minute time period is frequent enough to prevent inadvertent violation of the LCO. The 30 minute portion of the Frequency has been modified by a Note indicating this SR is only required when T_{avg} < 530°F. While Surveillance is required whenever the reactor is critical and temperature is below 530°F, in practice the Surveillance is most appropriate during the period when the reactor is brought critical.~~

6

REFERENCES

1. ~~F~~SAR, Chapter ~~[15]~~. 14.
2. 10 CFR 50.36.

edit

28

<INSERT B3.4-8A>

RCS average temperature is required to be verified at or above 525°F every 12 hours. The SR to verify RCS average temperature every 12 hours takes into account indications that are continuously available to the operator in the control room and is consistent with other routine surveillances which are typically performed once per shift. In addition, Operators are trained to be sensitive to RCS temperature during approach to criticality and will ensure that the minimum temperature for criticality is met as criticality is approached.

3.4A-06

RCS T_{avg} is normally calculated as the average of the unit T_{hot} (hot temperature average of loops A and B) and the unit T_{cold} (cold temperature average of loops A and B). During operation with 3 RCPs in operation, T_{avg} is calculated as the average of the loop T_{hot} and loop T_{cold} in the loop that has 2 RCPs running.

B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.3 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 unit ~~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. edit

INSERT B 3.4-9A

~~The PTLR contains P/T limit curves for heatup, cooldown, and inservice leak and hydrostatic (ISLH) testing, and data for the maximum rate of change of reactor coolant temperature (Ref. 1).~~ 7

~~Each P/T limit curve defines an acceptable region for normal operation. The usual use of the curves is operational~~ guidance for use 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~~. edit

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 due to the lost neutron embrittlement it experiences during power operation, and the LCO limits apply mainly to the vessel. The limits do not apply to the pressurizer, which has different design characteristics and operating functions. edit

10 CFR 50, Appendix G (Ref. 2), requires the establishment of P/T limits for material fracture toughness requirements of the ~~RCPB~~ materials. Reference 2 requires an adequate margin to brittle failure during normal operation, abnormalities ~~anticipated operational occurrences~~, and system hydrostatic tests. It mandates the use of the American Society of Mechanical Engineers (ASME), Boiler and Pressure Vessel Code, Section III, Appendix G (Ref. 3). edit

Linear elastic fracture mechanics (LEFM) methodology is used to determine the stresses and material toughness at locations within the RCPB. The LEFM methodology follows the

(continued)

<INSERT B3.4-9A>

Figures 3.4.3-1, 3.4.3-2, and 3.4.3-3 contain P/T limit curves for heatup, cooldown, inservice hydrostatic testing, and physics testing at RCS temperatures ≤ 525°F, and the maximum rate of change of reactor coolant temperature. The methods and criteria employed to establish operating pressure and temperature limits are described in BAW-10046A (Ref. 1). These limit curves are applicable through thirty-one effective full power years (EFPY) of operation. The pressure limit is adjusted for the pressure differential between the point of system pressure measurement and the limiting component for the various operating reactor coolant pump combinations.

Each P/T curve defines an acceptable region for normal operation below and to the right of the limit curve. The curves are used to develop operational

BASES

BACKGROUND (continued)

guidance given by 10 CFR 50, Appendix G; ASME Code, Section III, Appendix G; and Regulatory Guide 1.99 (Ref. 4).

INSERT B 3.4-10A

~~Material toughness properties of the ferritic materials of the reactor vessel are determined in accordance with the NRC Standard Review Plan (Ref. 5), ASTM E 185 (Ref. 6), and additional reactor vessel requirements. These properties are then evaluated in accordance with Reference 3.~~ 7

edit

beltline region

surveillance

The actual shift in the ~~nil ductility reference temperature~~ ~~(~~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. 6) and~~ Appendix H of 10 CFR 50 (Ref. ~~7~~ 9). 7

INSERT B3.4-10B

The operating P/T limit curves will be adjusted, as necessary, based on the evaluation findings and the recommendations of Reference 3. 7

INSERT B3.4-10C

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

The heatup curve represents a different set of restrictions than the cooldown curve because the directions of the thermal gradients through the vessel wall are reversed. The thermal gradient reversal alters the location of the tensile stress between the outer and inner walls.

inservice hydrostatic

edit

The calculation to generate the ~~ISLH~~ testing curve uses different safety factors (per Ref. 3) than the heatup and cooldown curves. The ~~ISLH~~ testing curve also extends to the RCS design pressure of 2500 psia.

The P/T limit curves and associated temperature rate of change limits are developed in conjunction with stress analyses for large numbers of operating cycles and provide conservative margins to nonductile failure. Although created to provide limits for these specific normal operations, the curves also can be used to determine if an evaluation is necessary for an abnormal transient.

(continued)

<INSERT B3.4-10A>

The major components of the reactor coolant pressure boundary have been analyzed in accordance with Appendix G to 10CFR50. Results of this analysis, including the actual pressure-temperature limitations of the reactor coolant pressure boundary, are given in FTI Document 77-1258569-01 (Ref. 5). The service period was reduced by one effective full power year from that assumed in Reference 5 to be conservative with respect to independent calculations performed by the NRC staff. The limiting weld material is being irradiated as part of the B&W Owners Group Integrated Reactor Vessel Material Surveillance Program and the identification and locations of the capsules containing the limiting weld material is discussed in the latest revision to B&W report, BAW-1543 (Ref. 6). The chemical composition of the limiting weld material is reported in the B&W report, BAW-2121P (Rev. 7). The effect of neutron irradiation on the nil-ductility reference temperature (RT_{NDT}) of the limiting weld material is reported in FTI Calculations 32-1245917-00 and 32-1257716-00 (Rev. 8).

<INSERT B3.4-10B>

These specimens are installed near the inside wall of this or a similar reactor vessel in the core region.

<INSERT B3.4-10C>

Prior to reaching thirty-one effective full power years of operation, Figures 3.4.3-1, 3.4.3-2, and 3.4.3-3 must be updated for the next service period in accordance with 10 CFR 50, Appendix G. The service period must be of sufficient duration to permit the scheduled evaluation of a portion of the surveillance data scheduled in accordance with the latest revision of Topical Report BAW-1543 (Ref. 6). The highest predicted adjusted reference temperature of all the beltline region materials is used to determine the adjusted reference temperature at the end of the service period. The basis for this prediction is submitted for NRC staff review at least 90 days prior to the end of the service period.

BASES

BACKGROUND (continued)

The consequence of violating the LCO limits is that the RCS has been operated under conditions that can result in brittle failure of the RCPB, possibly leading to a nonisolable leak or loss of coolant accident. In the event these limits are exceeded, an evaluation must be performed to determine the effect on the structural integrity of the RCPB components. The ASME Code, Section XI, Appendix E (Ref. ~~8~~ 10) provides a recommended methodology for evaluating an operating event that causes an excursion outside the limits. edit

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, an unanalyzed condition. Reference 1 establishes the methodology for determining the P/T limits. ~~Since the P/T limits are not derived from any DBA analysis, 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.~~ 26

RCS P/T limits satisfy Criterion 2 of ~~the NRC Policy Statement~~ 10 CFR 50.36 (Ref. 11). 28

LCO

The ~~two~~ three elements of this LCO are:

a. The limit curves for heatup, cooldown, normal operation, PHYSICS TESTING, and ~~ISLH~~ inservice hydrostatic testing; ~~and~~ edit

b. Limits on the rate of change of temperature~~.~~; and

c. Limits on RCP combinations

The LCO limits (as indicated by the Note) apply to all components of the RCS, except the pressurizer. These limits define allowable operating regions and permit a large number of operating cycles while providing a wide margin to nonductile failure. 7

The limits for the rate of change of temperature control the thermal gradient through the vessel wall and are used as inputs for calculating the ~~heatup, cooldown, and ISLH~~ P/T limit curves. Thus, the LCO for the rate of change of edit

(continued)

BASES

LCO (continued)

temperature restricts stresses caused by thermal gradients and also ensures the validity of the P/T limit curves. 7

3.4A-18 INSERT B 3.4-12A

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

edit

a. The ~~severity~~ magnitude of the departure from the allowable operating P/T regime or the ~~severity~~ magnitude of the rate of change of temperature;

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

c. The existences, sizes, and orientations of flaws in the vessel material.

APPLICABILITY

The RCS P/T limits Specification provides a definition of acceptable operation for prevention of nonductile failure in accordance with 10 CFR 50, Appendix G (Ref. 2). Although the P/T limits were developed to provide guidance for operation during heatup or cooldown (MODES 3, 4, and 5) or ~~ISLH~~ inservice hydrostatic testing, their applicability is at all times in keeping with the concern for nonductile failure. The limits do not apply to the pressurizer. edit

~~During MODES 1 and 2, other Technical Specifications provide limits for operation that can be more restrictive than or can supplement these P/T limits. LCO 3.4.1, "RCS Pressure, Temperature, and Flow Departure from Nucleate Boiling (DNB) Limits"; LCO 3.4.2, "RCS Minimum Temperature for Criticality"; and Safety Limit (SL) 2.1, "SLs," also provide operational restrictions for pressure and temperature and maximum pressure. MODES 1 and 2 are above the temperature range of concern for nonductile failure, and stress analyses have been performed for normal maneuvering profiles, such as power ascension or descent.~~ 26

(continued)

<INSERT B3.4-12A>

The limit curves include the limiting pressure differential between the point of system pressure measurement and the pressure on the reactor vessel region controlling the limit curve. However, the limit curves are not adjusted for possible instrument error and should not be used for operation.

The heatup and cooldown rates stated are intended as the maximum changes in temperature in one direction in the stated time periods. The actual temperature linear ramp rate may exceed the stated limits for a shorter time period provided that the maximum total temperature difference does not exceed the limit and that a temperature hold is observed to prevent the total temperature difference from exceeding the limit for the stated time period.

The acceptable P/T combinations are below and to the right of the limit curves which are applicable for the first 31 EFPY. The limit curves include the limiting pressure differential between the point of system pressure measurement and the pressure on the reactor vessel region controlling the limit curve. However, the limit curves are not adjusted for possible instrument error and should not be used for operation.

BASES (continued)

3.4A-07

ACTIONS <INSERT B3.4-13A>

A.1 and A.2

31

Operation outside the P/T limits during MODE 1, 2, 3, or 4 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.

3.4A-07

beyond the 72 hour Completion Time of Required Action B.2.

Besides restoring operation to within limits, an evaluation is required to determine if RCS operation can continue. The evaluation must verify the RCPB integrity remains acceptable and must be completed before continuing operation. Several methods may be used, including comparison with pre-analyzed transients in the stress analyses, new analyses, or inspection of the components. The evaluation must be completed, documented, and approved in accordance with established ~~plant~~ unit procedures and administrative controls. edit

edit

10

ASME Code, Section XI, Appendix E (Ref. 8) may be used to support the evaluation. However, its use is restricted to evaluation of the vessel beltline. The evaluation must extend to all components of the RCPB. edit

The 72 hour Completion Time is reasonable to accomplish the evaluation. The evaluation for a mild violation is possible within this time, but more severe violations may require special, event specific stress analyses or inspections. A favorable evaluation must be completed before continuing to operate beyond the 72 hour Completion Time. edit

3.4A-07

31

Condition ~~A~~ B is modified by a Note requiring Required Action ~~A~~ B.2 to 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~~ B.1 is insufficient because higher than analyzed stresses may have occurred and may have affected the RCPB integrity.

(continued)

<INSERT B3.4-13A>

3.4A-07

With RCS pressure and temperature not within criticality limit of Figure 3.4.3-1 during PHYSICS TESTS with RCS temperature ≤ 525°F, 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 in 30 minutes. Rapid reactor shutdown can be readily and practically achieved in a 30 minute period. The Completion Time reflects the ability to perform this Action and maintain the plant within the analyzed range. If RCS pressure and temperature can be restored within the 30 minute time period, shutdown is not required.

B.1 and B.2

BASES

3.4A-07

ACTIONS (continued)

~~B~~ C.1 and ~~B~~ C.2 H31

If a Required Action and associated Completion Time of Condition ~~A~~ B are not met, the ~~plant~~ unit must be brought to a lower MODE because: (a) the RCS remained in an unacceptable pressure and temperature region for an extended period of increased stress, or (b) a sufficiently severe event caused entry into an unacceptable region. Either possibility indicates a need for more careful examination of the event. Performing this examination in the required lower MODES reduces ~~best accomplished with~~ the RCS ~~at reduced~~ pressure and temperature~~. With reduced pressure and temperature conditions,~~ which decreases the possibility of propagation of undetected flaws~~ is decreased~~. edit H31 edit

3.4A-07

If the required restoration activity cannot be accomplished within 30 minutes, Required Action ~~B~~ C.1 and Required Action ~~B~~ C.2 must be ~~implemented~~ initiated to reduce pressure and temperature. H31 edit

3.4A-07

If the required evaluation for continued operation cannot be accomplished within 72 hours, or the results are indeterminate or unfavorable, action must proceed to reduce pressure and temperature as specified in Required Actions ~~B~~ C.1 and ~~B~~ C.2. A favorable evaluation must be completed and documented before returning to operating pressure and temperature conditions. However, if the favorable evaluation is accomplished while reducing pressure and temperature conditions, a return to power operation may be ~~considered~~ initiated without completing these Required Actions ~~B.2~~. H31 edit

Pressure and temperature are reduced by bringing the ~~plant~~ unit to MODE 3 within 6 hours and to MODE 5 within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required MODE from full power conditions in an orderly manner and without challenging ~~plant~~ unit systems. edit edit

3.4A-07

~~C~~ D.1 and ~~C~~ D.2 H31

Actions must be initiated immediately to correct operation outside of the P/T limits at times other than MODE 1, 2, 3, or 4, so that the RCPB is returned to a condition that has been verified acceptable by stress analysis.

(continued)

BASES

3.4A-07

ACTIONS D D ~~C~~.1 and ~~C~~.2 (continued) 31

The immediate Completion Time reflects the urgency of initiating action to restore the parameters to within the analyzed range. Most violations will not be severe, and the activity can be accomplished ~~within this time~~ promptly in a controlled manner. edit

3.4A-19

In addition to restoring operation to within limits, an evaluation is required to ~~determine if RCS operation can continue. The evaluation must~~ verify that the RCPB integrity remains acceptable. ~~and~~ The evaluation must be completed prior to entry into MODE 4. Several methods may be used, including comparison with pre-analyzed transients in the stress analysis, or inspection of the components. edit

ASME Code, Section XI, Appendix E (Ref. ~~8~~ 10), may also be used to support the evaluation. However, its use is restricted to evaluation of the vessel beltline. edit

3.4A-07

Condition ~~C~~ D is modified by a Note requiring Required Action ~~C~~.2 to 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 D ~~C~~.1, is insufficient because higher than analyzed stresses may have occurred and may have affected RCPB integrity. 31

SURVEILLANCE REQUIREMENTS

SR 3.4.3.1, SR 3.4.3.2, SR 3.4.3.3, and SR 3.4.3.4 7

Verification that operation is within the ~~PTLR~~ limits of the appropriate figure is required every 30 minutes when RCS pressure and temperature conditions are undergoing planned changes.

This Frequency is considered reasonable in view of the control room indication available to monitor RCS status. Also, since temperature rate of change limits are specified in hourly increments, 30 minutes permits assessment and correction for minor deviations within a reasonable time.

Surveillance for heatup, cooldown, or ~~ISLH~~ inservice hydrostatic testing may be discontinued when the definition given in the relevant ~~plant~~ unit procedure for ending the activity is satisfied. edit edit

(continued)

BASES

SURVEILLANCE REQUIREMENTS

INSERT B3.4-16A

~~SR 3.4.3.1 (continued)~~

~~This SR is modified by a Note that requires this SR to be performed only during system heatup, cooldown, and ISLH testing.~~

"Methods of Compliance with Fracture Toughness and Operational Requirements of 10CFR50, Appendix G."

7

edit

REFERENCES

1. BAW-10046A, Rev. ~~1~~ 2, ~~July 1977.~~ June 1986.
2. 10 CFR 50, Appendix G.
3. ASME, Boiler and Pressure Vessel Code, Section III, Appendix G.
4. Regulatory Guide 1.99, Revision 2, May 1988.
5. ~~NUREG-0800, Section 5.3.1, Rev. 1, July 1981.~~
6. ~~ASTM E 185-82, July 1982.~~

9 ~~7~~. 10 CFR 50, Appendix H.

10 ~~8~~. ASME, Boiler and Pressure Vessel Code, Section XI, Appendix E.

11. 10 CFR 50.36

7

28

3.4A-18

5. FTI Document 77-1258569-01.
6. BAW-1543, Integrated Reactor Vessel Material Surveillance Program (latest revision).
7. BAW-2121P, Irradiation Induced Reduction in Charpy Upper Shelf Energy of Reactor Vessel Welds.
8. FTI Calculations 32-1245917-00 and 32-1257716-00.

<INSERT B3.4-16A>

SR 3.4.3.1, SR 3.4.3.2, and SR 3.4.3.3 (continued)

The acceptable P/T combinations are below and to the right of the limit curves which are applicable for the first 31 EFPYs. The limit curves include the limiting pressure differential between the point of system pressure measurement and the pressure on the reactor vessel region controlling the limit curve. However, the limit curves are not adjusted for possible instrument error and should not be used for operation (as identified in Note 1 on each applicable Figure).

SR 3.4.3.1 is modified by a Note that requires this SR to be performed only during system heatup operations with fuel in the reactor vessel. This SR refers to Figure 3.4.3-1 which provides applicable heatup limitations, including reactor coolant pump (RCP) operating restrictions and allowable heatup rates. Figure 3.4.3-1 Note 2 identifies that when the decay heat removal system is operating with no RCPs operating, the indicated DHR system return temperature to the reactor vessel is the appropriate temperature indicator.

SR 3.4.3.2 is modified by a Note that requires this SR to be performed only during system cooldown operations with fuel in the reactor vessel. This SR refers to Figure 3.4.3-2 which provides applicable cooldown limitations, including reactor coolant pump (RCP) operating restrictions and allowable cooldown rates. During system cooldown operations with fuel in the reactor vessel, the RCPs are eventually removed from service. Figure 3.4.3-2 Note 2 identifies that when the decay heat removal system is operating with no RCPs operating, the indicated decay heat removal system return temperature to the reactor vessel is the appropriate temperature indicator. Figure 3.4.3-2 Note 2 also indicates that a maximum step temperature change of 25°F is allowable when removing all RCPs from operation with the decay heat removal system operating. The step temperature change is defined as the reactor coolant temperature (prior to stopping all RCPs) minus the decay heat removal (DHR) system return temperature to the reactor vessel (after stopping all RCPs). The step change of 25°F is applicable only during transition from RCP operation to DHR. This step change must be included when determining the cooldown rate.

SR 3.4.3.3 is modified by a Note that requires this SR to be performed only during system heatup and cooldown operations with no fuel in the reactor vessel. This SR refers to Figure 3.4.3-2 which provides applicable heatup and cooldown limitations, including reactor coolant pump (RCP) operating restrictions and allowable heatup and cooldown rates. These curves are used during inservice hydrostatic testing that is performed in a defueled condition. The Notes on Figure 3.4.3-1 and Figure 3.4.3-2 are applicable to heatups and cooldowns performed within these limits.

SR 3.4.3.4 is modified by a Note that requires this SR to be performed only during PHYSICS TESTS with the average RCS temperature ≤ 525°F. This SR refers to Figure 3.4.3-1 which provides applicable limitations under which the unit may be critical, including Reactor Coolant Pump (RCP) operating restrictions and allowable heatup rates. This curve is used during PHYSICS TESTING. This is because LCO 3.4.2, "RCS Minimum Temperature for Criticality," normally limits the temperature for criticality to well above this curve. However, an exception to LCO 3.4.2 is provided by LCO 3.1.9, "PHYSICS TEST Exceptions - MODE 2," during PHYSICS TESTS initiated in MODE 2.

When the decay heat removal (DHR) system is operating with no RCPs operating, the indicated DHR system return temperature to the reactor vessel is the appropriate temperature indicator.

B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.4 RCS Loops—MODES 1 and 2

BASES

BACKGROUND

The primary function of the RCS is removal of the heat generated in the fuel due to the fission process, and transfer of this heat, via the steam generators (SGs), to the secondary plant. edit

reactor coolant

The secondary functions of the RCS include: edit

a. Moderating the neutron energy level to the thermal state, to increase the probability of fission;

b. Improving the neutron economy by acting as a reflector;

c. Carrying the soluble neutron poison, boric acid;

d. Providing a second barrier against fission product release to the environment; and

e. Removing the heat generated in the fuel due to fission product decay following a unit shutdown.

The RCS configuration for heat transport uses two RCS loops. Each RCS loop contains an SG and two reactor coolant pumps (RCPs). An RCP is located in each of the two SG cold legs. The pump flow rate has been sized to provide core heat removal with appropriate margin to departure from nucleate boiling (DNB) during power operation and for anticipated transients originating from power operation. This Specification requires two RCS loops with either three or four pumps to be in operation. With three pumps in operation the reactor power level is restricted to ~~[79.9]% RTP~~ to preserve the core power to flow relationship, thus maintaining the margin to DNB. The intent of the Specification is to require core heat removal with forced flow during power operation. Specifying the minimum number of pumps is an effective technique for designating the proper forced flow rate for heat transport, and specifying two loops provides for the needed amount of heat removal capability for the allowed power levels. Specifying two RCS loops also provides the minimum necessary paths (two SGs) for heat removal.

only two or

21

a nominal 49% RTP or 75% RTP, respectively,

(continued)

BASES (RCS Flow and Measured AXIAL POWER IMBALANCE)

BACKGROUND (continued)

The Reactor Protection System (RPS) ~~n~~uclear ~~o~~verpower trip setpoint is automatically reduced when ~~one~~ pump is taken out of service; manual resetting is not necessary. (a) edit edit

APPLICABLE SAFETY ANALYSES

Safety analyses (Ref. 1) ~~contain~~ various assumptions for the Design Bases Accident (DBA) initial conditions including: RCS pressure, RCS temperature, reactor power level, core parameters, and safety system setpoints. The important aspect for this LCO is the reactor coolant forced flow rate, which is represented by the number of pumps in service. edit

INSERT B 3.4-18A

~~Both transient and steady state analyses have been performed to establish the effect of flow on DNB. The transient or accident analysis for the plant has been performed assuming either three or four pumps are in operation. The majority of the plant safety analysis is based on initial conditions at high core power or zero power. The accident analyses that are of most importance to RCP operation are the four pump coastdown, single pump locked rotor, and single pump (broken shaft or coastdown) (Ref. 1).~~ 21

Steady state DNB analysis has been performed for four, three, and two pump combinations. For four pump operation, the steady state DNB analysis, which generates the pressure and temperature SL (i.e., the departure from nucleate boiling ratio (DNBR) limit), assumes a maximum power level of [112]% RTP. This is the design overpower condition for four pump operation. The [112]% value is the accident analysis ~~setpoint~~ limit of the nuclear overpower (high flux) trip and is based on an analysis assumption that bounds possible instrumentation errors. The DNBR limit defines a locus of pressure and temperature points that result in a minimum DNBR ~~greater than or equal to~~ that protects the critical heat flux correlation limit. edit edit

The three pump pressure temperature limit is tied to the steady state DNB analysis, which is evaluated each cycle. The flow used is the minimum allowed for three pump operation. The actual RCS flow rate will exceed the assumed flow rate. With three pumps operating, overpower protection is automatically provided by the ~~power to flow ratio of the~~ RPS nuclear overpower ~~based on~~ RCS flow and measured AXIAL POWER IMBALANCE ~~setpoint~~ Function. The maximum power level for three pump 21

(continued)

<INSERT B3.4-18A>

3.4A-20

Both transient and steady state analyses have been performed to establish the effect of RCS flow on DNB. The initial condition DNB protection for the limiting loss of coolant flow event for four, three, and two pump operation is provided by the RCS flow surveillance criteria specified in the COLR for SR 3.4.1.3 and SR 3.4.1.4. The loss of coolant flow event which has been found to produce the limiting DNB is the four-to-two pump coastdown. In addition to the coastdown events, the single pump locked rotor event has been analyzed and shows that either the minimum DNB ratio is not less than the applicable critical heat flux correlation limit, or fuel cladding was shown to experience no significant temperature excursions.

BASES

APPLICABLE SAFETY ANALYSES (continued)

operation is ~~[79.9]% RTP~~ identified in the COLR and is based on the three pump flow as a fraction of the four pump flow at full power. edit

INSERT B3.4-19A

~~Although the Specification limits operation to a minimum of three pumps total, existing design analyses show that operation with one pump in each loop (two pumps total) is acceptable when core THERMAL POWER is restricted to be proportionate to the flow. However, continued power operation with two RCPs removed from service is not allowed by this Specification.~~ 21

RCS Loops—MODES 1 and 2 satisfy Criterion 2 of ~~the NRC Policy Statement~~ 10 CFR 50.36 (Ref. 3). 28

LCO

The purpose of this LCO is to require adequate forced flow for core heat removal. ~~Flow is represented by the number of RCPs in operation in both RCS loops for removal of heat by the two SGs.~~ INSERT B3.4-19B 21 To meet safety analysis acceptance criteria for DNB, four pumps are required at rated power; if ~~only three~~ fewer pumps are available, power must be reduced as specified in the COLR. 1

APPLICABILITY

In MODES 1 and 2, the reactor ~~is~~ may be critical and has the potential to produce maximum THERMAL POWER. To ensure that the assumptions of the accident analyses remain valid, all RCS loops are required to be OPERABLE and in operation in these MODES to prevent DNB and core damage. edit

The decay heat production rate is much lower than the full power heat rate. As such, the forced circulation flow and heat sink requirements are reduced for lower, noncritical MODES as indicated by the LCOs for MODES 3, 4, and 5.

Operation in other MODES is covered by:

LCO 3.4.5, "RCS Loops—MODE 3";
LCO 3.4.6, "RCS Loops—MODE 4";
LCO 3.4.7, "RCS Loops—MODE 5, Loops Filled";
LCO 3.4.8, "RCS Loops—MODE 5, Loops Not Filled";
LCO 3.9.4, "Decay Heat Removal (DHR) and Coolant Circulation—High Water Level" (MODE 6); and
LCO 3.9.5, "Decay Heat Removal (DHR) and Coolant Circulation—Low Water Level" (MODE 6).

(continued)

<INSERT B3.4-19A>

Although the Specification limits operation to a minimum of three pumps total, design evaluation (including analyses at steady state, ECCS initial conditions, and DNB conditions) also shows that operation with one pump in each loop (two pumps total) is acceptable when core THERMAL POWER is restricted to be proportionate to the flow. However, continued power operation with two RCPs removed from service is restricted to 24 hours (Ref. 2) since not all transient and accident conditions have been analyzed.

<INSERT B3.4-19B>

via two RCS loops. An operating loop consists of at least one operating RCP and a SG capable of heat removal.

BASES (continued)

ACTIONS

INSERT B 3.4-20A

A.1

~~If the requirements of the LCO are not met, the Required Action is to reduce power and bring the plant to MODE 3. This lowers power level and thus reduces the core heat removal needs and minimizes the possibility of violating DNB limits.~~

~~The Completion Time of 6 hours is reasonable, based on operating experience, to reach MODE 3 from full power conditions in an orderly manner and without challenging safety systems.~~

SURVEILLANCE REQUIREMENTS

SR 3.4.4.1

This SR requires verification every 12 hours of the required number of loops in operation. Verification includes flow rate, temperature, or pump status monitoring, which helps ensure that forced flow is providing heat removal while maintaining the margin to DNB. The 12 hour interval has been shown by operating practice to be sufficient to regularly assess degradation and verify operation within safety analyses assumptions. In addition, control room indication and alarms will normally indicate loop status.

edit

REFERENCES

1. SSAR, Chapter ~~4/A~~. 14 and 3A

2. BAW-10103A, Revision 3, July 1977.

3. 10 CFR 50.36.

edit

28

edit

<INSERT B3.4-20A>

A.1

With one RCP not in operation in each loop, the assumptions of the safety analyses are not met, but design evaluation provided in Reference 2 concludes that events initiated during two pump operation would be expected to respond within the acceptance criteria for the ECCS. However, since no analysis was performed, Technical Specifications for two pump operation will only allow operation in MODES 1 or 2 for a period not to exceed 24 hours. The Completion Time of 18 hours provides sufficient time to restore operation of an additional RCP, while allowing time to place the unit in MODE 3 within the 24 hour limitation if restoration of a third RCP is not accomplished.

B.1

If the Required Action and associated Completion Time of Condition A are not met, or if the LCO is not met for any reason other than provided in Condition A, the unit must be placed in a MODE in which the requirements are not applicable. This is accomplished by placing the unit in MODE 3. This reduces the core heat removal needs and minimizes the possibility of violating DNB limits. The Completion Time of 6 hours is reasonable, based on operating experience, to reach MODE 3 from power conditions in an orderly manner and without challenging safety systems.

B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.5 RCS Loops—MODE 3

BASES

BACKGROUND

The primary function of the reactor coolant in MODE 3 is removal of decay heat and transfer of this heat, via the steam generators (SGs), to the secondary plant fluid. The secondary function of the reactor coolant is to act as a carrier for soluble neutron poison, boric acid.

In MODE 3, reactor coolant pumps (RCPs) are used to provide forced circulation for heat removal during heatup and cooldown. The number of RCPs in operation will vary depending on operational needs, and the intent of this LCO is to provide forced flow from at least one RCP for core heat removal and transport. The flow provided by one RCP is adequate for heat removal and for boron mixing. However, two RCS loops are required to be OPERABLE to provide redundant paths for heat removal.

Reactor coolant natural circulation is not normally used~~; however, the natural circulation flow rate is sufficient for core cooling~~. If entry into natural circulation is required, the reactor coolant at the highest elevation of the hot leg must be maintained subcooled for single phase circulation. When in natural circulation, it is preferable to remove heat using both SGs to avoid idle loop stagnation that might occur if only one SG were in service. One generator will provide adequate heat removal. Boron reduction in natural circulation is prohibited because mixing to obtain a homogeneous concentration in all portions of the RCS cannot be ensured. 23

APPLICABLE SAFETY ANALYSES

No safety analyses are performed with initial conditions in MODE 3.

Failure to provide heat removal may result in challenges to a fission product barrier. The RCS loops are part of the primary success path that functions or actuates to prevent or mitigate a Design Basis Accident or transient that either assumes the failure of, or presents a challenge to, the integrity of a fission product barrier.

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

RCS Loops—MODE 3 satisfy Criterion ~~2~~ 4 of ~~the NRC Policy Statement.~~ 10 CFR 50.36 (Ref. 1). 28

LCO

The purpose of this LCO is to require two loops to be available for heat removal thus providing redundancy. The LCO requires the two loops to be OPERABLE with the intent of requiring both SGs to be capable of transferring heat from the reactor coolant at a controlled rate. Forced reactor coolant flow is the ~~required~~ preferred way to transport heat, although natural circulation flow ~~provides adequate removal~~ is also acceptable under certain conditions. A minimum of one running RCP meets the LCO requirement for one loop in operation. 23

3.4A-09 ANO-247 3.4A-10

The Note permits a limited period of operation without RCPs. All RCPs may be ~~de-energized~~ removed from operation for ≤ 8 hours per 24 hour period for the transition to or from the Decay Heat Removal (DHR) System, and otherwise may be ~~de-energized~~ removed from operation for ≤ 1 hour per 8 hour period. ~~This means that natural circulation has been established. When in natural circulation,~~ During this Condition, boron reduction with coolant at boron concentrations less than required to assure the SDM of LCO 3.1.1, is prohibited because an even concentration distribution throughout the RCS cannot be ensured. Core outlet temperature is to be maintained at least [10]°F below the saturation temperature so that no vapor bubble may form and possibly cause a natural circulation flow obstruction; and 5 23 8

<INSERT B 3.4-22A>

In MODES 3, 4, and 5, it is sometimes necessary to stop all RCP or DHR pump forced circulation (e.g., change operation from one DHR train to the other, to perform surveillance or startup testing, to perform the transition to and from DHR System cooling, or to avoid operation below the RCP minimum net positive suction head limit). ~~The time period~~ This is acceptable because ~~natural circulation is adequate for heat removal, or~~ the reactor coolant temperature can be maintained subcooled and boron stratification affecting reactivity control is not expected. 8 23

An OPERABLE RCS loop consists of at least one OPERABLE RCP and an SG that is OPERABLE ~~in accordance with the Steam Generator Tube Surveillance Program. An RCP is OPERABLE if it is capable of being powered and is able to provide forced flow if required.~~

<INSERT B 3.4-22B> edit

(continued)

<INSERT B3.4-22A>

b) pump restart criteria (which vary with pressure) are met.

<INSERT B3.4-22B>

To be considered OPERABLE, an RCP must be capable of being powered and able to provide forced flow if required. Similarly, an SG must be capable of transferring heat from the reactor coolant at a controlled rate and be in compliance with the Steam Generator Tube Surveillance Program.

BASES (continued)

APPLICABILITY

In MODE 3, the heat load is lower than at power; therefore, one RCS loop in operation is adequate for transport and heat removal. A second RCS loop is required to be OPERABLE but not in operation for redundant heat removal capability.

Operation in other MODES is covered by:

LCO 3.4.4, "RCS Loops—MODES 1 and 2";
LCO 3.4.6, "RCS Loops—MODE 4";
LCO 3.4.7, "RCS Loops—MODE 5, Loops Filled";
LCO 3.4.8, "RCS Loops—MODE 5, Loops Not Filled";
LCO 3.9.4, "Decay Heat Removal (DHR) and Coolant Circulation—High Water Level" (MODE 6); and
LCO 3.9.5, "Decay Heat Removal (DHR) and Coolant Circulation—Low Water Level" (MODE 6).

ACTIONS

A.1

If one ~~required~~ RCS loop is inoperable, redundancy for forced flow heat removal is lost. The Required Action is restoration of the ~~required~~ RCS loop to OPERABLE status within a Completion Time of 72 hours. This time allowance is a justified period to be without the redundant nonoperating loop because a single loop in operation has a heat transfer capability greater than that needed to remove the decay heat produced in the reactor core. 9

B.1

the Required Action and associated Completion Time of Condition A are not met,

If ~~restoration is not possible within 72 hours,~~ the unit must be brought to MODE 4. In MODE 4, the ~~plant~~ unit may be placed on the DHR System. The allowed Completion Time of 12 hours is reasonable, based on operating experience, to achieve cooldown and depressurization from the existing ~~plant~~ unit conditions and without challenging ~~plant~~ unit systems. edit edit edit

the conditions of 9

C.1 and C.2

a required RCS loop is not

no RCS loop is required to be in operation

If no RCS loop is OPERABLE or in operation, ~~except as~~ provided ~~in~~ the Note in the LCO section are met), all operations involving ~~a reduction of RCS boron concentration~~ Introduction of coolant into the RCS with boron concentration less than required to meet the minimum SDM of LCO 3.1.1 must be immediately suspended. ~~This is necessary because boron~~ 30

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(continued)

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BASES

ACTIONS

C.1 and C.2 (continued)

~~dilution requires forced circulation for proper homogenization.~~ Action to restore one RCS loop to operation shall be immediately initiated and continued until one RCS loop is restored to operation and to OPERABLE status. The immediate Completion Time reflects the importance of maintaining operation for decay heat removal. 30

<INSERT B 3.4-24A>

SURVEILLANCE REQUIREMENTS

SR 3.4.5.1

This SR requires verification every 12 hours that the required ~~number of~~ loop~~s~~ and pump~~s~~ is in operation. Verification includes flow rate, temperature, or pump status monitoring, which help ensure that forced flow is providing heat removal. The 12 hour interval has been shown by operating practice to be sufficient to regularly assess RCS loop status. In addition, control room indication and alarms will normally indicate loop status. edit

SR 3.4.5.2

Verification that ~~the~~ each required ~~number of~~ RCP~~s are~~ is OPERABLE ensures ~~that the single failure criterion is met and~~ that an ~~additional~~ RCS loop can be placed in operation, if needed, to maintain decay heat removal and reactor coolant circulation. Verification is performed by verifying proper breaker alignment and power availability to ~~the~~ each required pump that is not in operation. Alternatively, verification that a pump is in operation also verifies proper breaker alignment and power availability. The Frequency of 7 days is considered reasonable in view of other administrative controls available and has been shown to be acceptable by operating experience. edit 10

<INSERT B 3.4-24 B>

REFERENCES

~~None.~~
1. 10 CFR 50.36. 28

<INSERT B3.4-24A>

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Suspending the introduction of coolant into the RCS with boron concentration less than required to meet the minimum SDM of LCO 3.1.1 is required to assure continued safe operation. With coolant added without forced circulation, unmixed coolant could be introduced to the core, however coolant added with boron concentration meeting the minimum SDM maintains acceptable margin to subcritical operations.

<INSERT B3.4-24B>

This SR is modified by a Note that states the SR is not required to be performed until 24 hours after a required pump is not in operation.

B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.6 RCS Loops—MODE 4

BASES

BACKGROUND

In MODE 4, the primary function of the reactor coolant is the removal of decay heat and transfer of this heat to the steam generators (SGs) or decay heat removal (DHR) heat exchangers. The secondary function of the reactor coolant is to act as a carrier for soluble neutron poison, boric acid.

In MODE 4, either reactor coolant pumps (RCPs) or DHR pumps can be used for coolant circulation. The number of pumps in operation can vary to suit the operational needs. The intent of this LCO is to provide forced flow from at least one RCP or one DHR pump for decay heat removal and transport. The flow provided by one RCP or one DHR pump is adequate for heat removal. The other intent of this LCO is to require that two paths (loops) be available to provide redundancy for heat removal.

APPLICABLE SAFETY ANALYSES

AND-248

No safety analyses are performed with initial condition in MODE 4.

RCS Loops—MODE 4 ~~have been identified in the NRC Policy Statement as an important contributor to risk reduction.~~ Satisfies Criterion 4 of 10 CFR 50.36 (Ref. 1) 28

LCO

The purpose of this LCO is to require that two loops, RCS or DHR, be OPERABLE in MODE 4 and one of these loops be in operation. The LCO allows the two loops that are required to be OPERABLE to consist of any combination of RCS or DHR System loops. Any one loop in operation provides enough flow to remove the decay heat from the core with forced circulation. The second loop that is required to be OPERABLE provides redundant paths for heat removal.

The Note permits a limited period of operation with ~~out RCPs.~~ the normally required RCP or DHR pump removed from operation. ~~All RCPs may be de-energized for ≤ 8 hours per 24 hour period for the transition to or from the DHR System and otherwise may be de-energized for ≤ 1 hour per 8 hour period.~~ 11 ~~This means that natural circulation has been~~ 23

(continued)

BASES

LCO (continued)

~~established using the SGs.~~ The Note prohibits boron dilution when forced flow is stopped because an even concentration distribution cannot be ensured. Core outlet temperature is to be maintained ~~at least 10°F~~ below saturation temperature so that no vapor bubble may form and possibly cause a natural circulation flow obstruction.

by ≥ 10°F

with coolant at boron concentrations less than required to assure the SDM of LCO 3.1.1 is maintained

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23 30 edit 11

~~The Note also permits the DHR pumps to be stopped for ≤ 1 hour per 8 hour period.~~ When the DHR pumps are stopped, no alternate heat removal path exists, unless the RCS and SGs have been placed in service in forced or natural circulation. The response of the RCS without the DHR System depends on the core decay heat load and the length of time that the DHR pumps are stopped. As decay heat diminishes, the effects on RCS temperature and pressure diminish. Without cooling by DHR, higher heat loads will cause the reactor coolant temperature and pressure to increase at a rate proportional to the decay heat load. Because pressure can increase, the applicable system pressure limits (pressure and temperature (P/T) or low temperature overpressure protection (LTOP) limits) must be observed and forced DHR flow or heat removal via the SGs must be re-established prior to reaching the pressure limit. ~~The circumstances for stopping both DHR trains are to be limited to situations where:~~

~~a. Pressure and pressure and temperature increases can be maintained well within the allowable pressure (P/T and LTOP) and 10°F subcooling limits; or~~

~~b. An alternate heat removal path through the SG is in operation.~~

heat removal through

or the SGs

if the SGs are not capable of removing heat,

11

An OPERABLE RCS loop consists of at least one OPERABLE RCP and an SG ~~that is~~ OPERABLE ~~in accordance with the Steam Generator Tube Surveillance Program.~~

INSERT B 3.4-26A

edit

Similarly for the DHR System, an OPERABLE DHR loop is comprised of the OPERABLE DHR pump(s) capable of ~~providing forced flow to~~ the DHR heat exchanger(s). ~~DHR pumps are OPERABLE if they are~~ capable of being powered and ~~are~~ able to provide flow if required.

circulating RCS fluid through

and back to the RCS

To be considered OPERABLE, a DHR pump must be

INSERT B 3.4-26B

edit

19

(continued)

<INSERT B3.4-26A>

To be considered OPERABLE, an SG must be capable of transferring heat from the reactor coolant at a controlled rate and be in compliance with the Steam Generator Tube Surveillance Program.

<INSERT B3.4-26B>

and a DHR heat exchanger must be capable of transferring heat from the reactor coolant at a controlled rate.

A DHR loop may be considered OPERABLE during alignment and when aligned for low pressure injection if it is capable of being manually (locally or remotely) realigned to the DHR mode of operation and is not otherwise inoperable. This provision arises because of the dual requirements of the components that comprise the low pressure injection/decay heat removal system.

BASES (continued)

APPLICABILITY

In MODE 4, this LCO applies because it is possible to remove core decay heat and to provide proper boron mixing with either the RCS loops and SGs or the DHR System.

Operation in other MODES is covered by:

LCO 3.4.4, "RCS Loops—MODES 1 and 2";
LCO 3.4.5, "RCS Loops—MODE 3";
LCO 3.4.7, "RCS Loops—MODE 5, Loops Filled";
LCO 3.4.8, "RCS Loops—MODE 5, Loops Not Filled";
LCO 3.9.4, "Decay Heat Removal (DHR) and Coolant Circulation—High Water Level" (MODE 6); and
LCO 3.9.5, "Decay Heat Removal (DHR) and Coolant Circulation—Low Water Level" (MODE 6).

ACTIONS

A.1

If only one required RCS loop or DHR loop is OPERABLE and in operation, redundancy for heat removal is lost. Action must be initiated to restore a second loop to OPERABLE status. The immediate Completion Time reflects the importance of maintaining the availability of two paths for heat removal.

A.2 ~~B.1 and B.2~~

~~If only one DHR loop is operable, an inoperable RCS or DHR loop must be restored to OPERABLE status to satisfy single failure considerations. The action must be started immediately and the immediate Completion Time reflects the urgency of restoring redundancy for heat removal. One loop is still available for cooldown for the reduced heat loads of this operating MODE.~~

If restoration ~~cannot be~~ is not accomplished and a DHR loop is OPERABLE, the unit must be brought to MODE 5 within the following 24 hours. Bringing the unit to MODE 5 is a conservative action with regard to decay heat removal. With only one DHR loop OPERABLE, redundancy for decay heat removal is lost and, in the event of a loss of the remaining DHR loop, it would be safer to initiate that loss from MODE 5 ~~(≤ 200°F)~~ rather than MODE 4 ~~(200°F to 300°F)~~. The Completion Time of 24 hours is reasonable, based on (12)

(continued)

BASES

ACTIONS A.2 ~~B.1 and B.2~~ (continued)

<INSERT B3.4-28A>

operating experience, to reach MODE 5 in an orderly manner and without challenging ~~plant~~ unit systems. edit 12

~~C~~B.1 and ~~C~~B.2

ANO-247

If no RCS or DHR loops are of OPERABLE or a required loop is not in operation, ~~except during~~ (no loop is required to be in operation provided the) conditions ~~permitted by~~ the Note in the LCO section, are met all operations involving ~~a reduction of RCS boron concentration~~ (Introduction of coolant into the RCS with boron concentration less than required to meet the minimum SDM of LCO 3.1.1) must be suspended and action to restore one RCS or DHR loop to OPERABLE status and operation must be initiated. ~~Boron dilution requires forced circulation for proper mixing, and~~ The required margin to criticality must not be reduced in this type of operation. The immediate Completion Times reflect the importance of maintaining operation for decay heat removal. The action to restore must continue until one loop is restored to operation. 9 30

<INSERT B3.4-28B>

SURVEILLANCE REQUIREMENTS

SR 3.4.6.1

This Surveillance requires verification every 12 hours of the required ~~number of~~ DHR or RCS loop~~s~~ in operation to ensure forced flow is providing decay heat removal. Verification includes flow rate, temperature, or pump status monitoring. The 12 hour interval has been shown by operating practice to be sufficient to regularly assess RCS loop status. In addition, control room indication and alarms will normally indicate loop status. 9

SR 3.4.6.2

<INSERT B3.4-28C>

Verification that ~~the~~ each required pump is OPERABLE ensures that an ~~additional~~ RCS or DHR loop can be placed in operation if needed to maintain decay heat removal and reactor coolant circulation. Verification is performed by verifying proper breaker alignment and power available to ~~the~~ each required pump~~s~~. The Frequency of 7 days is considered reasonable in view of other administrative controls and has been shown to be acceptable by operating experience. edit edit 10

<INSERT B3.4-28D>

(continued)

<INSERT B3.4-28A>

This Required Action is modified by a Note which indicates that the unit must be placed in MODE 5 only if a DHR loop is OPERABLE. With no DHR loop OPERABLE, the unit is in a condition with only limited cooldown capabilities. Therefore, the actions are to be concentrated on restoration of a DHR loop, rather than a cooldown of extended duration.

<INSERT B3.4-28B>

ANO-247

Suspending the introduction of coolant into the RCS with boron concentration less than required to meet the minimum SDM of LCO 3.1.1 is required to assure continued safe operation. With coolant added without forced circulation, unmixed coolant could be introduced to the core, however coolant added with boron concentration meeting the minimum SDM maintains acceptable margin to subcritical operations.

<INSERT B3.4-28C>

Alternatively, verification that a pump is in operation also verifies proper breaker alignment and power availability.

<INSERT B3.4-28D>

This SR is modified by a Note that states the SR is not required to be performed until 24 hours after a required pump is not in operation.

BASES (continued)

REFERENCES ~~None.~~ 1. 10 CFR 50.36. (28)

B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.7 RCS Loops—MODE 5, Loops Filled

BASES

BACKGROUND

service — edit

do not typically — edit

In MODE 5 with RCS loops filled, the primary function of the reactor coolant is the removal of decay heat and transfer of this heat either to the steam generator (SG) secondary side coolant or the ~~component cooling~~ water via the decay heat removal (DHR) heat exchangers. While the principal means for decay heat removal is via the DHR System, the SGs are specified as a backup means for redundancy. Although the SGs ~~cannot~~ remove heat unless steaming occurs ~~(which is not possible in MODE 5)~~, they are available as a temporary heat sink and can be used by allowing the RCS to heat up into the temperature region of MODE 4 where steaming can be effective for heat removal. The secondary function of the reactor coolant is to act as a carrier for soluble neutron poison, boric acid.

a backup method

In MODE 5 with RCS loops filled, DHR loops are the principal means for heat removal. The number of loops in operation can vary to suit the operational needs. The intent of this LCO is to provide forced flow from at least one DHR loop for decay heat removal and transport. The flow provided by one DHR loop is adequate for decay heat removal. The other intent of this LCO is to require that a second path be available to provide ~~redundancy~~ for heat removal.

Move to LCO Bases — edit

the auxiliary feedwater — edit

emergency — edit

3.4A-12

The LCO provides for either SG heat removal or DHR System heat removal. In this MODE, reactor coolant pump (RCP) operation may be restricted because of net positive suction head (NPSH) limitations, and the SG will not be able to provide steam for the turbine driven feed pumps. However, to ensure that the SG(s) can be used as a heat sink, a motor driven feedwater pump is needed, because it is independent of steam. Condensate pumps, ~~startup~~ pump(s), or the motor driven ~~auxiliary~~ feedwater pump can be used. If RCPs are available, the steam generator level need not be adjusted. If RCPs are not available, the water level must be adjusted for natural circulation. The high entry point in the generator should be accessible from the feedwater pumps so that natural circulation can be stimulated. The SGs are primarily a backup to the DHR pumps, which are used for forced flow. By requiring the SGs to be a backup heat

(continued)

BASES

Move to LCO Bases

BACKGROUND (continued)

removal path, the option to increase RCS pressure and temperature for heat removal in MODE 4 is provided.

APPLICABLE SAFETY ANALYSES

No safety analyses are performed with initial conditions in MODE 5.

ANO-248

RCS Loops—MODE 5 (Loops Filled) ~~have been identified in the NRC Policy Statement as important contributors to risk reduction~~. Satisfies Criterion 4 of 10 CFR 50.36 (Ref. 1) (28)

LCO

3.4A-12

The purpose of this LCO is to require that at least one of the DHR loops be OPERABLE and in operation with an additional DHR loop OPERABLE or both SGs with secondary side water level ≥ ~~[50]%~~ 20 inches. One DHR loop provides sufficient forced circulation to perform the safety functions of the reactor coolant under these conditions. The second DHR loop is normally maintained as a backup to the operating DHR loop to provide redundancy for decay heat removal. However, if the standby DHR loop is not OPERABLE, a sufficient alternate method of providing redundant heat removal paths is to provide both SGs with their secondary side water levels ≥ ~~[50]%~~. Should the operating DHR loop fail, the SGs could be used to remove the decay heat.

INSERT "MOVE" from BASES BACKGROUND

Note 1 permits the DHR pumps to be stopped for up to 1 hours ~~per 8 hour period~~. The circumstances for stopping both DHR trains are to be limited to situations where: (a) Pressure and temperature increases can be maintained well within the allowable pressure (P/T and low temperature overpressure protection) and 10°F subcooling limits; ~~or (b) Alternate heat paths through the SGs are in operation~~. and (b) no operations are in process that would cause reduction of the RCS boron concentration. H13 edit

ANO-247

The Note prohibits boron dilution with coolant at boron concentrations less than required to assure the SDM of LCO 3.1.1 is maintained when DHR forced flow is stopped because an even concentration distribution cannot be ensured. Core outlet temperature is to be maintained ~~at least 10°F~~ by ≥ 10°F below saturation temperature so that no vapor bubble would form and possibly cause a natural circulation flow obstruction. In this MODE, the steam generators are used as a backup for decay heat removal and, to ensure their availability, the RCS loop flow path is to be maintained with subcooled liquid. (30) edit edit

(continued)

BASES

For example, this may be necessary

LCO (continued)

In MODE 5, it is sometimes necessary to stop all RCP or DHR pump forced circulation. ~~This is permitted~~ to change operation from one DHR train to the other, perform surveillance or startup testing, perform the transition to and from the DHR System, or to avoid operation below the RCP minimum NPSH limit. The time period is acceptable because ~~natural circulation is acceptable for heat removal,~~ the reactor coolant temperature can be maintained subcooled, and boron stratification affecting reactivity control is not expected. 13 23

Note 2 allows one required DHR loop to be inoperable for a period of ≤ ~~up to~~ 2 hours provided that the other loop is OPERABLE and in operation. This permits periodic surveillance tests to be performed on the inoperable loop during the only time when such testing is safe and possible. edit

Note 3 provides for an orderly transition from MODE 5 to MODE 4 during a planned heatup by permitting removal of DHR loops from operation when at least one RCP is in operation. This Note provides for the transition to MODE 4 where an RCP is permitted to be in operation and replaces the RCS circulation function provided by the DHR loops. 19

INSERT B 3.4-32A

An OPERABLE DHR loop is composed of an OPERABLE DHR pump and an OPERABLE DHR heat exchanger.

To be considered DHR pumps ~~are~~ OPERABLE, ~~if they are~~ must be capable of being powered and are able to provide flow if required. Similarly, an ~~An~~ OPERABLE SG can perform as a heat sink when it has an adequate water level and is OPERABLE in ~~accordance~~ compliance with the Steam Generator Tube Surveillance Program. edit

ANO-08
3.4A-12

APPLICABILITY

In MODE 5 with loops filled, forced circulation is provided by this LCO to remove decay heat from the core and to provide proper boron mixing. One loop of DHR provides sufficient circulation for these purposes.

Operation in other MODES is covered by:

LCO 3.4.4, "RCS Loops—MODES 1 and 2";
LCO 3.4.5, "RCS Loops—MODE 3";
LCO 3.4.6, "RCS Loops—MODE 4";
LCO 3.4.8, "RCS Loops—MODE 5, Loops Not Filled";

(continued)

<INSERT B3.4-32A>

A DHR loop may be considered OPERABLE during alignment and when aligned for low pressure injection if it is capable of being manually (locally or remotely) realigned to the DHR mode of operation and is not otherwise inoperable. This provision arises because of the dual requirements of the components that comprise the low pressure injection/decay heat removal system.

BASES

APPLICABILITY (continued)

LCO 3.9.4, "Decay Heat Removal (DHR) and Coolant Circulation—High Water Level" (MODE 6); and
LCO 3.9.5, "Decay Heat Removal (DHR) and Coolant Circulation—Low Water Level" (MODE 6).

edit

ACTIONS

, B.1 and B.2

3.4A-11
3.4A-12

A.1 ~~and~~ A.2 required 9

If one required DHR loop is inoperable and any required SG has secondary side water level ≤ ~~[50]%~~ 20 inches redundancy for heat removal is lost. Action must be initiated to restore a second DHR loop to OPERABLE status or initiate action to restore the secondary side water level in the SGs, and action must be taken immediately. Either Required Action ~~A.1 or Required Action A.2~~ will restore redundant decay heat removal paths. The immediate Completion Time reflects the importance of maintaining the availability of two paths for heat removal. 9

3.4A-11

~~B~~C.1 and ~~B~~C.2 required

ANO-247

If no required DHR loop is in operation, except as provided in Note 1, or no required DHR loop is OPERABLE, all operations involving ~~the reduction of RCS boron concentration~~ Introduction of coolant into the RCS with boron concentration less than required to meet the minimum SDM of LCO 3.1.1 must be suspended and action to restore a DHR loop to OPERABLE status and operation must be initiated. ~~Boron dilution requires forced circulation for proper mixing, and the~~ required margin to criticality must not be reduced in this type of operation. The immediate Completion Time reflects the importance of maintaining operation for decay heat removal. 30

<INSERT B 3.4-33 A>

SURVEILLANCE REQUIREMENTS

SR 3.4.7.1

This SR requires verification every 12 hours that the required DHR loop is in operation. Verification includes flow rate, temperature, or pump status monitoring, which help ensure that forced flow is providing heat removal. The 12 hour Frequency has been shown by operating practice to be sufficient to regularly assess degradation. In addition, control room indication and alarms will normally indicate loop status.

(continued)

<INSERT B3.4-33A>

ANO-247

Suspending the introduction of coolant into the RCS with boron concentration less than required to meet the minimum SDM of LCO 3.1.1 is required to assure continued safe operation. With coolant added without forced circulation, unmixed coolant could be introduced to the core, however coolant added with boron concentration meeting the minimum SDM maintains acceptable margin to subcritical operations.

3.4A-12

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.4.7.2

20 inches

Verifying the SGs are OPERABLE by ensuring their secondary side water levels are ≥ 50% ensures that redundant heat removal paths are available if the second DHR loop is not OPERABLE. If both DHR loops are OPERABLE, this Surveillance is not needed. The 12 hour Frequency has been shown by operating practice to be sufficient to regularly assess ~~degradation and verify operation within safety analyses assumptions.~~ RCS loop status. 27

SR 3.4.7.3

each required

Verification that ~~the second~~ DHR pump is OPERABLE ensures that ~~redundant paths for heat removal are available. The requirement also ensures that the additional~~ a DHR loop can be placed in operation if needed to maintain decay heat removal and reactor coolant circulation. If the secondary side water level is ≥ 50% 20 inches in both SGs, this Surveillance is not needed. Verification is performed by verifying proper breaker alignment and power available to ~~the~~ each required pump. 10 edit The Frequency of 7 days is considered reasonable in view of other administrative controls available and has been shown to be acceptable by operating experience. 10

edit

3.4A-12

INSERT B 3.4-34A

INSERT B3.4-34B

REFERENCES ~~None.~~ 1. 10 CFR 50.36. 28

<INSERT B3.4-34A>

Alternatively, verification that a pump is in operation also verifies proper breaker alignment and power availability.

<INSERT B3.4-34B>

This SR is modified by a Note that states the SR is not required to be performed until 24 hours after a required pump is not in operation.

B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.8 RCS Loops—MODE 5, Loops Not Filled

BASES

BACKGROUND

In MODE 5 with loops not filled, the primary function of the reactor coolant is the removal of decay heat and transfer of this heat to the decay heat removal (DHR) heat exchangers. The steam generators (SGs) are not available as a heat sink when the loops are not filled. The secondary function of the reactor coolant is to act as a carrier for the soluble neutron poison, boric acid.

considered
RCS draining

Loops are not filled when the ~~reactor coolant water level is within the horizontal portion of the hot leg~~ as might be the case for refueling or maintenance ~~on the reactor coolant pumps or SGs.~~ initiated GL 88-17 (Ref. 1) expresses concerns for loss of decay heat removal for this operating condition. With water at this low level, the margin above the decay heat suction piping connection to the hot leg is small. The possibility of loss of level or inlet vortexing exists and if it were to occur, the operating DHR pump could become air bound and fail resulting in a loss of forced flow for heat removal. As a consequence the water in the core will heat up and could boil with the possibility of core uncovering due to boil off. ~~Because the containment hatch may be open at this time, a pathway to the outside for fission product release exists if core damage were to occur.~~

edit

Additionally, reductions of RCS inventory below el. 375 ft. are termed reduced inventory operations.

22

edit

In MODE 5 with loops not filled, only DHR pumps can be used for coolant circulation. The number of pumps in operation can vary to suit the operational needs. The intent of this LCO is to provide require forced flow from at least one DHR pump for and decay heat removal and transport, to require that two paths be available to provide redundancy for heat removal.

edit

APPLICABLE SAFETY ANALYSES

No safety analyses are performed with initial conditions in MODE 5 with loops not filled. ~~The flow provided by one DHR pump is adequate for heat removal and for boron mixing.~~

edit

RCS Loops—MODE 5 (Loops Not Filled) ~~have been identified in the NRC Policy Statement as important contributors to risk reduction.~~ Satisfies Criterion 4 of 10 CFR 50.36 (Ref. 2)

28

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(continued)

BASES (continued)

LCO

The purpose of this LCO is to require that a minimum of two DHR loops be OPERABLE and that one of these loops be in operation. An OPERABLE loop is one that has the capability of transferring heat from the reactor coolant at a controlled rate. Heat cannot be removed via the DHR system unless forced flow is used. A minimum of one running decay heat removal pump meets the LCO requirement for one loop in operation. An additional DHR loop is required to be OPERABLE to provide redundancy for heat removal.

Note 1 permits the DHR pumps to be de-energized for ≤ 1 hour. ~~≤ 15 minutes when switching from one train to the other. The circumstances for stopping both DHR pumps are to be limited to situations where the outage time is short [and temperature is maintained ≤ [160]°F].~~ The Note prohibits boron dilution with coolant at boron concentrations less than required to assure the SDM of LCO 3.1.1 is maintained or draining operations when DHR forced flow is stopped. 15 30

ANO-247

Note 2 allows one DHR loop to be inoperable for a period of ≤ 2 hours provided that the other loop is OPERABLE and in operation. This permits periodic surveillance tests to be performed on the inoperable loop during ~~the only time~~ MODE 5 when these tests are safe and possible. 19

<INSERT B 3.4-36A>

An OPERABLE ~~DHR loop is composed of~~ circulating RCS fluid through an OPERABLE DHR pump capable of ~~providing forced flow to~~ an OPERABLE DHR heat exchanger and back to the RCS. To be considered OPERABLE, the DHR pumps ~~are OPERABLE if they are~~ must be capable of being powered and ~~are~~ able to provide flow if required. edit

APPLICABILITY

In MODE 5 with loops not filled, this LCO requires core heat removal and coolant circulation by the DHR System.

Operation in other MODES is covered by:

LCO 3.4.4, "RCS Loops—MODES 1 and 2";
LCO 3.4.5, "RCS Loops—MODE 3";
LCO 3.4.6, "RCS Loops—MODE 4";
LCO 3.4.7, "RCS Loops—MODE 5, Loops Filled";
LCO 3.9.4, "Decay Heat Removal (DHR) and Coolant Circulation—High Water Level" (MODE 6); and
LCO 3.9.5, "Decay Heat Removal (DHR) and Coolant Circulation—Low Water Level" (MODE 6).

(continued)

<INSERT B3.4-36A>

A DHR loop may be considered OPERABLE during alignment and when aligned for low pressure injection if it is capable of being manually (locally or remotely) realigned to the DHR mode of operation and is not otherwise inoperable. This provision arises because of the dual requirements of the components that comprise the low pressure injection/decay heat removal system.

BASES (continued)

ACTIONS

A.1

required inoperable 9

If only one DHR loop is OPERABLE, redundancy for heat removal is lost. Required Action A.1 is to immediately initiate activities to restore a second loop to OPERABLE status. The immediate Completion Time reflects the importance of maintaining the availability of two paths for heat removal.

3.4A-11

, B.2, 25

B.1 and B.2 3

is OPERABLE 9

10

If both required loops ~~are inoperable~~ or the required loop is not in operation, except as provided by Note 1 in the LCO, the Required Action requires immediate suspension of all operations involving boron reduction and requires initiation of action to immediately restore one DHR loop to OPERABLE status and operation. The Required Action for restoration does not apply to the condition of both loops not in operation when the exception Note in the LCO is in force. The immediate Completion Time reflects the importance of maintaining operations for decay heat removal. The action to restore must continue until one loop is restored.

<INSERT B 3.4-37A> 25

or reduction of RCS water inventory 30

<INSERT B 3.4-37B>

SURVEILLANCE REQUIREMENTS

SR 3.4.8.1

This Surveillance requires verification every 12 hours that at least one loop is in operation. Verification includes flow rate, temperature, or pump status monitoring, which help ensure that forced flow is providing heat removal. The 12 hour interval has been shown by operating practice to be sufficient to regularly assess ~~degradation and verify operation within safety analyses assumptions.~~ RCS loop status. 27

SR 3.4.8.2

each a DHR is

Verification that the required ~~number of~~ pumps ~~are~~ OPERABLE ensures that redundancy for heat removal is provided. The requirement also ensures that additional loops can be placed in operation if needed to maintain decay heat removal and reactor coolant circulation. Verification is performed by verifying proper breaker alignment and power available to edit

(continued)

<INSERT B3.4-37A>

ANO-247

introduction of coolant into the RCS with boron concentration less than required to meet the minimum SDM of LCO 3.1.1

<INSERT B3.4-37B>

ANO-247

Suspending the introduction of coolant into the RCS with boron concentration less than required to meet the minimum SDM of LCO 3.1.1 is required to assure continued safe operation. With coolant added without forced circulation, unmixed coolant could be introduced to the core, however coolant added with boron concentration meeting the minimum SDM maintains acceptable margin to subcritical operations.

Alternatively, verification that a pump is in operation also verifies proper breaker alignment and power availability.

BASES

SURVEILLANCE REQUIREMENTS

SR 3.4.8.2 (continued)

each ~~the~~ required pump~~s~~. The Frequency of 7 days is considered reasonable in view of other administrative controls available and has been shown to be acceptable by operating experience.

INSERT 3.4-38A

10 edit

10

REFERENCES

1. Generic Letter 88-17, October 17, 1988.
2. 10 CFR 50.36.

28

<INSERT B3.4-38A>

This SR is modified by a Note that states the SR is not required to be performed until 24 hours after a required pump is not in operation.

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.9 Pressurizer

LCO 3.4.9 The pressurizer shall be OPERABLE with:

a. Pressurizer water level ≥ 45 inches and ≤ 320 inches; and

b. A minimum of 126 kW of Engineered Safeguards (ES) bus powered pressurizer heaters OPERABLE.

--NOTE--
OPERABILITY requirements on pressurizer heaters do not apply in MODE 4.

APPLICABILITY: MODES 1, 2, and 3,
MODE 4 with RCS temperature > 262°F.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Pressurizer water level not within limits.	A.1 Restore level to within limits.	1 hour
B. Required Action and associated Completion Time of Condition A not met.	B.1 Be in MODE 3. <u>AND</u> B.2 Be in MODE 4 with RCS temperature ≤ 262°F.	6 hours 24 hours
C. Capacity of ES bus powered pressurizer heaters less than limit.	C.1 Restore pressurizer heater capacity.	72 hours
D. Required Action and associated Completion Time of Condition C not met.	D.1 Be in MODE 3. <u>AND</u> D.2 Be in MODE 4.	6 hours 12 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.4.9.1	Verify pressurizer water level ≥ 45 inches and ≤ 320 inches.	12 hours
SR 3.4.9.2	Verify capacity of ES bus powered pressurizer heaters ≥ 126 kW.	18 months

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.10 Pressurizer Safety Valves

LCO 3.4.10 Two pressurizer safety valves shall be OPERABLE.

-----NOTES-----

1. Only one pressurizer safety valve is required to be OPERABLE in MODE 3, and in MODE 4 with RCS temperature > 262°F.

2. The lift settings are not required to be within limits for entry into MODE 3 or the applicable portions of MODE 4 for the purpose of setting the pressurizer safety valves under ambient (hot) conditions. This exception is allowed for 36 hours following entry into MODE 3 provided a preliminary cold setting was made prior to heatup.

3. Not applicable in MODE 3, and in MODE 4 with RCS temperature > 262°F during hydrostatic tests in accordance with ASME Boiler and Pressure Vessel Code, Section III.

4. The provisions of LCO 3.0.3 are not applicable in MODE 3, and in MODE 4 with RCS temperature > 262°F.

APPLICABILITY: MODES 1, 2, and 3,
MODE 4 with RCS temperature > 262°F.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One pressurizer safety valve inoperable in MODES 1 or 2.	A.1 Restore valve to OPERABLE status.	15 minutes
B. Required Action and associated Completion Time of Condition A not met. <u>OR</u> Two pressurizer safety valves inoperable in MODES 1 or 2.	B.1 Be in MODE 3.	6 hours

CONDITION	REQUIRED ACTION	COMPLETION TIME
C. Required pressurizer safety valve inoperable in MODE 3 or MODE 4 with RCS temperature > 262°F.	C.1 Be in MODE 4 with RCS temperature ≤ 262°F.	18 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.4.10.1	Verify each required pressurizer safety valve is OPERABLE in accordance with the Inservice Testing Program. Following testing, as-left lift settings shall be within ± 1%.	In accordance with the Inservice Testing Program

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.11 Low Temperature Overpressure Protection (LTOP) System

LCO 3.4.11 An LTOP System shall be OPERABLE with high pressure injection (HPI) deactivated and the core flood tanks (CFTs) isolated and:

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

1. HPI deactivation and CFT isolation not applicable during ASME Section XI testing.
2. HPI deactivation not applicable during fill and vent of the RCS.
3. HPI deactivation not applicable during emergency RCS makeup.
4. HPI deactivation not applicable during valve maintenance.
5. CFT isolation is only required when CFT pressure is greater than or equal to the maximum RCS pressure for the existing RCS temperature allowed by the pressure and temperature curves provided in LCO 3.4.3, "RCS Pressure and Temperature (P/T) Limits."

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a. Pressurizer level such that the unit is not in a water solid condition and an OPERABLE electromatic relief valve (ERV) with a setpoint of $\leq$ 460 psig; or

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

1. Pressurizer level not applicable as allowed by Emergency Operating Procedures.
2. Pressurizer level not applicable during system hydrotest.

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b. The RCS depressurized and the RCS open.

APPLICABILITY: MODE 4 with RCS temperature $\leq$ 262°F,
MODE 5,
MODE 6 when the reactor vessel head is on.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Pressurizer level not within required limits.	A.1 Restore pressurizer level to within required limits.	1 hour

CONDITION	REQUIRED ACTION		COMPLETION TIME
B. Required Action and associated Completion Time of Condition A not met.	B.1	Close and maintain closed the makeup control valve and its associated isolation valve.	12 hours
	AND		
	B.2	Stop RCS heatup.	12 hours
C. Required Electromatic Relief Valve (ERV) inoperable.	C.1	Restore required ERV to OPERABLE status.	1 hour
D. Required Action and associated Completion Time of Condition C not met.	D.1	Reduce makeup tank level to $\leq$ 73 inches.	12 hours
E. LCO requirements not met for any reason other than Condition A through Condition D.	E.1	Initiate action to restore compliance with LCO requirements.	Immediately

SURVEILLANCE REQUIREMENTS

	SURVEILLANCE	FREQUENCY
SR 3.4.11.1	Verify pressurizer level does not represent a water solid condition.	30 minutes during RCS heatup and cooldown <u>AND</u> 12 hours
SR 3.4.11.2	Verify HPI is deactivated.	12 hours
SR 3.4.11.3	Verify each pressurized CFT is isolated.	12 hours
SR 3.4.11.4	-----------------------------NOTE----------------------------- Verification of locked, sealed, or otherwise secured open vent path(s) only required to be performed every 31 days. --- Verify OPERABLE pressure relief capability.	 12 hours
SR 3.4.11.5	Perform CHANNEL CALIBRATION of ERV opening circuitry.	18 months

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.12 RCS Specific Activity

LCO 3.4.12 The specific activity of the reactor coolant shall be:

a. $\leq$ 3.5 μCi/gm DOSE EQUIVALENT I-131; and

b. $\leq 72/\bar{E}$ μCi/gm total.

APPLICABILITY: MODES 1 and 2,
MODE 3 with RCS average temperature (T_{avg}) $\geq$ 500°F.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Specific activity not within limits.	A.1 Restore specific activity to within limit(s).	24 hours
B. Required Action and associated Completion Time not met.	B.1 Be in MODE 3 with T_{avg} < 500°F.	6 hours

SURVEILLANCE REQUIREMENTS

	SURVEILLANCE	FREQUENCY
SR 3.4.12.1	Verify reactor coolant gross specific activity $\leq 72/\bar{E}$ μCi/gm.	7 days
SR 3.4.12.2	------------------------------NOTE------------------------------ Only required to be performed in MODE 1. -- Verify reactor coolant DOSE EQUIVALENT I-131 specific activity $\leq$ 3.5 μCi/gm.	14 days

SURVEILLANCE		FREQUENCY
SR 3.4.12.3	------------------------------NOTE------------------------------ Not required to be performed until 31 days after a minimum of 2 EFPD and 20 days of MODE 1 operation have elapsed since the reactor was last subcritical for ≥ 48 hours. -- Determine $\bar{E}$.	 184 days

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.13 RCS Operational LEAKAGE

LCO 3.4.13 RCS operational LEAKAGE shall be limited to:

a. No pressure boundary LEAKAGE;

b. 1 gpm unidentified LEAKAGE;

c. 10 gpm identified LEAKAGE; and

d. 150 gallons per day primary to secondary LEAKAGE through any one SG.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. RCS primary to secondary LEAKAGE not within limits.	A.1 Reduce LEAKAGE to within limits.	4 hours
B. RCS unidentified or identified LEAKAGE not within limits.	B.1 Reduce LEAKAGE to within limits.	18 hours
C. Required Action and associated Completion Time of Condition A or B not met. OR Pressure boundary LEAKAGE exists.	C.1 Be in MODE 3. AND C.2 Be in MODE 5.	6 hours 36 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.4.13.1	------------------------------NOTE------------------------------ Not required to be performed until 12 hours after establishment of steady state operation at or near operating pressure. --- Verify RCS operational LEAKAGE is within limits by performance of an RCS water inventory balance.	72 hours
SR 3.4.13.2	Verify steam generator tube integrity is in accordance with the Steam Generator Tube Surveillance Program.	In accordance with the Steam Generator Tube Surveillance Program

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.14 RCS Pressure Isolation Valve (PIV) Leakage

LCO 3.4.14 Leakage from each PIV shall be within limits.

APPLICABILITY: MODES 1, 2, and 3.
MODE 4, except valves in the decay heat removal (DHR) flow path when in, or during the transition to or from, the DHR 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 an inoperable pressure isolation function.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more flow paths with leakage from one or more RCS pressure isolation check valves not within limit.	A.1 Isolate the high pressure portion of the affected system from the low pressure portion by use of one closed deactivated automatic valve and one OPERABLE check valve.	4 hours
B. Required Decay Heat Removal (DHR) System autoclosure interlock function inoperable.	B.1 Isolate the affected penetration by use of one closed manual or deactivated automatic valve.	4 hours
C. Required Action and associated Completion Time not met.	C.1 Be in MODE 3.	6 hours
	AND	
	C.2 Be in MODE 5.	36 hours

SURVEILLANCE REQUIREMENTS

	SURVEILLANCE	FREQUENCY
SR 3.4.14.1	-----------------------------NOTE----------------------------- Not required to be performed in MODES 3 and 4. --- Verify leakage from each RCS pressure isolation check valve, or pair of check valves, as applicable, is less than or equal to an equivalent of the Allowable Leakage Limit identified below at a differential test pressure ≥ 150 psid. Pressure Isolation Check Valves(s) — Allowable Leakage Limit DH-14A — ≤ 5 gpm DH-13A and DH-17 — ≤ 5 gpm total DH-14B — ≤ 5 gpm DH-13B and DH-18 — ≤ 5 gpm total	In accordance with the Inservice Testing Program <u>AND</u> Once prior to entering MODE 2 whenever the unit has been in MODE 5 for 7 days or more, if leakage testing has not been performed in the previous 9 months
SR 3.4.14.2	Verify DHR System autoclosure interlock prevents the valves from being opened with a simulated or actual high RCS pressure signal.	18 months
SR 3.4.14.3	Verify DHR System autoclosure interlock causes the valves to close automatically with a simulated or actual high RCS pressure signal: a. ≤ 340 psig for one valve; and b. ≤ 400 psig for the other valve.	18 months
SR 3.4.14.4	Verify DHR System autoclosure interlock prevents the valves from being opened with a simulated or actual Core Flood Tank isolation valve "not closed" signal.	18 months
SR 3.4.14.5	Verify DHR System autoclosure interlock causes the valves to close automatically with a simulated or actual Core Flood Tank isolation valve "not closed" signal.	18 months

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.15 RCS Leakage Detection Instrumentation

LCO 3.4.15 The following RCS leakage detection instrumentation shall be OPERABLE:

a. One reactor building sump monitor; and

b. One reactor building atmosphere radioactivity monitor (gaseous or particulate).

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTIONS

-----NOTE-----
LCO 3.0.4 is not applicable.

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. Required reactor building sump monitor inoperable.	A.1	-----NOTE----- Not required until 12 hours after establishment of steady state operation at or near operating pressure. --------------	
		Perform SR 3.4.13.1.	Once per 24 hours
	<u>AND</u>		
	A.2	Restore required reactor building sump monitor to OPERABLE status.	30 days

CONDITION	REQUIRED ACTION	COMPLETION TIME
B. Required reactor building atmosphere radioactivity monitor inoperable.	B.1.1 Analyze grab samples of the reactor building atmosphere.	Once per 24 hours
	<u>OR</u>	
	B.1.2 --------------NOTE-------------- Not required until 12 hours after establishment of steady state operation at or near operating pressure. ------------------------------------ Perform SR 3.4.13.1.	Once per 24 hours
	<u>AND</u>	
	B.2 Restore required reactor building atmosphere radioactivity monitor to OPERABLE status.	30 days
C. Required Action and associated Completion Time not met.	C.1 Be in MODE 3.	6 hours
	<u>AND</u>	
	C.2 Be in MODE 5.	36 hours
D. Both required monitors inoperable.	D.1 Enter LCO 3.0.3.	Immediately

SURVEILLANCE REQUIREMENTS

	SURVEILLANCE	FREQUENCY
SR 3.4.15.1	Perform CHANNEL CHECK of required reactor building atmosphere radioactivity monitor.	12 hours
SR 3.4.15.2	Perform CHANNEL FUNCTIONAL TEST of required reactor building atmosphere radioactivity monitor.	92 days
SR 3.4.15.3	Perform CHANNEL CALIBRATION of required reactor building atmosphere radioactivity monitor.	18 months
SR 3.4.15.4	Perform CHANNEL CALIBRATION of required reactor building sump monitor.	18 months

B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.9 Pressurizer

BASES

BACKGROUND

The pressurizer provides a point in the RCS where liquid and vapor are maintained in equilibrium under saturated conditions for pressure control purposes to prevent bulk boiling in the remainder of the RCS. Key functions include maintaining required primary system pressure during steady state operation and limiting the pressure changes caused by reactor coolant thermal expansion and contraction during normal load transients.

The pressure control components addressed by this LCO include the pressurizer water level, the required heaters, and their controls. Pressurizer safety valves are addressed by LCO 3.4.10, "Pressurizer Safety Valves."

The maximum water level limit has been established to ensure that a liquid to vapor interface exists to permit RCS pressure control during normal operation and proper pressure response for abnormalities. The water level limit thus serves two purposes:

a. Provides pressure control during normal operation; and

b. Prevents the peak RCS pressure from exceeding the safety limit of 2750 psig during an abnormality.

The maximum water level limit thus permits pressure control equipment to function as designed. The limit preserves the steam space during normal operation, so that both sprays and heaters can operate to maintain the design operating pressure. The level limit also prevents filling the pressurizer (water solid) during abnormalities, thus ensuring that pressure relief devices (electromatic relief valve (ERV) or code safety valves) can control pressure by steam relief rather than water relief. If the level limits were exceeded prior to an abnormality that creates a large pressurizer insurge volume leading to water relief, the maximum RCS pressure might exceed the design Safety Limit (SL) of 2750 psig or damage may occur to the ERV or pressurizer code safety valves.

The minimum water level limit has been established to ensure that water level is above the minimum detectable level.

The pressurizer heaters are used to maintain a pressure in the RCS so reactor coolant in the loops is subcooled and thus in the preferred state for heat transport to the steam generators (SGs). This function must be maintained with a loss of offsite power. Consequently, the emphasis of this LCO is to ensure that the

Engineered Safeguards (ES) bus powered heaters are adequate to maintain pressure for RCS loop subcooling with an extended loss of offsite power.

A minimum required available capacity of 126 kW ensures that the RCS pressure can be maintained. Unless adequate heater capacity is available, reactor coolant subcooling may not be maintained (although the pressure control provided by the high head high pressure injection pumps is an alternate method of maintaining subcooling). Inability to control the system pressure and maintain subcooling under conditions of natural circulation flow in the primary system could lead to loss of single phase natural circulation and decreased capability to remove core decay heat.

APPLICABLE SAFETY ANALYSES

In MODES 1 and 2, the LCO requirement for a steam bubble is reflected implicitly in the accident analyses. No safety analyses are performed in lower MODES. All analyses performed from a critical reactor condition assume the existence of a steam bubble and saturated conditions in the pressurizer. In making this assumption, the analyses neglect the small fraction of noncondensible gases normally present.

Safety analyses presented in the SAR do not take credit for pressurizer heater operation; however, an implicit initial condition assumption of the safety analyses is that the RCS is operating at normal pressure.

The maximum level limit is provided to prevent the peak RCS pressure from exceeding the safety limit of 2750 psig in the event of a rod withdrawal accident or a startup accident. Assuming proper response by reactor protection systems, the level limit prevents water relief through the pressurizer safety valves. If the level limits were exceeded prior to an abnormality that creates a large pressurizer insurge volume leading to water relief, the maximum RCS pressure might exceed the design SL of 2750 psig or damage may occur to the ERV or pressurizer code safety valves. The value for pressurizer level is the safety analysis value. Therefore, the implementing procedures must contain allowances for instrument error.

The requirement for emergency power supplies is based on NUREG-0578 (Ref. 1), item 2.1.1. The intent is to maintain the reactor coolant in a subcooled condition with natural circulation at hot, high pressure conditions for an extended time period after a loss of offsite power. While loss of offsite power is an initial condition or coincident event assumed in many accident analyses, maintaining hot, high pressure conditions over an extended time period is not evaluated as part of SAR accident analyses.

In MODES 1 and 2, the maximum pressurizer water level limit satisfies Criterion 2 of 10 CFR 50.36 (Ref. 2). In MODE 3 and MODE 4 above the LTOP enable temperature, the maximum pressurizer water level limit satisfies Criterion 4 of 10 CFR 50.36. Although the heaters are not specifically used in accident analysis, the need to maintain subcooling in the long term during loss of offsite power, as

indicated in NUREG-0578 (Ref. 1), is the reason for providing an LCO. Therefore, the pressurizer heaters satisfy Criterion 4 of 10 CFR 50.36 (Ref. 2).

LCO

The LCO requirement for the pressurizer to be OPERABLE with a water level ≥ 45 inches and ≤ 320 inches ensures that a steam bubble exists prior to criticality and that the indication of the level is above the minimum detectable level. Limiting the maximum operating water level preserves the steam space for pressure control. The LCO has been established to ensure the capability to establish and maintain pressure control for steady state operation and to minimize the consequences of potential overpressure transients. Requiring the presence of a steam bubble is also consistent with analytical assumptions.

The LCO requires a minimum of 126 kW of pressurizer heaters OPERABLE. To be considered OPERABLE, the required heaters must be powered from an ES bus. This provides assurance that sufficient heater capacity is available to provide RCS pressure control during a loss of off-site power. The amount needed to maintain pressure is dependent on the insulation losses, which can vary due to tightness of fit and condition.

APPLICABILITY

The need for pressure control is most pertinent when core heat can cause the greatest effect on RCS temperature, resulting in the greatest effect on pressurizer level and RCS pressure control. Thus Applicability has been designated for MODES 1 and 2. The Applicability is also provided for MODE 3 and, for pressurizer water level, for MODE 4 with RCS temperature > 262°F. The purpose is to prevent water solid RCS operation during heatup and cooldown to avoid rapid pressure rises caused by normal operational perturbations, such as reactor coolant pump startup. The temperature of 262°F has been designated as the cutoff for applicability because LCO 3.4.11, "Low Temperature Overpressure Protection (LTOP)," provides a requirement for pressurizer level at or below 262°F. The LCO does not apply to MODE 5 with loops filled because LCO 3.4.11 applies and provides adequate overpressure protection. This parameter value does not contain allowances for instrument uncertainty. Additional allowances for instrument uncertainty are contained in the implementing procedures. The LCO does not apply to MODES 5 and 6 with partial loop operation.

In MODES 1, 2, and 3, there is the need to maintain the availability of pressurizer heaters capable of being powered from an emergency power supply. In the event of a loss of offsite power, the initial conditions of these MODES give the greatest demand for maintaining the RCS in a hot pressurized condition with loop subcooling for an extended period. The Applicability is modified by a Note stating that the OPERABILITY requirements on pressurizer heaters do not apply in MODE 4. For MODE 4, 5, or 6, the need to control pressure (by heaters) to ensure loop

subcooling for heat transfer is significantly reduced when the Decay Heat Removal System is in service, and therefore the LCO is not applicable.

ACTIONS

A.1

With pressurizer water level outside the limits, action must be taken to restore pressurizer operation to within the bounds assumed in the analysis. This is done by restoring the pressurizer water level to within the limits. The 1 hour Completion Time is considered to be a reasonable time for adjusting pressurizer level.

B.1 and B.2

If the water level cannot be restored, reducing core power constrains heat input effects that drive pressurizer insurge that could result from an anticipated transient. By shutting down the reactor and reducing reactor coolant temperature to at least MODE 3, the potential thermal energy of the reactor coolant mass for mass and energy releases is reduced.

Six hours is a reasonable time based upon operating experience to reach MODE 3 from full power in an orderly manner and without challenging unit systems. Further pressure and temperature reduction to MODE 4 with RCS temperature ≤ 262°F places the unit into a MODE where the LCO is not applicable. The 24 hour Completion Time to reach the nonapplicable MODE is reasonable based upon operating experience.

C.1

If the required pressurizer heaters are inoperable, restoration is required in 72 hours. The Completion Time of 72 hours is reasonable considering the anticipation that a demand caused by loss of offsite power will not occur in this period. Pressure control may be maintained during this time using non-ES bus powered heaters.

D.1 and D.2

If the Required Action and associated Completion Time are not met, the unit must be brought to a MODE in which the LCO does not apply. To achieve this status, the unit must be brought to MODE 3 within 6 hours and to MODE 4 within the following 6 hours. The Completion Time of 6 hours is reasonable, based on operating experience, to reach MODE 3 from full power conditions in an orderly manner and without challenging unit systems. Similarly, the Completion Time of 12 hours to reach MODE 4 is reasonable based on operating experience to achieve power

reduction from full power conditions in an orderly manner and without challenging unit systems.

SURVEILLANCE REQUIREMENTS

SR 3.4.9.1

This SR requires that pressurizer water level is maintained below the upper limit to provide a minimum space for a steam bubble. The values specified for pressurizer level do not contain an allowance for instrument error. Therefore, additional allowances for instrument uncertainties must be provided in the implementing procedures. The 12 hour interval has been shown by operating practice to be sufficient to regularly assess the level for any deviation and verify that operation is within safety analyses assumptions. Alarms are also available for early detection of abnormal level.

SR 3.4.9.2

The SR requires sufficient pressurizer heaters which are connected to an ES bus verified to be capable of providing the required capacity. (This may be done by testing the power supply output and by performing an electrical check on heater element continuity and resistance.) The Frequency of 18 months is considered adequate to detect heater degradation and has been shown by operating experience to be acceptable.

REFERENCES

1. NUREG-0578, "TMI-2 Lessons Learned Task Force Status Report and Short-Term Recommendations," July 1979.

2. 10 CFR 50.36.

B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.10 Pressurizer Safety Valves

BASES

BACKGROUND

The purpose of the two spring loaded pressurizer safety valves is to provide RCS overpressure protection (Ref. 1). Operating in conjunction with the Reactor Protection System (RPS), two valves are used to ensure that the Safety Limit (SL) of 2750 psig is not exceeded for analyzed transients during operation in MODES 1 and 2. One safety valve is required for MODE 3 and portions of MODE 4. For the remainder of MODE 4, MODE 5, and MODE 6 with the reactor head on, overpressure protection is provided by operating procedures and LCO 3.4.11, "Low Temperature Overpressure Protection (LTOP)."

The self actuated pressurizer safety valves are designed in accordance with the requirements set forth in the ASME Boiler and Pressure Vessel Code, Section III (Ref. 2). The required lift pressure is 2500 psig + 1%, - 3%. The safety valves discharge steam from the pressurizer to a quench tank located in the reactor building. The discharge flow is indicated by acoustic flow monitoring devices, by an increase in temperature downstream of the safety valves, and by an increase in the quench tank temperature, pressure, and level.

The upper and lower as-left pressure limits are based on the ± 1% tolerance requirement for lifting pressures above 1000 psig. The lift setting is for the ambient conditions associated with MODES 1, 2, and 3. This requires either that the valves be set hot or that a correlation between hot and cold settings be established.

The pressurizer safety valves are part of the primary success path and mitigate the effects of postulated accidents. OPERABILITY of the safety valves ensures that the RCS pressure will be limited to 110% of design pressure. The consequences of exceeding the ASME pressure limit could include damage to RCS components, increased leakage, or a requirement to perform additional stress analyses prior to resumption of reactor operation.

APPLICABLE SAFETY ANALYSES

The overpressure protection analysis (Ref. 3) is based on operation of both safety valves and assumes that the valves open at the high range of the setting (2500 psig system design pressure plus 1%). One pressurizer code safety valve is capable of preventing overpressurization in MODE 3 and in MODE 4 with RCS temperature > 262°F since its relieving capacity is greater than that required by the sum of the available heat sources, i.e., pump energy, pressurizer heaters, and reactor decay heat (Ref. 1 and 4). These valves must accommodate pressurizer insurges that could occur during a startup, rod withdrawal, or ejected rod event. The startup

accident establishes the minimum safety valve capacity. The startup accident is assumed to occur at low power. Single failure of a safety valve is neither assumed in the accident analysis nor required to be addressed by the ASME Code. Compliance with this Specification is required to ensure that the accident analysis and design basis calculations remain valid.

In MODES 1 and 2, pressurizer safety valves satisfy Criterion 3 of the 10 CFR 50.36 (Ref. 5). In MODE 3 and MODE 4 above the LTOP enable temperature, the pressurizer safety valves satisfy Criterion 4 of 10 CFR 50.36.

LCO

The two pressurizer safety valves are set to open at the RCS design pressure (2500 psig) and within the specified tolerance to avoid exceeding the maximum RCS design pressure SL, to maintain accident analysis assumptions and to comply with ASME Code requirements. The upper and lower as-left pressure tolerance limits are based on the ± 1% tolerance requirements (Ref. 2) for lifting pressures above 1000 psig. The limit protected by this Specification is the reactor coolant pressure boundary (RCPB) SL of 110% of design pressure. Inoperability of one or both valves could result in exceeding the SL if a transient were to occur.

The consequences of exceeding the ASME pressure limit could include damage to one or more RCS components, increased leakage, or additional stress analysis being required prior to resumption of reactor operation.

The LCO is modified by four Notes. Note 1 states that in MODE 3 and MODE 4 with RCS temperature above 262°F, only one pressurizer safety valve is required to be OPERABLE. In this condition, one pressurizer safety valve is capable of preventing overpressurization when the reactor is not critical since its relieving capacity is greater than the sum of the available heat sources.

Note 2 allows entry into MODE 3, and into MODE 4 with RCS temperature > 262°F, with the lift settings potentially outside the limits. This permits testing of the safety valves at high pressure and temperature near their normal operating range, but only after the valves have had a preliminary cold setting. The cold setting gives assurance that the valves are OPERABLE near their design condition. Only one valve at a time will be removed from service for testing. The 36 hour exception is based on an 18 hour outage time for each of the two valves. The 18 hour period is derived from operating experience that hot testing can be performed in this timeframe.

Note 3 states that the LCO is not applicable in MODE 3, and in MODE 4 with RCS temperature > 262°F during hydrostatic tests in accordance with ASME Boiler and Pressure Vessel Code, Section III. During hydrostatic tests, the code safeties must be gagged to prevent them from relieving at the target test pressure. RCS pressure is carefully observed and compensatory measures are in place to provide assurance that the pressure is appropriately controlled during the performance of hydrostatic tests.

Note 4 states that the provisions of LCO 3.0.3 are not applicable in MODE 3, and in MODE 4 with RCS temperature > 262°F. In the event no code safety valve is OPERABLE in this MODE, the Required Actions ensure that the RCS is placed in a condition in which the ERV is capable of relieving any potential LTOP pressure transient.

The parameter value (262°F) does not contain allowances for instrument uncertainty. Additional allowances for instrument uncertainty are contained in the implementing procedures.

APPLICABILITY

In MODES 1, 2, and 3, and portions of MODE 4 above the LTOP enable temperature, OPERABILITY of pressurizer safety valve(s) is required to ensure adequate relieving capacity is available to keep reactor coolant pressure below 110% of its design value during certain accidents.

The LCO is not applicable in MODE 4 with RCS temperature ≤ 262°F, in MODE 5, nor in MODE 6 when the reactor vessel head is on because LTOP protection is provided. Overpressure protection is not required in MODE 6 with the reactor vessel head removed.

The parameter value (262°F) does not contain allowances for instrument uncertainty. Additional allowances for instrument uncertainty are contained in the implementing procedures.

ACTIONS

A.1

With one pressurizer safety valve inoperable in MODES 1 and 2, restoration must take place within 15 minutes. The Completion Time of 15 minutes reflects the importance of maintaining the RCS overpressure protection system. An inoperable safety valve coincident with an RCS overpressure event could challenge the integrity of the RCPB.

B.1

If the Required Action and associated Completion Time of Condition A are not met, or if both pressurizer safety valves are inoperable in MODES 1 and 2, the unit must be brought to a MODE in which the requirement does not apply. To achieve this status, the unit must be brought to at least MODE 3 within 6 hours. The 6 hours allowed is reasonable, based on operating experience, to reach MODE 3 from full power conditions in an orderly manner and without challenging unit systems. The

change from MODE 1, 2, or 3 to MODE 4 reduces the RCS energy (core power and pressure), lowers the potential for large pressurizer insurges, and thereby removes the need for overpressure protection by two pressurizer safety valves.

C.1

With the required pressurizer code safety valve inoperable, the RCS overpressure protection capability is significantly reduced and an overpressure event could challenge the integrity of the RCPB. Therefore, the unit must be placed in a condition in which the requirement does not apply. To achieve this status, the unit must be brought to at least MODE 4 with RCS temperature at or below the LTOP enable temperature within 18 hours. The 18 hours allowed is reasonable, based on operating experience, to reach a low temperature within MODE 4 without challenging unit systems. With RCS temperature at or below 262°F, overpressure protection is provided by LTOP.

SURVEILLANCE REQUIREMENTS

SR 3.4.10.1

SRs are specified in the Inservice Testing Program. Pressurizer safety valves are to be tested in accordance with the requirements of Section XI of the ASME Code (Ref. 6), which provides the activities and the Frequency necessary to satisfy the SRs. No additional requirements are specified.

The pressurizer safety valve setpoint is + 1%, - 3% for OPERABILITY (Ref. 7); however, the valves are reset to ± 1% during the Surveillance to allow for drift.

REFERENCES

1. SAR, Section 4.2.4.
2. ASME, Boiler and Pressure Vessel Code, Section III, Article 9, Summer 1968.
3. SAR, Section 4.3.8.
4. SAR, Section 4.3.11.4.
5. 10 CFR 50.36.
6. ASME, Boiler and Pressure Vessel Code, Section XI.
7. ASME/ANSI, Operations and Maintenance Codes (OM), Part 10, 1987, Part 10 Addenda, 1988, and Part 1, 1987.

B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.11 Low Temperature Overpressure Protection (LTOP) System

BASES

BACKGROUND

The LTOP controls prevent RCS overpressure at low temperatures so the integrity of the reactor coolant pressure boundary (RCPB) is not compromised by violating the pressure and temperature (P/T) requirements of 10 CFR 50, Appendix G (Ref. 1) as modified by approved exemptions. The reactor vessel is the limiting RCPB component requiring such protection. LCO 3.4.3, "RCS Pressure and Temperature (P/T) Limits," provides the allowable combinations for pressure and temperature during cooldown, shutdown, and heatup to keep from violating the Reference 1 limits.

The reactor vessel material is less tough at reduced temperatures than at normal operating temperature. Also, as vessel neutron irradiation accumulates, the material becomes less resistant to pressure stress at low temperatures (Ref. 2). RCS pressure must be maintained low when temperature is low and may be increased only as temperature is increased.

Operational maneuvering during cooldown, heatup, or any anticipated operational occurrence must be controlled to not violate LCO 3.4.3. Exceeding these limits could lead to brittle fracture of the reactor vessel. LCO 3.4.3 presents requirements for administrative control of RCS pressure and temperature to prevent exceeding the P/T limits.

This LCO provides RCS overpressure protection in the applicable MODES by ensuring an adequate pressure relief capacity and a minimum coolant addition capability. The pressure relief capacity requires the (power operated) electromatic relief valve (ERV) to be OPERABLE with the lift setpoint reduced and pressurizer coolant level at or below a maximum limit for the RCS pressure, or the RCS depressurized and with an RCS vent of sufficient size to handle the limiting LTOP transient.

The LTOP approach to protecting the vessel by limiting coolant addition capability requires deactivating HPI, and isolating the core flood tanks (CFTs).
Should an HPI pump inject on an HPI actuation, the pressurizer level and ERV or another RCS vent may not prevent overpressurizing the RCS. As indicated in Reference 3, the deactivation of HPI injection capability, along with the LTOP alarms, provides sufficient basis for excluding the inadvertent actuation of HPI as a design basis event. Additionally, the CFT controls preclude the inadvertent mass input from the CFT. Finally, maintaining the pressurizer level to prevent operation in a water solid condition with the RCS pressure boundary intact provides a compressible vapor space or cushion (either steam or nitrogen) that can accommodate a coolant insurge and prevent a rapid pressure increase, allowing the

operator time to stop the increase. The ERV, with reduced lift setting, or the RCS vent is the overpressure protection device that acts as backup to the operator in terminating an increasing pressure event.

With HPI deactivated, the ability to provide RCS coolant addition is restricted. To allow for coolant addition, the LCO does not require the makeup function to be deactivated. Due to the lower pressures associated with the LTOP MODES and the expected decay heat levels, the makeup function can provide flow through the makeup control valve.

ERV Requirements

As designed for the LTOP, the ERV is signaled to open if the RCS pressure reaches a limit set in the LTOP actuation circuit. The LTOP actuation circuit monitors RCS pressure and determines when an overpressure condition is approached. When the monitored pressure meets or exceeds the setting, the ERV is signaled to open. Maintaining the lowered setpoint ensures the Reference 1 limits will be met in any event analyzed for LTOP.

RCS Vent Requirements

Once the RCS is depressurized, adequate pressure relief capability may be provided by a vent path to the reactor building atmosphere which is capable of relieving the flow of the limiting LTOP transient and maintaining pressure below P/T limits. The required vent capacity may be provided by one or more vent paths. Acceptable RCS vent paths include any of the following: removing a pressurizer safety valve, locking the ERV in the open position and disabling its block valve in the open position, or similarly establishing a vent by removing a steam generator (SG) primary manway, removing a SG primary hand hole cover, removing all control rod drive top closure assemblies (excluding reactor vessel level probe), or removing a pressurizer manway. The vent path(s) must be above the level of reactor coolant, so as not to drain the RCS when open.

APPLICABLE SAFETY ANALYSES

Safety analyses (Refs. 4, 5, 6, and 7) demonstrate that the reactor vessel can be adequately protected against overpressurization transients during shutdown. The pressure and temperature limits are derived from fracture mechanics analyses. Transients are then evaluated to determine a required ERV setpoint and other unit conditions that will ensure that the P/T limits are not exceeded.

Fracture mechanics analyses (using the safety margins of Reference 8) established the temperature of LTOP Applicability at 262°F. Above this temperature, the pressurizer safety valves provide the reactor vessel overpressure protection. The actual temperature at which the allowable pressure falls below the pressurizer safety valve setpoint increases as vessel material ductility decreases due to neutron embrittlement. P/T limits are periodically determined using neutron fluence projections and the results of examinations of the reactor vessel material irradiation surveillance specimens. The Bases for LCO 3.4.3 discuss these examinations. For

the current limits, vessel materials are assumed to have a neutron irradiation accumulation equivalent to 31 effective full power years (EFPYs) of operation. Each time the P/T limit curves are revised, the LTOP is re-evaluated to ensure that its functional requirements can still be met. The ERV setpoint is revised if necessary.

Transients that are capable of overpressurizing the RCS at low temperature result in either excessive mass input or excessive heat input. Such transients include: HPI actuation, CFT discharge, energization of the pressurizer heaters, failing the makeup control valve open, loss of decay heat removal, starting a reactor coolant pump (RCP) with a large temperature mismatch between the primary and secondary coolant systems, and addition of nitrogen to the pressurizer. Without controls, HPI actuation and CFT discharge would be transients that result in exceeding P/T limits within the 10 minute period in which time no operator action can be assumed to take place. For the remaining events, operator action after that time precludes overpressurization.

This specification prevents exceeding the P/T limits by: 1) limiting the capability for rapid mass input to the RCS; and 2) ensuring that adequate vent capability exists to accommodate inadvertent mass or energy addition to the RCS. Pressurizer level is also limited to ensure that increasing pressure during a transient will be slow enough to preclude exceeding pressure limits within the 10 minutes assumed to be required for operator action to mitigate the transient. Mass input into the system is limited by disabling HPI (with specific exceptions) and by deactivating pressurized CFT discharge isolation valves in the closed position with their power breakers open (with specific exceptions). The analyses demonstrate that HPI transients involving one HPI pump can be accommodated by the ERV without exceeding the maximum allowable pressure.

The ERV setpoint is determined by modeling LTOP performance assuming the most limiting LTOP transient of a makeup control valve failing open. Pressure overshoot beyond the setpoint resulting from signal processing and valve stroke times is considered. The resulting ERV setpoint ensures the reference 1 limits will not be exceeded.

Vent capability is required to ensure that the maximum allowable pressure is not exceeded in the event of full opening of the makeup control valve while one makeup pump is running. Acceptable vent paths have adequate capacity at a system pressure of 100 psig which is less than the maximum RCS pressure on the P/T limit curve in LCO 3.4.3.

The ERV is an active component. Therefore, its failure represents the worst case single active failure of LTOP features. The other vent paths are passive and not subject to active failure.

The LTOP satisfies Criterion 2 of 10 CFR 50.36 (Ref. 9).

LCO

The LCO requires an LTOP system OPERABLE with a limited coolant input capability and a pressure relief capability. To limit coolant input, the LCO requires the HPI deactivated, and the CFT discharge isolation valves closed and deactivated. For pressure relief, the LCO requires the pressurizer coolant level to be below a level which represents a water solid condition, and the ERV OPERABLE with a lowered lift setting or the RCS depressurized and a vent established.

HPI deactivation requires that the motor operated valves be closed and the opening control circuits for the motor operators disabled. CFT isolation requires the CFT discharge valves to be closed and the circuit breakers for the motor operators to be opened.

The HPI deactivation and CFT isolation requirements are modified by five Notes. Note 1 indicates that the requirements are not applicable during ASME Section XI testing. This exception provides for required testing during these shutdown conditions rather than at power when the HPI and CFTs are required to be OPERABLE for the ECCS function. Note 2 indicates that the requirements are not applicable for the HPI deactivation during fill and vent of the RCS. The HPI pumps are used for this normal makeup function and must be available. Specific procedural controls are provided to prevent overpressurization during this activity. Note 3 indicates that the requirements are not applicable for the HPI deactivation during emergency RCS makeup. This exception is necessary to enhance the response capability to a loss of decay heat removal event without violating the TS (Ref. 10). Note 4 indicates that the requirements are not applicable for the HPI deactivation during valve maintenance. This exception allows maintenance to be performed during these shutdown conditions rather than at power when the HPI is required to be OPERABLE for the ECCS function. Note 5 states that CFT isolation is only required when CFT pressure is more than or equal to the maximum RCS pressure for the existing RCS temperature, as allowed in LCO 3.4.3. This is acceptable since the CFT can not be the source of an overpressurization event when its pressure is less than the allowable RCS pressure.

The pressurizer is considered to represent a water solid condition when coolant level is > 105 inches, when RCS pressure is > 100 psig, or > 150 inches, when RCS pressure is ≤ 100 psig. Although a vapor space still exists with pressurizer level above these values, from an analytical point of view, the unit is considered to be water solid. These parameter values contain allowances for instrument error.

The pressurizer level requirements are modified by two Notes. Note 1 indicates that the requirements are not applicable during operation allowed by the Emergency Operating Procedures (EOPs). This exception provides for use of the "feed and bleed" process when necessary as determined by the EOPs. Note 2 indicates that the requirements are not applicable during RCS hydrotesting. Specific procedural controls are provided to prevent overpressurization during this activity.

OPERABLE pressure relief capability may be provided by an OPERABLE ERV, or by depressurizing the RCS and providing an alternate RCS vent path. For the ERV to be considered OPERABLE, its block valve must be open, its lift setpoint must be

set at ≤ 460 psig, testing must have proven its ability to open at that setpoint, and motive power must be available to the ERV and its control circuits. With the RCS depressurized, acceptable alternate vent paths include removing a pressurizer safety valve, locking the ERV in the open position and disabling its block valve in the open position, removing a SG primary manway, removing a SG primary hand hole cover, removing all control rod drive top closure assemblies (excluding reactor vessel level probe), or removing a pressurizer manway.

APPLICABILITY

This LCO is applicable in MODE 4 with RCS temperature ≤ 262°F, in MODE 5, and in MODE 6 when the reactor vessel head is on. The Applicability temperature of 262°F is established by fracture mechanics analyses. The pressurizer safety valves provide overpressure protection to meet LCO 3.4.3 P/T limits above 262°F. With the vessel head off, overpressurization is not possible.

LCO 3.4.3 provides the operational P/T limits for all MODES. LCO 3.4.10, "Pressurizer Safety Valves," requires the pressurizer safety valves OPERABLE to provide overpressure protection during MODES 1, 2, and 3, and MODE 4 above 262°F.

The parameter value (262°F) does not contain allowances for instrument uncertainty. Additional allowances for instrument uncertainty are contained in the implementing procedures.

ACTIONS

A.1, B.1, and B.2

With the pressurizer level not within its required limits, the time for operator action in a pressure increasing event is reduced. The postulated event most affected in the LTOP MODES is failure of the makeup control valve, which fills the pressurizer relatively rapidly. Restoration is required within 1 hour.

If restoration within 1 hour in either case cannot be accomplished, Required Actions B.1 and B.2 must be performed within 12 hours to close the makeup control valve and its isolation valve. These Required Actions limit the makeup capability, which is not required with a high pressurizer level, and permit cooldown and depressurization to continue. Heatup must be stopped because heat addition decreases the reactor coolant density and increases the pressurizer level.

The Completion Times again are based on operating experience that these activities can be accomplished in these time periods and that a limiting LTOP transient is not likely in the allowed times.

C.1 and D.1

With the required ERV inoperable, overpressure relieving capability is lost, and restoration of the ERV within 1 hour is required. If that cannot be accomplished, the ability of the Makeup System to add water must be limited within the next 12 hours.

If restoration cannot be completed within 1 hour, Required Action D.1 must be performed to limit RCS water addition capability. Makeup is not deactivated to maintain the RCS coolant level. Required Action D.1 requires reducing the makeup tank level to ≤ 73 inches. This makes the available makeup water volume insufficient to exceed the LTOP limit by a makeup control valve full opening (Ref. 3). This parameter value does contain allowances for instrument error. No additional allowances for instrument error are required in the implementing procedures.

These Completion Times also consider these activities can be accomplished in these time periods. A limiting LTOP event is not likely in those times.

Some ERV testing or maintenance can only be performed at unit shutdown. Such activity is permitted if Required Action D.1 is taken to compensate for required ERV unavailability.

E.1

With the LTOP requirements not met for any reason other than cited in Condition A through D, action must be initiated to restore compliance immediately. The immediate Completion Time reflects the urgency of quickly proceeding with the Required Actions.

SURVEILLANCE REQUIREMENTS

SR 3.4.11.1

Verification of the pressurizer level at ≤ 105 inches when RCS pressure is > 100 psig or ≤ 150 inches when RCS pressure is ≤ 100 psig, by observing control room or other indications ensures that the unit is not in a water solid condition and that a cushion of sufficient size is available to reduce the rate of pressure increase from potential transients (Ref. 3).

The 30 minute Surveillance Frequency during heatup and cooldown must be performed for the LCO Applicability period when temperature changes can cause pressurizer level variations. This Frequency may be discontinued when these evolutions are complete, as defined in unit procedures. Thereafter, the Surveillance is required at 12 hour intervals.

These Frequencies are shown by operating practice sufficient to regularly assess indications of potential degradation and verify operation within the safety analysis.

SR 3.4.11.2 and SR 3.4.11.3

Verifications must be performed that the HPI is deactivated, and each pressurized CFT is isolated. These Surveillances ensure the minimum coolant input capability will not create an RCS overpressure condition to challenge the LTOP . The Surveillances are required at 12 hour intervals.

The 12 hour intervals are shown by operating practice to be sufficient to assess coolant input capability and verify operation within the safety analysis.

SR 3.4.11.4

OPERABLE pressure relief capability must be provided to prevent overpressurization due to inadvertent full makeup system operation. Such a vent keeps the pressure from full makeup flow within the LCO limit. OPERABLE pressure relief capability may be provided by an OPERABLE ERV, or by depressurizing the RCS and providing an alternate RCS vent path.

For the ERV to be considered OPERABLE, its block valve must be open, its lift setpoint must be set at ≤ 460 psig, testing must have proven its ability to open at that setpoint, and motive power must be available to the two valves and their control circuits. The parameter value of 460 psig does not contain allowances for instrument uncertainty. Additional allowances for instrument uncertainty are contained in the implementing procedures.

With the RCS depressurized, acceptable alternate vent paths include: a) removing a pressurizer safety valve; b) locking the ERV in the open position and disabling its block valve in the open position; c) removing a SG primary manway; c) removing a SG primary hand hole cover; d) removing all control rod drive top closure assemblies (excluding reactor vessel level probe); and e) removing a pressurizer manway.

For a vent path not locked open, the Frequency is every 12 hours. For a locked open vent path, the required Frequency is every 31 days.

The Frequency intervals are considered adequate based on operating practice to determine adequacy of pressure relief capability and verify operation within the safety analysis.

SR 3.4.11.5

The performance of a CHANNEL CALIBRATION is required every 18 months. The CHANNEL CALIBRATION for the LTOP ERV opening logic, including the ERV setpoint, ensures that the ERV will be actuated at the appropriate RCS pressure by

verifying the accuracy of the instrument string. The calibration can only be performed in shutdown.

The 18 month Frequency considers a typical refueling cycle and industry accepted practice.

REFERENCES

1. 10 CFR 50, Appendix G.

2. Generic Letter 88-11.

3. ANO-1 LTOP Safety Evaluation Report (1CNA058302) dated May 5, 1983.

4. Response to NRC Request for Additional Information (1CAN117608) dated November 15, 1976.

5. Response to NRC Request for Additional Information (1CAN127602) dated December 3, 1976.

6. Response to NRC Request for Additional Information (1CAN037716) dated March 24, 1977.

7. ANO-1 License Amendment Request (1CAN119608), dated November 26, 1988, and Operating License Amendment 188, (1CNA039703) dated March 14, 1997.

8. ANO-1 Request for Exemption (1CAN119608), dated November 26, 1996, and Exemption from Requirements of 10 CFR 50.60, (1CNA039702) dated March 12, 1997.

9. 10 CFR 50.36.

10. ANO-1 License Amendment Request (1CAN059008), dated May 22, 1990, and Operating License Amendment 138, (1CNA119002) dated November 1, 1990.

B 3.4 REACTOR COOLANT SYSTEM

B 3.4.12 RCS Specific Activity

BASES

BACKGROUND

The Code of Federal Regulations, 10 CFR 100 (Ref. 1), specifies the maximum dose to the whole body and the thyroid an individual at the site boundary can receive for 2 hours during an accident. The limits on specific activity ensure that the doses are held to a small fraction of the 10 CFR 100 limits during analyzed transients and accidents.

The RCS specific activity LCO limits the allowable concentration level of radionuclides in the reactor coolant. The LCO limits are established to minimize the offsite radioactivity dose consequences in the event of a steam generator tube rupture (SGTR) accident.

The LCO contains specific activity limits for both DOSE EQUIVALENT I-131 and total specific activity. The allowable levels are intended to limit the 2 hour dose at the site boundary to a small fraction of the 10 CFR 100 dose guideline limits.

APPLICABLE SAFETY ANALYSES

The LCO limits on the specific activity of the reactor coolant ensure that the resulting 2 hour doses at the site boundary will not exceed a small fraction of the 10 CFR 100 dose guideline limits following an SGTR accident. The thyroid dose conversion factors used in the calculation of DOSE EQUIVALENT I-131 are identified in Section 1.1, "Definitions."

Rupture of a steam generator tube would allow primary coolant activity to enter the secondary coolant. The major portion of this activity is noble gases and would be released to the atmosphere from the condenser vacuum pump or a relief valve. Activity would continue to be released until the operator could reduce the primary system pressure below the setpoint of the secondary relief valves and could isolate the faulty steam generator. The worst credible set of circumstances is considered to be a double-ended break of a single steam generator tube, followed by isolation of the faulty steam generator within 34 minutes after the tube break. Assuming the full differential pressure across the steam generator, no more than one-quarter of the total primary coolant could be released to the secondary coolant in this period. The decay heat during this period of 1 hour for pressure reduction will generate steam in the secondary system representing less than 15 weight percent of the secondary system.

The parameters assumed in the dose analysis (Ref. 2) for the single steam generator tube failure included the following values:

1. total primary coolant volume (mass) = 5.2×10^5 lbs.

2. total secondary coolant volume (mass) = 2×10^6 lbs.

3. leakage rate from primary to secondary system = 1 gpm.

4. fission product decay heat energy for 1 hour = 1.56×10^8 BTU.

5. steam mass released to environs = 2.84×10^5 lbs.

6. primary coolant released to secondary (34 minutes) = 8.7×10^4 lbs.

7. minimum primary to secondary iodine equilibrium activity ratio = 20 to 1 (for 1 gpm leakage).

8. DOSE EQUIVALENT I-131 specific activity = 3.5 μCi/gm (Primary).

9. DOSE EQUIVALENT I-131 specific activity = 0.17 μCi/gm (Secondary).

10. total specific activity in primary = $72/\bar{E}$ μCi/gm.

11. X/Q = 7.0×10^{-4} sec/m^3 at limiting point beyond site boundary of 1046 meters for 30 m release height - equivalent to ground level release due to topography including building wake effect for 5 percentile meteorology.

12. total radioactivity in primary coolant released to secondary coolant released to environs.

13. ten percent of the combined radioiodine activity from primary activity in secondary coolant and secondary activity present in steam mass (released to environs) assumed released to environs.

The whole body dose resulting from immersion in the cloud containing the released activity would include both gamma and beta radiation. The gamma dose is dependent on the finite size and configuration of the cloud. However, the analysis employed the simple model of the semi-infinite cloud, which gives an upper limit to the potential gamma dose. The semi-infinite cloud model is applicable to the beta dose, because of the short range of beta radiation in air. The resulting whole body dose was determined to be less than 0.5 Rem for this accident.

The thyroid dose from the steam generator tube rupture accident has been analyzed assuming a tube rupture at full load and loss of offsite power at the time of the reactor trip, which results in steam release through the relief valves in the period before the faulty steam generator is isolated and primary system pressure is reduced. The limiting iodine activities for the primary and secondary systems are used in the initial conditions. One-tenth of the iodine contained in the liquid which is converted to steam and passed through the relief valves is assumed to reach the site boundary. The resulting thyroid dose from the combined primary and secondary

iodine activity released to the environs was determined to be 1.5 Rem for this accident.

The limit for secondary iodine activity is consistent with the limits on primary system iodine activity and primary-to-secondary leakage of 1 gpm. If the activity should exceed the specified limits following a power transient, the major concern would be whether additional fuel defects had developed bringing the total to above expected levels. From the observed removal of excess activity by decay and cleanup, it should be apparent whether activity is returning to a level below the specification limit. Appropriate action to be taken to bring the activity within specification include one or more of the following: gradual decrease in power to a lower base power, increase in letdown flow rate, and venting of the makeup tank gases to the waste gas decay tanks.

The analysis shows the radiological consequences of an SGTR accident are within a small fraction of the Reference 1 dose guideline limits.

RCS Specific Activity satisfies Criterion 2 of 10 CFR 50.36 (Ref. 3).

LCO

The specific iodine activity is limited to $\leq$ 3.5 μCi/gm DOSE EQUIVALENT I-131, and the total specific activity in the primary coolant is limited to the number of μCi/gm equal to 72 divided by $\bar{E}$. The limit on DOSE EQUIVALENT I-131 ensures the 2 hour thyroid dose to an individual at the site boundary during the SGTR will be a small fraction of the allowed thyroid dose. The limit on total specific activity ensures the 2 hour whole body dose to an individual at the site boundary during the SGTR will be a small fraction of the allowed whole body dose.

The analysis shows that the 2 hour site boundary dose levels are within acceptable limits. Violation of the LCO may result in reactor coolant radioactivity levels that could, in the event of an SGTR, lead to site boundary doses that exceed the 10 CFR 100 dose guideline limits.

APPLICABILITY

In MODES 1 and 2, and in MODE 3 with RCS average temperature $\geq$ 500°F, operation within the LCO limits for DOSE EQUIVALENT I-131 and total specific activity are necessary to limit the potential consequences of an SGTR to within the acceptable site boundary dose values.

For operation in MODE 3 with RCS average temperature < 500°F, and in MODES 4 and 5, the release of radioactivity in the event of an SGTR is unlikely since the saturation pressure of the reactor coolant is below the lift pressure settings of the atmospheric dump valves and main steam safety valves.

ACTIONS

A.1

With the specific activity of the reactor coolant greater than the LCO limits, the specific activity must be restored to within limits within 24 hours. The Completion Time of 24 hours is adequate to determine and implement appropriate actions to return specific activity to within limits.

B.1

If the Required Action and associated Completion Time are not met, the reactor must be brought to MODE 3 with RCS average temperature < 500°F within 6 hours. Placing the unit in MODE 3 and RCS average temperature < 500°F lowers the saturation pressure of the reactor coolant below the setpoints of the main steam safety valves, and prevents venting the SG to the environment in an SGTR event. The Completion Time of 6 hours is required to reach MODE 3 from full power conditions in an orderly manner and without challenging unit systems.

SURVEILLANCE REQUIREMENTS

SR 3.4.12.1

SR 3.4.12.1 requires performing a gamma isotopic analysis as a measure of the gross specific activity of the reactor coolant at least once per 7 days. The gross specific activity analysis consists of the quantitative measurement of the total activity of the primary coolant in units of microcuries per gram (μCi/gm). The total primary coolant activity is the sum of the degassed beta-gamma activity and the total of all identified gaseous activities 15 minutes after the primary system is sampled and any identified beta emitters (i.e., tritium, SR89, SR90, etc.). This Surveillance provides an indication of any increase in gross specific activity.

Trending the results of this Surveillance allows proper remedial action to be taken before reaching the LCO limit under normal operating conditions. The Surveillance is applicable in MODES 1 and 2, and in MODE 3 with RCS average temperature at least 500°F. The 7 day Frequency is based on the low probability of a gross fuel failure during that time period.

SR 3.4.12.2

This Surveillance is performed in MODE 1 only to ensure the iodine remains within limit during normal operation and following fast power changes when fuel failure is more apt to occur. The 14 day Frequency is adequate to trend changes in the iodine activity level considering gross specific activity is monitored every 7 days.

SR 3.4.12.3

SR 3.4.12.3 requires radiochemical analysis for $\bar{E}$ determination every 184 days. The $\bar{E}$ determination directly relates to the LCO and is required to verify plant operation within the total specific activity LCO limit. The Frequency of 184 days recognizes $\bar{E}$ does not change rapidly.

The radiochemical analysis consists of the quantitative measurement of the activity for each radionuclide which is identified in the primary coolant 15 minutes after the primary system is sampled. The activities for the individual isotopes are used in the determination of $\bar{E}$. The gamma energy per disintegration for those radioisotopes determined to be present shall be as given in "Table of Isotopes" (1967) (Ref. 4) and beta energy per disintegration shall be as given in USNRDL-TR-802 (Part II) (Ref. 5) or other references using the equivalent values for the radioisotopes. Iodine isotopic activities are weighted to give DOSE EQUIVALENT I-131 activity.

This SR is modified by a NOTE that requires the determination be performed within 31 days after a minimum of 2 EFPD and 20 days of MODE 1 operation have elapsed since the reactor was last subcritical for at least 48 hours. This ensures the radioactive materials are at equilibrium so the analysis for $\bar{E}$ is representative and not skewed by a crud burst or other similar abnormal event.

REFERENCES

1. 10 CFR 100.11.

2. ANO-1 Operating License Amendment 2, (1CNA057502) dated May 9, 1975.

3. 10 CFR 50.36.

4. "Table of Isotopes" (1967).

5. USNRDL-TR-802 (Part II).

B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.13 RCS Operational LEAKAGE

BASES

BACKGROUND

Components that contain or transport the coolant to or from the reactor core make up the RCS. During plant life, the joint and valve interfaces can produce varying amounts of reactor coolant LEAKAGE, through either normal operational wear or mechanical deterioration. The purpose of the RCS Operational LEAKAGE LCO is to limit LEAKAGE from these sources to amounts that do not compromise safe operation. This LCO specifies the types and amounts of allowable LEAKAGE.

SAR Section 1.4, GDC 30 (Ref. 1), requires means for detecting and, to the extent practical, identifying the source of reactor coolant LEAKAGE. Regulatory Guide 1.45 (Ref. 2) describes acceptable criteria for selecting Leakage Detection Systems. Reference 3 provides a comparison of the ANO-1 RCS leak detection systems to Regulatory Guide 1.45 (Ref. 2).

The safety significance of RCS LEAKAGE varies widely depending on its source, rate, and duration. Therefore, detecting and monitoring reactor coolant LEAKAGE into the reactor building are necessary.

A limited amount of leakage inside the reactor building is expected from auxiliary systems that cannot be made leaktight. Leakage from these systems should be detected, located, and isolated from the reactor building atmosphere, if possible, to not interfere with RCS leakage detection.

This LCO deals with protection of the reactor coolant pressure boundary (RCPB) from degradation. The consequences of violating this LCO include increasing the probability of a loss of coolant accident (LOCA). However, the ability to monitor leakage provides advance warning to permit unit shutdown before a LOCA occurs. This advantage has been shown by "leak before break" studies.

APPLICABLE SAFETY ANALYSES

Except for primary to secondary LEAKAGE, the safety analyses do not address operational LEAKAGE. However, other operational LEAKAGE is related to the safety analyses for LOCA; the amount of leakage can affect the probability of such an event. The safety analysis for an event resulting in steam discharge to the atmosphere assumes 1 gpm primary to secondary LEAKAGE as the initial condition.

Primary to secondary LEAKAGE is a factor in the radioactivity releases resulting from a steam line break (SLB) accident. To a lesser extent, other accidents or

transients involve secondary steam release to the atmosphere, such as a steam generator tube rupture (SGTR). The leakage contaminates the secondary fluid.

The SAR (Ref. 4) analysis for SGTR assumes the contaminated secondary fluid is released via turbine bypass valves to the condenser. The 1 gpm primary to secondary LEAKAGE is relatively inconsequential.

The safety analysis for the SLB accident assumes 1 gpm primary to secondary LEAKAGE in one generator as an initial condition. The dose consequences resulting from the SLB accident are well within the limits defined in 10 CFR 100.

RCS leakage detection capabilities and methods are identified and discussed in SAR Section 4.2.3.8 (Ref. 5) and in the Bases for LCO 3.4.15, "RCS Leakage Detection Instrumentation."

In MODES 1 and 2, RCS operational LEAKAGE satisfies Criterion 2 of 10 CFR 50.36 (Ref. 6). In MODES 3 and 4, RCS operational LEAKAGE satisfies Criterion 4 of 10 CFR 50.36.

LCO

RCS operational LEAKAGE shall be limited to:

a. Pressure Boundary LEAKAGE

No pressure boundary LEAKAGE is allowed, being indicative of material deterioration. 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.

b. Unidentified LEAKAGE

One gallon per minute (gpm) of unidentified LEAKAGE is allowed as a reasonable minimum detectable amount that the reactor building air monitoring and reactor building sump level monitoring equipment can detect within a reasonable time period. Violation of this LCO could result in continued degradation of the RCPB, if the LEAKAGE is from the pressure boundary. Controlled reactor coolant pump (RCP) seal leakoff is a normal function and is not considered as LEAKAGE.

c. Identified LEAKAGE

Up to 10 gpm of identified LEAKAGE is considered allowable because LEAKAGE is from known sources that do not interfere with detection of unidentified LEAKAGE and is well within the capability of the RCS makeup system. Identified LEAKAGE includes LEAKAGE to the reactor building from specifically known and located sources and LEAKAGE through a SG to the

secondary system, but does not include pressure boundary LEAKAGE or controlled reactor coolant pump (RCP) seal leakoff (a normal function not considered LEAKAGE). Violation of this LCO could result in continued degradation of a component or system.

d. Primary to Secondary LEAKAGE through Any One SG

The 150 gallon per day (0.104 gpm) limit on one SG is intended to assure timely shutdown of the plant for appropriate corrective action before rupture of the steam generator tube(s) occurs under normal operating or postulated accident conditions. These limits also serve to provide added assurance that the dosage contribution from tube leakage will be limited to a small fraction of 10 CFR 100 (Ref. 7) limits for a design basis steam generator tube rupture or main steam line break. Primary to secondary LEAKAGE must be included in the total allowable limit for identified LEAKAGE.

APPLICABILITY

In MODES 1, 2, 3, and 4, the LEAKAGE limits are required because the RCS is pressurized and the potential for RCPB LEAKAGE is greatest.

In MODES 5 and 6, LEAKAGE limits are not required because the reactor coolant pressure is far lower, resulting in lower stresses and reduced potential for LEAKAGE.

LCO 3.4.14, "RCS Pressure Isolation Valve (PIV) Leakage," measures leakage through RCS pressure isolation valves (PIVs) and can impact this LCO. Of the two PIVs in series in each isolated line, leakage measured through one PIV does not result in RCS LEAKAGE when the other is leaktight. If both valves in series leak and result in a loss of coolant mass from the RCS, the loss must be included in the allowable identified LEAKAGE.

ACTIONS

A.1

If primary to secondary LEAKAGE is in excess of the LCO limits, the LEAKAGE must be reduced to within limits within 4 hours. This Completion Time allows time to verify leakage rates and reduce LEAKAGE to within limits before the reactor must be shut down. This action is necessary to prevent further deterioration of the primary to secondary RCPB.

B.1

If unidentified LEAKAGE, or identified LEAKAGE, or both, are in excess of the LCO limits, the LEAKAGE must be reduced to within limits within 18 hours. This

Completion Time allows time to verify leakage rates and either identify unidentified LEAKAGE or reduce LEAKAGE to within limits before the reactor must be shut down. This action is necessary to prevent further deterioration of the RCPB.

C.1 and C.2

If any pressure boundary LEAKAGE exists or if the Required Action and associated Completion Time of Condition A or B are not met, the reactor must be brought to lower pressure conditions to reduce the severity of the LEAKAGE and its potential consequences. The reactor must be brought to MODE 3 within 6 hours and MODE 5 within 36 hours. This action reduces the LEAKAGE and also reduces the factors that tend to degrade the pressure boundary.

The Completion Times allowed are reasonable, based on operating experience, to reach the required conditions from full power conditions in an orderly manner and without challenging unit systems. In MODE 5, the pressure stresses acting on the RCPB are much lower and further deterioration is much less likely.

SURVEILLANCE REQUIREMENTS

SR 3.4.13.1

Verifying RCS LEAKAGE within the LCO limits ensures that the integrity of the RCPB is maintained. Pressure boundary LEAKAGE would at first appear as unidentified LEAKAGE and may be positively identified by inspection. Total LEAKAGE is determined by performance of an RCS water inventory balance.

The RCS water inventory balance must be performed with the reactor at steady state operating conditions (stable temperature, power level, pressurizer and makeup tank levels, makeup and letdown, and RCP seal injection and return flows) at or near operating pressure. Therefore, a Note is added allowing that this SR is not required to be performed until 12 hours after establishing steady state operation at or near operating pressure (i.e., at or near 2155 psig). The 12 hour allowance provides sufficient time to collect and process all necessary data after stable unit conditions are established.

Steady state operation is required to perform a proper water inventory balance since calculations during maneuvering are not useful. For RCS operational LEAKAGE determination by water inventory balance, steady state is defined as stable RCS pressure, temperature, power level, pressurizer and makeup tank levels, makeup and letdown, and RCP pump seal injection and return flows.

An early warning of pressure boundary LEAKAGE or unidentified LEAKAGE is provided by the systems that monitor the reactor building atmosphere radioactivity and the reactor building sump level. These leakage detection systems are specified in LCO 3.4.15, "RCS Leakage Detection Instrumentation."

The 72 hour Frequency is a reasonable interval to trend LEAKAGE and recognizes the importance of early leakage detection in the prevention of accidents.

SR 3.4.13.2

This SR provides the means necessary to determine SG OPERABILITY in an operational MODE. The requirement to demonstrate SG tube integrity in accordance with the Steam Generator Tube Surveillance Program emphasizes the importance of SG tube integrity, even though this Surveillance cannot be performed at normal operating conditions.

REFERENCES

1. SAR, Section 1.4, GDC 30.
2. Regulatory Guide 1.45, Reactor Coolant Pressure Boundary Leakage Detection Systems, May 1973.
3. Information Submittal - Comparison of ANO-1 RCS Leak Detection Systems to Regulatory Guide 1.45 (1CAN108607), dated October 14, 1986.
4. SAR, Chapter 14.
5. SAR, Section 4.2.3.8.
6. 10 CFR 50.36.
7. 10 CFR 100.

B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.14 RCS Pressure Isolation Valve (PIV) Leakage

BASES

BACKGROUND

RCS pressure isolation valves (PIVs) are identified in Reference 1 as any two normally closed valves in series within the RCS pressure boundary that separate the high pressure RCS from an attached low pressure system. During their lives, these valves can produce varying amounts of reactor coolant leakage through either normal operational wear or mechanical deterioration. The RCS PIV Leakage 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 isolation check valve which is closest to the reactor vessel in the decay heat system injection lines and to each parallel pair of check valves which protect an individual low pressure injection line (Ref. 1). Leakage through both series PIVs in a line must be included as part of the identified LEAKAGE, governed by LCO 3.4.13, "RCS Operational LEAKAGE." This is true during operation only when the loss of RCS mass through two series valves is determined by a water inventory balance (SR 3.4.13.1). A known component of the identified LEAKAGE before operation begins is the least of the two individual leakage rates determined for leaking series PIVs during the required surveillance testing; leakage measured through one PIV in a line is not RCS operational LEAKAGE if the other is leaktight.

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. Leakage exceeding the limit is an indication that the PIVs between the RCS and the connecting systems are degraded or degrading. PIV leakage could lead to overpressurization of the low pressure piping or components. Failure consequences could be a loss of coolant accident (LOCA) outside of the reactor building, an unanalyzed accident that could degrade low pressure injection capability.

The 1975 NRC "Reactor Safety Study" (Ref. 2) identified potential intersystem LOCAs as a significant contributor to the risk of core melt.

A subsequent study (Ref. 3) evaluated various PIV configurations to determine the probability of intersystem LOCAs. In 1981, PIV requirements were issued as an order for modification of the ANO-1 Operating License (Ref. 1).

PIVs are provided to isolate the RCS from the low pressure portion of the Decay Heat Removal (DHR) System.

Violation of this LCO could result in continued degradation of a PIV, which could lead to overpressurization of the DHR System and the loss of the integrity of a fission product barrier.

APPLICABLE SAFETY ANALYSES

Except for primary to secondary LEAKAGE, the safety analyses do not address operational LEAKAGE. However, other operational LEAKAGE is related to the safety analyses for LOCA; the amount of leakage can affect the probability of such an event. Reference 2 identified potential intersystem LOCAs as a significant contributor to the risk of core melt. The dominant accident sequence in the intersystem LOCA category is the failure of the low pressure portion of the DHR System outside of the reactor building. The accident is the result of a postulated failure of the PIVs, which are part of the reactor coolant pressure boundary (RCPB), and the subsequent pressurization of the DHR System. Overpressurization failure of the DHR low pressure line would result in a LOCA outside the reactor building and subsequent risk of core melt.

Reference 3 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.

RCS PIV Leakage satisfies Criterion 4 of the 10 CFR 50.36 (Ref. 4).

LCO

RCS PIV leakage is identified 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.

The LCO PIV leakage limit is 5 gpm.

Reference 5 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) in those types of valves in which the higher service pressure will tend to diminish the overall leakage channel opening. In such cases, the observed rate may be adjusted to account for the maximum pressure differential by assuming leakage is directly proportional to the square root of the pressure differential.

APPLICABILITY

In MODES 1, 2, 3, and 4, this LCO applies because the PIV leakage potential is greatest when the RCS is pressurized. In MODE 4, valves in the DHR flow path are

not required to meet the requirements of this LCO when in, or during the transition to or from, the DHR mode of operation.

In MODES 5 and 6, leakage limits are not provided because the lower reactor coolant pressure results in a reduced potential for leakage and for a LOCA outside the reactor building.

ACTIONS

The ACTIONS are modified by two Notes. Note 1 is added to provide clarification that each flow path allows separate entry into a Condition. This is allowed based upon the functional independence of the flow path. Note 2 requires an evaluation of affected systems if a PIV is inoperable.

The Required Action may have degraded the ability of the interconnected system to perform its safety function.

A.1 and A.2

The leaking flow path must be isolated by two valves. When using this automatic MOV for isolation, deactivation makes the low pressure injection subsystem of one train of the ECCS inoperable since the MOV must automatically open to provide the LPI ECCS function. The ECCS Specification will effectively limit continued operation.

Required Action A.1 requires that the isolation must be performed within 4 hours. Four hours provides time to isolate the affected system and restricts the operation with leaking isolation valves.

B.1

The inoperability of the DHR autoclosure interlock renders the DHR suction isolation valves incapable of isolating in response to a high pressure condition and preventing inadvertent opening of the valves at RCS pressures in excess of the DHR systems design pressure. If the DHR autoclosure interlock is required and inoperable, operation may continue as long as the DHR suction penetration is closed by at least one closed manual or deactivated automatic valve within 4 hours. This action accomplishes the purpose of the autoclosure function.

C.1 and C.2

If Required Actions and associated Completion Times are not met, the unit must be brought to a MODE in which the requirement does not apply.

To achieve this status, the unit must be brought to MODE 3 within 6 hours and to MODE 5 within 36 hours. This Required Action may reduce the leakage and also

reduces the potential for a LOCA outside the reactor building. The allowed Completion Times are reasonable based on operating experience, to reach the required unit conditions from full power conditions in an orderly manner and without challenging unit systems.

SURVEILLANCE REQUIREMENTS

SR 3.4.14.1

Performance of leakage testing on RCS pressure isolation check valve(s) is required to verify that leakage is below the specified limit and to identify leaking valve(s). The leakage limit of 5 gpm maximum applies to each isolation check valve which is closest to the reactor vessel in the DHR System injection lines (DH-14A and DH-14B) and to each parallel pair of check valves which protect an individual low pressure injection line (total for DH-13A and DH-17, and total for DH-13B and DH-18). Leakage testing requires a stable pressure condition. Reference 5 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) in those types of valves in which the higher service pressure will tend to diminish the overall leakage channel opening. In such cases, the observed rate may be adjusted to account for the maximum pressure differential by assuming leakage is directly proportional to the square root of the pressure differential.

If the in series PIVs are not separately 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 in series valves would be lost.

Testing is to be performed on a Frequency consistent with 10 CFR 50.55a(g) (Ref. 6) as contained in the Inservice Testing Program, and allowed by the American Society of Mechanical Engineers (ASME) Code, Section XI (Ref. 5). This Frequency is based on the need to perform such surveillances under conditions that apply during an outage and the potential for an unplanned transient if the Surveillance were performed with the unit at power.

The leakage surveillance is to be performed at the RCS pressure associated with MODES 1 and 2. This permits leakage testing at high differential pressures with stable conditions not possible in the MODES with lower pressures.

To satisfy ALARA requirements, leakage may be measured indirectly (as from the performance of pressure indicators) if accomplished in accordance with approved procedures and supported by computations showing that the method is capable of demonstrating valve compliance with the leakage criteria.

Entry into MODES 3 and 4 is allowed to establish the necessary differential pressures and stable conditions to allow for performance of this Surveillance. The Note that allows this provision is complimentary to the Frequency of prior to entry

into MODE 2 whenever the unit has been in MODE 5 for 7 days or more, if leakage testing has not been performed in the previous 9 months.

SR 3.4.14.2, SR 3.4.14.3, SR 3.4.14.4, and SR 3.4.14.5

Verifying that the DHR autoclosure interlocks are OPERABLE ensures that RCS pressure will not over pressurize the DHR system. The interlock(s) that prevent the valves from being opened and that close the valves are designed to protect the DHR System from gross overpressurization. Although the specified values include certain process measurement uncertainties, additional allowances for instrument uncertainty are contained in the implementing procedures. The 18 month Frequency is based on the need to perform this Surveillance under the conditions that apply during a plant outage and on the potential for an unplanned transient if the Surveillance was performed with the reactor at power. The 18 month Frequency is also acceptable based on consideration of the design reliability (and confirming operating experience) of the equipment.

REFERENCES

1. "Order for Modification of License Concerning Primary Coolant System Pressure Isolation Valves," issued April 20, 1981.
2. NUREG-75/014, Reactor Safety Study, Appendix V, October 1975.
3. NUREG-0677, The Probability of Intersystem LOCA: Impact Due to Leak Testing and Operational Changes, May 1980.
4. 10 CFR 50.36.
5. ASME, Boiler and Pressure Vessel Code, Section XI.
6. 10 CFR 50.55a(g).

B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.15 RCS Leakage Detection Instrumentation

BASES

BACKGROUND

SAR, Section 1.4, GDC 30 (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 criteria for selecting leakage detection systems.

Leakage detection systems must have the capability to detect significant reactor coolant pressure boundary (RCPB) degradation as soon after occurrence as practical to minimize the potential for propagation to a gross failure. Thus, an early indication is necessary to permit proper evaluation of all unidentified LEAKAGE.

Industry practice has shown that water flow changes of 0.5 to 1.0 gpm can readily be detected in contained volumes by monitoring changes in water level, in flow rate, or in the operating frequency of a pump. The reactor building sump used to collect unidentified LEAKAGE is instrumented to detect increases of 1.0 gpm in the fill rates. This sensitivity is acceptable for detecting increases in unidentified LEAKAGE.

The reactor coolant contains radioactivity that, when released to the reactor building, can be detected by radiation monitoring instrumentation. Radioactivity detection systems are included for monitoring both particulate and gaseous activities because of their sensitivities and rapid responses to RCS LEAKAGE.

Air temperature and pressure monitoring methods may also be used to infer unidentified LEAKAGE to the reactor building. Reactor building temperature and pressure fluctuate slightly during unit operation, but a rise above the normally indicated range of values may indicate RCS LEAKAGE into the reactor building. The relevance of temperature and pressure measurements are affected by reactor building free volume and, for temperature, detector location. Indications from these instruments can be valuable in recognizing rapid and sizable leakage to the reactor building. Temperature and pressure monitors are not required by this LCO.

APPLICABLE SAFETY ANALYSES

Except for primary to secondary LEAKAGE, the safety analyses do not address operational LEAKAGE. However, other operational LEAKAGE is related to the safety analyses for LOCA; the amount of leakage can affect the probability of such an event. Therefore, the need to evaluate the severity of an alarm or an indication is important to the operators, and the ability to compare and verify with indications

from other systems is necessary. The system response times and sensitivities are described in the SAR (Ref. 3).

The safety significance of RCS LEAKAGE varies widely depending on its source, rate, and duration. Therefore, detecting and monitoring reactor coolant LEAKAGE into the reactor building are necessary.

In MODES 1 and 2, RCS leakage detection instrumentation satisfies Criterion 1 of 10 CFR 50.36 (Ref. 4). In MODES 3 and 4, RCS leakage detection instrumentation satisfies Criterion 4 of 10 CFR 50.36.

LCO

One method of protecting against large RCS LEAKAGE derives from the ability of instruments to rapidly detect small leaks. This LCO requires instruments of diverse monitoring principles to be OPERABLE to provide a high degree of confidence that small leaks are detected in time to allow actions to place the unit in a safe condition when RCS LEAKAGE indicates possible RCPB degradation.

The LCO requirements are satisfied when monitors of diverse measurement means are available. Thus, the reactor building sump monitor, in combination with a particulate or gaseous radioactivity monitor, provides an acceptable minimum.

APPLICABILITY

Because of elevated RCS temperature and pressure in MODES 1, 2, 3, and 4, RCS leakage detection instrumentation is required to be OPERABLE.

In MODE 5 or 6, the temperature and pressure are maintained low. Since the temperatures and pressures are lower than those for MODES 1, 2, 3, and 4, the likelihood of leakage and crack propagation is sufficiently smaller. Therefore, the requirements of this LCO are not applicable in MODES 5 and 6.

ACTIONS

The Actions are modified by a Note indicating that the provisions of LCO 3.0.4 do not apply. As a result, a MODE change is allowed when the sump and required radiation monitors are inoperable. This allowance is provided because other instrumentation is available to monitor RCS LEAKAGE.

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

With the required reactor building sump monitor inoperable, no other form of sampling can provide the equivalent information.

However, the reactor building atmosphere activity monitor will provide indications of changes in leakage. Together with the atmosphere monitor, performing the periodic surveillance for RCS inventory balance, SR 3.4.13.1, at an increased frequency of 24 hours provides information that is adequate to detect leakage. A Note is added allowing that SR 3.4.13.1 is not required to be performed until 12 hours after establishing steady state operation (stable temperature, power level, pressurizer and makeup tank levels, makeup and letdown, and RCP seal injection and return flows) at or near operating pressure. The 12 hour allowance provides sufficient time to collect and process all necessary data after stable unit conditions are established.

Restoration of the required sump monitor to OPERABLE status is required to regain the function in a Completion Time of 30 days after the monitor's failure. This time is acceptable considering the Frequency and adequacy of the RCS water inventory balance required by Required Action A.1.

B.1.1, B.1.2, and B.2

With the required gaseous or particulate reactor building atmosphere radioactivity monitoring instrumentation channel inoperable, alternative action is required. Either grab samples of the reactor building atmosphere must be taken and analyzed or water inventory balances, in accordance with SR 3.4.13.1, must be performed to provide alternate periodic information. With a sample obtained and analyzed or a water inventory balance performed every 24 hours, the reactor may be operated for up to 30 days to allow restoration of at least one of the radioactivity monitors.

The 24 hour interval provides periodic information that is adequate to detect leakage. A Note is added allowing that SR 3.4.13.1 is not required to be performed until 12 hours after establishing steady state operation (stable temperature, power level, pressurizer and makeup tank levels, makeup and letdown, and RCP seal injection and return flows) at or near operating pressure. The 12 hour allowance provides sufficient time to collect and process all necessary data after stable unit conditions are established. The 30 day Completion Time recognizes at least one other form of leak detection is available.

C.1 and C.2

If the Required Action and associated Completion Time are not met, the unit must be brought to a MODE in which the LCO does not apply. To achieve this status, the unit must be brought to at least MODE 3 within 6 hours and to MODE 5 within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required unit conditions from full power conditions in an orderly manner and without challenging unit systems.

D.1

With both required monitors inoperable, no indicated means of monitoring leakage are available, and immediate plant shutdown in accordance with LCO 3.0.3 is required.

SURVEILLANCE REQUIREMENTS

SR 3.4.15.1

SR 3.4.15.1 requires the performance of a CHANNEL CHECK of the required reactor building atmosphere radioactivity monitor. The check gives reasonable confidence that each channel is operating properly. The Frequency of 12 hours is based on instrument reliability and is reasonable for detecting off normal conditions.

SR 3.4.15.2

SR 3.4.15.2 requires the performance of a CHANNEL FUNCTIONAL TEST of the required reactor building atmosphere radioactivity monitor. The test ensures that the monitor can perform its function in the desired manner. The test verifies the alarm function and relative accuracy of the instrument string. The Frequency of 92 days considers instrument reliability, and operating experience has shown it proper for detecting degradation.

SR 3.4.15.3 and SR 3.4.15.4

These SRs require the performance of a CHANNEL CALIBRATION for each of the required RCS leakage detection instrumentation channels. The calibration verifies the accuracy of the instrument string, including the instruments located inside the reactor building. The Frequency of 18 months is a typical refueling cycle and considers channel reliability. Additionally, operating experience has shown this Frequency is acceptable.

REFERENCES

1. SAR, Section 1.4, GDC 30.
2. Regulatory Guide 1.45, Reactor Coolant Pressure Boundary Leakage Detection Systems, May 1973.
3. SAR, Section 4.2.3.8.
4. 10 CFR 50.36.

CTS DISCUSSION OF CHANGES

ITS Section 3.4B: Reactor Coolant System

Note: The ITS Section 3.4B package includes the following ITS:

ITS 3.4.9	Pressurizer
ITS 3.4.10	Pressurizer Safety Valves
ITS 3.4.11	Low Temperature Overpressure Protection (LTOP)
ITS 3.4.12	RCS Specific Activity
ITS 3.4.13	RCS Operational Leakage
ITS 3.4.14	RCS Pressure Isolation
ITS 3.4.15	RCS Leakage Detection Instrumentation

which address the following NUREG-1430 RSTS:

RSTS 3.4.9	Pressurizer
RSTS 3.4.10	Pressurizer Safety Valves
RSTS 3.4.11	Pressurizer Power Operated Relief Valve (PORV) -- Not used
RSTS 3.4.12	Low Temperature Overpressure Protection (LTOP) System
RSTS 3.4.13	RCS Operational Leakage
RSTS 3.4.14	RCS Pressure Isolation Valve (PIV) Leakage
RSTS 3.4.15	RCS Leakage Detection Instrumentation
RSTS 3.4.16	RCS Specific Activity

ADMINISTRATIVE

A1 The designated change represents a non-technical, non-intent change to the Arkansas Nuclear One, Unit 1 Current Technical Specifications (CTS) made to make the ANO-1 Improved Technical Specifications (ITS) consistent with the Babcock and Wilcox (B&W) revised Standard Technical Specification (RSTS), NUREG-1430, Revision 1. This change does not alter the requirements of the CTS or RSTS. Examples of this type of change include: wording preference; convention adoption; editorial, numbering and formatting changes; and hierarchy structure.

A2 The ANO-1 CTS Bases will be administratively deleted in their entirety in favor of the NUREG-1430 Bases. The CTS Bases will be reviewed for technical content that will be identified for retention in the ITS Bases.

A3 The CTS 3.1.3.6 requirements for 2 out of 3 emergency powered pressurizer heater groups to be OPERABLE are revised to require that a minimum of 126 kW of pressurizer heaters be OPERABLE. Since the 2 out of 3 was specified to assure a minimum of 126 kW were available, as indicated in the CTS Bases, this is considered an administrative change consistent with NUREG-1430.

A4 The CTS 3.1.6.6 requirements which prevent reactor restart until compliance is restored are not specifically identified in ITS 3.4.13. ITS LCO 3.0.4 provides the same restrictions, therefore, specific identification of the restriction is unnecessary. This is considered an administrative change due only to application and format consistent with NUREG-1430.

A5 An explicit Surveillance Requirement (SR 3.4.13.2) is included to verify steam generator tube integrity in accordance with the Steam Generator Tube Inspection Program. Such verifications are required by CTS 4.18 which has been moved to the Administrative Controls Section of the ITS. Therefore, this SR is merely a direction to implement the program and is, therefore, considered an administrative change in format consistent with NUREG-1430.

A6 CTS Table 4.1-3, Note (11) is omitted from ITS. This Note was only applicable until the end of Cycle 2 operation which was completed in the 1970's. As such, this Note provides no current or future requirements and its omission is purely administrative.

A7 An explicit Applicability of MODES 1, 2, 3, and 4 is included in ITS 3.4.14. CTS 3.1.6.9 contains no such applicability statement but noncompliance results in the unit ultimately being placed in cold shutdown (ITS MODE 5). Therefore, the addition of this explicit Applicability statement is considered to be equivalent and a purely administrative change.

Additionally, ITS 3.4.14 ACTIONS Note 1 is included to allow separate Condition entry for each flow path. The actions required by CTS 3.1.6.9 currently allow multiple entries, although not explicitly identified, since the actions require only the isolation of the affected system. This may be accomplished separately for each flow path within the allowed completion times. Since the addition of the Note retains current allowances, this change is considered to be administrative in nature.

A8 An explicit as-left acceptance criterion is included for ITS SR 3.4.10.1 which is equivalent to the Bases for CTS 2.2.2. Since this is currently a requirement for OPERABILITY, this change is considered administrative in nature.

A9 CTS 3.1.3.4 maximum indicated value for pressurizer level (305 inches) contains instrumentation uncertainty allowances and is inconsistent with other values in the CTS. For example, the minimum required pressurizer level (45 inches) does not contain instrumentation uncertainty allowances. Therefore, CTS 3.1.3.4 is administratively modified to present the safety analysis values for maximum pressurizer level and pressure, which establishes consistency with the minimum value. This change also establishes consistency with other parameters presented in the ITS. This change is considered to be administrative in that the same instrumentation uncertainty allowances for these parameters will exist in the future, and the same actual level limit is presented in the Technical Specifications.

A10 CTS 3.5.1.1 and 3.5.1.2 represent information on the proper action when the number of channels is less than required by CTS Table 3.5.1-1. For example, CTS 3.5.1 does not clearly specify that the number of channels identified in Table 3.5.1-1, Column 1, are required to be OPERABLE, and CTS 3.5.1.2 provides limitations for inoperable channels. Similarly, CTS Specifications 4.1.a and 4.1.b contain information on the proper application of CTS Table 4.1-1. These Specifications and the format of the referenced Tables are replaced with the appropriate ITS requirements. The CTS

markup for these Specifications and Tables does not attempt to depict all of the changes required to adopt the ITS format. Rather, the appropriate specific Discussion of Change (DOC) is indicated along with the appropriate CTS versus ITS cross-reference. Therefore, this change in format is considered administrative.

A11 CTS 3.1.6.2 includes the phrase "(exceeding normal evaporative losses)" which is not reflected in ITS. Since this phrase has no practical application, its omission has no impact on unit operation and is considered an administrative change.

3.4B-05

A12 LTOP requirements were incorporated into CTS with Amendment 95, and have been subsequently modified with Amendments 138, 140, 154, 161 and 188. These requirements are reflected in CTS 3.1.2.9, 3.1.2.10, 3.1.2.11, Table 4.1-1, item 60, and Table 4.1-2, item 17. Of these CTS 3.1.2.9, 3.1.2.10, and 3.1.2.11 are directly reflected in ITS LCO 3.4.11, and the associated CTS exceptions are reflected as Notes for each LCO item. Although not explicit in CTS, Table 4.1-2, item 17, provides indirect requirements for an OPERABLE electromatic relief valve (ERV). Since the ERV requirement is provided only for LTOP purposes, it is reflected in LCO item a. These changes are basically format changes and are considered to be administrative in nature. (The LTOP alarm logic required to be tested by Table 4.1-1, item 60, is addressed by DOC LA2).

The Applicability for the LTOP provisions is chosen consistent with the LTOP enable temperature (from CTS 3.1.2.10 for the HPI valves) and the associated LTOP analysis. One minor difference is the change from "< 262 F" to "≤ 262 F." However, since this change is so small as to be imperceptible and does not impact the actual application, this change is also considered to be administrative in nature.

The change in Applicability for CTS 3.1.2.9 is addressed by DOC L2. The new Applicability is also different than CTS 3.1.2.11. "When the RCS pressure boundary is intact" is considered to also include MODES 1, 2, and 3, and MODE 4 down to the LTOP enable temperature. However, these MODES are enveloped by ITS 3.4.9, "Pressurizer," level requirements, and therefore, ITS 3.4.11 need only address the remaining MODES down through MODE 6 when the reactor vessel head is on. Therefore, this change is also considered to be administrative in nature.

3.4B-09

A13 Surveillance frequencies in CTS Table 4.1-1 have been replaced with those from NUREG-1430. The CTS and corresponding ITS Frequencies are as follows:

CTS	ITS
S - Each shift	12 hours
W - Weekly	7 days
M - Monthly	31 days
D - Daily	24 hours
T/W - Twice per week	96 hours
Q - Quarterly	92 days
P - Prior to each startup if not done previous week	Not Used
B/M - Every 2 months	Not Used
R - Once every 18 months	18 months
PC - Prior to going Critical if not done within previous 31 days	Not Used
NA - Not Applicable	Not Used
SA - SA Twice per Year	184 days

(Note: Not all Frequencies are applicable to this package.)

TECHNICAL CHANGE -- MORE RESTRICTIVE

M1 Additional details are included to describe the "evaluate RCS leakage" test identified in CTS Table 4.1-2, item 6.a. The ITS SR 3.4.13.1 will require performance of an RCS water inventory balance which is the primary means of determining RCS leakage. This change is an additional restriction on unit operation consistent with NUREG-1430.

M2 The CTS 3.1.1.3.B requirement for OPERABILITY of one pressurizer safety valve while subcritical is retained in ITS 3.4.10 for MODE 3 and MODE 4 with RCS temperature above the LTOP enable temperature (LCO 3.4.10, Note 1, and SR 3.4.10.1, Note). Appropriate Required Actions are incorporated that provide a short time period to exit the MODE of Applicability if the required valve is not restored (Required Action C.1). This is considered a more restrictive change since CTS 3.1.1.3.B does not require any action for an inoperable valve, including shutdown pursuant to LCO 3.0.3 since it is not applicable. The CTS 3.1.1.3.A default action requirements for inoperable pressurizer safety valve(s) is revised to require that the unit be in MODE 3 within 6 hours. This is consistent with NUREG Required Action B.1. In MODE 3 (and MODE 4 with RCS temperatures above the LTOP enable temperature), the LCO requires one safety valve to be OPERABLE if all RCS openings are closed, except for ASME hydrostatic testing. The limitations and exceptions for a single OPERABLE safety valve are omitted. Further, if the single safety valve is not OPERABLE, i.e., both safety valves are inoperable, the unit will be required to reduce temperature to below the LTOP enable temperature where overpressure protection is

adequately provided by the LTOP requirements. These changes are appropriate to assure adequate LTOP. These are additional restrictions on unit operation as discussed above.

Additionally the requirements for OPERABILITY of two pressurizer safety valves is expanded from "when the reactor is critical" as identified in CTS 3.1.1.3.A to MODES 1 and 2. Since MODE 2 includes operation beginning with $k_{eff} \geq 0.99$, some operation in MODE 2 occurs prior to criticality. Therefore, this is also an additional restriction on unit operation consistent with NUREG-1430.

M3 The CTS is expanded to provide complete Specifications for LTOP. CTS 3.1.2.9, 3.1.2.10, 3.1.2.11 and Table 4.1-2, item 17 currently provide the requirements associated with LTOP. These requirements as reflected in ITS are addressed by DOC A12. However, the CTS does not provide specific ACTIONS for situations where the CTS LTOP requirements are not met. Conditions A and B provide appropriate specific Required Actions and Completion Times for pressurizer level not within the required limits. Conditions C and D provide appropriate specific Required Actions and Completion Times for insufficient pressure relief capability. Finally, Condition E provides appropriate specific Required Actions and Completion Times for any other Condition which does not meet the LTOP LCO requirements. These ACTIONS provided for ITS 3.4.11 represent additional restrictions on unit operation.

3.4B-11 The CTS also does not provide specific SRs for the associated CFT, HPI, pressurizer level, or pressure relief requirements (other than exercising the ERV as required by CTS Table 4.1-2, item 17). Specific periodic verification that the LTOP requirements are met is incorporated for ITS as SR 3.4.11.1 through SR 3.4.11.5. These specific SRs represent additional restrictions on unit operation.

M4 CTS Table 4.1-2, item 17 requires PORV (also known as the ERV) exercising at the end of each refueling outage. This is revised in ITS SR 3.4.11.5 to a Frequency of 18 months. CTS Table 4.1-2, item 11 is similarly revised from "each refueling outage" to "18 months." This change is appropriate since a fuel cycle is open ended and 18 months is consistent with the typical length of the fuel cycle. Although the standard Frequency of 18 months is intended to coincide with refueling outages, it is possible that the time between refueling outages could be more than 18 months. Therefore, this change is an additional restriction on unit operation.

M5 Text in CTS 3.1.6.8 is shown as deleted because it is included in the ITS definition of Identified LEAKAGE and is therefore subject to the requirements in ITS LCO 3.4.13.c. This text in CTS 3.1.6.8 provided an exception to CTS 3.1.6.1 which allowed up to 30 gallons per minute of leakage from reactor coolant system (RCS) valves provided it was capable of being returned to the RCS. This exception is inconsistent with the intent of the Identified LEAKAGE limitations. Therefore, this exception will not exist in the ITS. This change is consistent with NUREG-1430.

M6 The CTS 3.1.6.7 requirements that the unit be placed in HOT STANDBY within the next 6 hours (if the laboratory analysis of the reactor building air sample does not determine the RCS leakage to be acceptable) is revised to require the unit to be placed in ITS MODE 3. Since the CTS HOT STANDBY requires the unit to be ≤ 2% RTP and ITS MODE 3 is a subcritical condition, this change is an additional restriction on unit operation. The activity to reduce the unit to subcritical conditions provides consistency within the ITS for shutdown applications. This change is consistent with NUREG-1430.

M7 CTS 3.1.2.9 requirements are extended to be applicable during both cooldown and heatup operations. Low temperature overpressure conditions are also possible, and of concern, during heatup operation. Therefore, such an Applicability for ITS 3.4.11 is consistent with the assumptions of LTOP evaluations performed to date. This is an additional restriction on unit operation.

M8 Not used.

M9 The CTS 3.1.3.4 requirements for a pressurizer steam bubble are expanded to include ITS MODE 4 with the RCS temperature above the LTOP enable temperature. ITS 3.4.11 will provide for pressure control below the LTOP enable temperature. An additional Required Action (RA B.2) is included to require that the unit be placed in a MODE in which ITS 3.4.9 is not applicable. This additional Applicability is provided to prevent water solid RCS operation during heatup and cooldown which may result in rapid pressure fluctuations due to normal operational perturbations such as a pump start. An SR (3.4.9.1) is also included to periodically verify the pressurizer water level is being maintained consistent with the safety analysis assumptions. CTS 3.1.3.4 provides appropriate acceptance limits for this new SR, but the Frequency is not specified in CTS. These changes are additional restrictions on unit operation consistent with NUREG-1430.

CTS 3.1.3.6 requirements for OPERABLE pressurizer heaters are expanded to include ITS MODE 3. The Applicability is extended since MODE 3 is also a condition which would present a significant demand, in the event of a loss of offsite power, for maintaining the RCS in a hot pressurized condition with loop subcooling for an extended period. The LCO Note is included to address the difference in Applicability for the pressurizer water level and the heaters, i.e., the heaters are not required in ITS MODE 4. An additional Required Action (RA D.1) is included to require that the unit be placed in hot shutdown (ITS MODE 3) in 6 hours. This Required Action places the unit in a condition with reduced potential thermal energy should a LOCA occur. The 6 hour Completion Time provides a reasonable, consistent time to reach MODE 3, based on experience. These changes are additional restrictions on unit operation consistent with NUREG-1430.

M10 Not used.

M11 Not used.

M12 CTS 3.1.4.1 requires, if the activity is not returned to within the identified limits within the allowed time, that the unit be "brought to a hot shutdown condition using normal operating procedures." This is revised in ITS to require that the unit be in MODE 3 with RCS temperature < 500°F in 6 hours. Both CTS and ITS require that the unit be subcritical, but ITS additionally requires the temperature to be reduced to prevent significant releases following a SGTR event. The ITS also identifies a specific Completion Time for the action which allows for use of the normal operating procedures, but does not allow an unlimited time in which to use them. These changes are additional restrictions on unit operation consistent with NUREG-1430.

M13 An explicit ITS Applicability of MODES 1, 2, 3, and 4 is provided for CTS 3.1.6.1 and CTS 3.1.6.2 requirements. Even though an Applicability of "when the reactor is at power operation" provided by CTS 3.1.6.7 would typically be interpreted as ITS MODES 1 and 2, the actions of CTS 3.1.6.3 and CTS 3.1.6.7 require the unit to be in cold shutdown if the requirements are not met. Therefore, both ITS 3.4.13 and ITS 3.4.15 will be applicable in MODES 1, 2, 3, and 4.

ITS 3.4.13, Required Action A.1 will allow 4 hours to restore the primary to secondary leakage to within limits (consistent with CTS 3.1.6.3.b), and ITS 3.4.13, Required Action B.1 will allow 18 hours to restore the identified or unidentified leakage to within limits (consistent with CTS 3.1.6.1 and CTS 3.1.6.2). This 18 hour Completion Time, combined with the 6 hours allowed by Required Action C.1 to reach MODE 3, is consistent with the CTS 3.1.6.1 and CTS 3.1.6.2 requirements which require the unit to be shutdown, i.e., subcritical or MODE 3, in 24 hours. In addition, the CTS 3.1.6.1, 3.1.6.2, and 3.1.6.3.a requirements which require shutdown within 24 hours when the RCS leakage rate exceeds its limit are revised to also require that the unit be in MODE 5 in 36 hours; and the CTS 3.1.6.3.b requirements which require the unit to be in cold shutdown within 30 hours are revised to also require that the unit be in MODE 3 within 10 hours. These proposed requirements will continue to provide for a prompt change of the unit conditions in order to reduce the severity of the leakage and its potential consequences. Further reducing the unit pressure conditions to MODE 5 also reduces the leakage and the factors that tend to further degrade the pressure boundary. These changes are additional restrictions on unit operation consistent with NUREG-1430.

M14 CTS Table 3.5.1-1, Other Safety Related Systems, item 1, with Notes 1 and 5, require that, if the Decay Heat Removal System isolation valve automatic closure and interlock system is inoperable, the unit must be placed in hot shutdown in 12 hours, then 48 hours are allowed to attempt repairs, then the unit must be in cold shutdown in an additional 24 hours; a total of 84 hours. The proposed ITS 3.4.14 Condition will require isolation of the affected penetration by closing and de-activating the affected motor operated valves (MOVs) within 4 hours. This Required Action and its shortened Completion Time are an additional restriction on unit operation. (See also DOC L8.) These changes are additional restrictions on unit operation consistent with NUREG-1430.

M15 The CTS markup is annotated to show adoption of the ITS 3.4.15 Condition D. This Condition is entered if both the reactor building sump monitor and both of the reactor building atmosphere radioactivity monitors are inoperable. This results in a loss of both directly instrumented indications of abnormal reactor coolant system leakage. Although a loss of safety function may not have occurred because of the availability of an RCS inventory balance, ITS LCO 3.0.3 is immediately entered. This requirement is not directly indicated in the CTS and is therefore an additional restriction on unit operation. This change is consistent with NUREG-1430.

TECHNICAL CHANGE -- LESS RESTRICTIVE

L1 CTS Table 4.1-2, item 6b, refers to Note (1) to identify the Frequency associated with RCS pressure isolation valve leakage testing. These include "following each time the plant is placed in a cold shutdown condition if testing has not been accomplished in the preceding 9 months." ITS SR 3.4.14.1 includes this Frequency but requires the testing only if the unit is in the cold shutdown condition for 7 days or more. This will provide time for appropriate planning and scheduling of such testing which may not be possible for short forced outages.

L2 CTS 3.1.2.9, requires, for LTOP, each pressurized core flood tank to be isolated "before depressurizing the reactor coolant system below 600 psig" with some exceptions. CTS 3.1.2.10, requires, for LTOP, each high pressure motor operated valve to be closed with their opening control circuits for the motor operators disabled "when the reactor coolant temperature is less than 262°F" with some exceptions. CTS 3.1.2.11, requires, for LTOP, that the plant shall not be operated in a water solid condition "when the RCS pressure boundary is intact" with some exceptions. The Applicability for these requirements is revised to include only those conditions under which LTOP is necessary, i.e., only during low temperature conditions in conjunction with potential high pressure conditions. Since it is possible for the RCS to be below 600 psig with RCS temperature less than 262°F, this change is less restrictive. However, since overpressure protection for MODE 4 with RCS temperature > 262°F is adequately provided by the pressurizer safety valve(s) (ITS LCO 3.4.10), and the ITS 3.4.11 Applicability continues to provide the necessary LTOP provisions, the change is acceptable.

L3 CTS 3.1.6.3.b requires that if the primary to secondary leakage exceeds its limit, the unit be placed in cold shutdown within 34 hours. ITS 3.4.13, Required Action C.2 will provide for an additional 6 hours (40 hours total) to place the unit in MODE 5, i.e., Cold Shutdown. This Completion Time provides a consistent time frame for achieving this unit condition, and it has been determined to be reasonable, based on operating experience, to reach the required conditions from full power conditions in an orderly manner and without challenging unit systems. This change is consistent with NUREG-1430.

L4 CTS Table 4.1-1, Item 30, requires monthly testing of the decay heat removal system isolation valve automatic closure and interlock system. This testing is incorporated in ITS SR 3.4.14.2, SR 3.4.14.3, SR 3.4.14.4, and SR 3.4.14.5. However, the Frequency is revised to require this testing every 18 months. This Frequency is based on the preference to perform this Surveillance under the conditions that apply during a unit outage and the increased potential for an unplanned transient if the Surveillance is performed with the reactor at power. This Frequency is also acceptable based on consideration of the design reliability of the equipment.

L5 CTS Table 3.1.6.9, Footnote (a), items 1, 2, and 3 are not retained for ITS as stringent requirements. Items 1 and 2 actually identify acceptable leakage rates which will continue to be acceptable under ITS. Therefore, their omission results in no actual change to the requirements. Item 3 identifies an "unacceptable leakage rate" criterion based on a projection of exceeding the overall 5 gpm leakage rate criterion for each penetration. While it is appropriate to consider projections for determination of the need for maintenance and corrective actions, it is inappropriate to prevent any operation when the overall acceptance criteria are still being met. Therefore, this projection criterion is omitted. Requirements related to leakage of these primary coolant system pressure isolation valves are contained in the unit implementing procedures. Currently these requirements reside in OP 1102.001, "Plant Preheatup Precritical Checklist," Sup 4 and OP 1104.004, "Decay Heat Removal Operating Procedure." Since these procedures are included in SAR Section 12.5.2, changes to them are performed in accordance with the requirements of 10 CFR 50.59. This change is consistent with NUREG-1430.

3.4B-26

L6 The CTS 3.1.3.7 requirements to "restore..." in 15 minutes or be in "at least hot shutdown" within the next 15 minutes when CTS 3.1.3.2 is not met are revised to require the unit to "restore" in 1 hour or be in MODE 3 within the next 6 hours. These revised Completion Times are considered to be appropriate for the Required Actions, allowing the activity to be accomplished in a controlled, orderly manner without challenging unit systems, and are consistent with NUREG-1430.

L7 The CTS 3.1.4.1 requirements for applicability of the RCS activity limits are revised to MODES 1 and 2, and MODE 3 with the RCS temperature ≥ 500°F. Although the CTS applicability is not clearly stated in CTS 3.1.4.1, item c of this Specification requires that, upon noncompliance, the unit eventually be placed in cold shutdown (ITS MODE 5), and Table 4.1-3, item 1, which requires the sampling and analysis to verify compliance, includes Note (7) which indicates the analysis is not required in cold shutdown or refueling (ITS MODES 5 and 6). Therefore, the Applicability of CTS 3.1.4.1 is considered to be equivalent to ITS MODES 1, 2, 3, and 4. The proposed conditions are consistent with the steam generator tube rupture release assumptions. Below 500°F in MODE 3, and in MODES 4 and 5, such a release is unlikely since the saturation pressure of the reactor coolant is below the lift pressure settings of the atmospheric dump valves and main steam safety valves. This change is consistent with NUREG-1430.

L8 CTS Table 3.5.1-1, item 1 with Notes 1 and 5, requires that, if the Decay Heat Removal System isolation valve automatic closure and interlock system is inoperable, the unit must be placed in hot shutdown in 12 hours, then 48 hours are allowed to attempt repairs, then the unit must be in cold shutdown in an additional 24 hours; a total of 84 hours. The proposed ITS Condition will require isolation of the affected penetration by closing and de-activating the affected motor operated valves (MOVs) within 4 hours. The MOV is a valve in the decay heat removal injection line which also provides an emergency core cooling system (ECCS) low pressure injection function. Closing and de-activating this valve (within 4 hours) results in an inoperable ECCS train which will allow 72 hours (ITS 3.5.2) or 48 hours (ITS 3.5.3) for restoration of the system. Failure to restore OPERABILITY will then require the unit to be in MODE 3 in 6 hours and MODE 3 with RCS temperature ≤ 350°F in 12 hours if beginning from MODES 1, 2 , or 3 (ITS 3.5.2), and in MODE 5 in 24 hours if beginning in MODE 4 or in MODE 3 with RCS temperature > 350°F (ITS 3.5.3). Therefore, if the inoperability is discovered while in MODES 1, 2, or 3, the unit will be allowed a total of 184 hours ((4 + 72 + 36)+(48 + 24)) to reach MODE 5 (cold shutdown). This is less restrictive than the corresponding CTS requirements. However, this is acceptable because the valve has been placed in the safe position, i.e., closed and de-activated. (See also DOC M14.)

A Note is also included with the LCO (see DOD 20, which editorially relocated this Note from the Applicability) to limit the requirements for DHR System valves in MODE 4 when DHR is in, or being placed in, service. With DHR performing a vital function of removing decay heat, the specified actions (i.e., to isolate the system) may not be prudent. As such this change reflects an enhancement to safety. This is consistent with NUREG-1430 (except as described in DOD 20).

L9 The CTS Table 4.1-2, item 17, requirement is to test the PORV (ERV) by exercising at the "end of each refueling outage." This is revised in ITS SR 3.4.11.5 to a Frequency of "18 months." CTS Table 4.1-2, item 11 is similarly revised from "each refueling outage" to "18 months." These changes are appropriate since a fuel cycle is open ended and 18 months is consistent with the typical length of the fuel cycle. Further, the requirements are necessary both during startup and shutdown and may be required at any time during the fuel cycle. Therefore, specifying a particular time in the fuel cycle for performance of the SR is not justified. However, since the proposed Frequency does not specify that the SR may be performed only at the end of the refueling outage, i.e., the SR may be performed at any time during the fuel cycle, and because the refueling cycle may be less than 18 months, the change is less restrictive than CTS.

L10 Not used.

3.4B-13 3.4B-14 L11 CTS Table 4.1-3 item 1.b requires the determination of gross activity concentration in the RCS three times per week and at least every third day. The frequency is increased by either of two CTS Table 4.1-3 Notes. Note 1 requires that the frequency be increased to a minimum of once per day if the gross radioactivity concentration exceeds 10% of the limit specified in CTS 3.1.4.1 (limit is 3.5 μC/gm) or increases by 10 μC/gm from the previous measured level, until a steady state activity level is established. Note 6 requires additional sampling within 24 hours of any criticality if the gross radioactivity concentration of prior operation is < 1% but < 10% of the CTS 3.1.4.1 limit, and prior to any criticality if the gross radioactivity concentration is > 10% of the CTS 3.1.4.1 limit. ITS SR 3.4.12.1 will specify a Frequency of 7 days for this parameter.

CTS Table 4.1-3, item 1.c requires the determination of gross radioiodine concentration in the RCS once per week. The frequency is increased by either of two CTS Table 4.3-1 Notes. Note 3 requires that, in addition to the weekly sample, the radioiodine concentration be determined if the measured gross radioactivity concentration changes by more than 10 μC/gm from the previous measured level. Note 6 requires additional sampling within 24 hours of any criticality if the steady state radioiodine concentration of prior operation is < 1% but < 10% of the CTS 3.1.4.1 limit, and prior to any criticality if the steady state radioiodine concentration is > 10% of the CTS 3.1.4.1 limit. ITS SR 3.4.12.2 will specify a Frequency of 14 days for this parameter.

CTS Table 4.3-1, item 1.g requires the determination of E-bar monthly. CTS Table 4.3-1 Note 2 revises the frequency by requiring a radiochemical analysis and calculation of E-bar if the measured gross activity changes by more than 10μC/gm from the previous measured level. ITS SR 3.4.12.3 will specify a Frequency of 184 days for this parameter. This SR is also modified by a Note that requires sampling to be performed 31 days after a minimum of 2 EFPD and 20 days of MODE 1 operation have elapsed since the reactor was last subcritical for at least 48 hours.

The proposed Frequencies, and the deletion of the frequency modifying requirements are acceptable because these measurements are considered to be verifications of actual RCS conditions. Installed instrumentation continuously monitors the RCS for changes in radioactivity. If the Failed Fuel monitor indicates a high alarm on failed fuel, an unexplained increase in gross activity, a rise in gross/iodine ratio, or a drop in gross/iodine ratio, the operator follows the guidance contained in the abnormal operating procedure related to high activity in the RCS. This abnormal operating procedure requires evaluations of the RCS condition including sampling to confirm these parameters. SAR Section 12.5.2 requires abnormal operating procedures to be maintained. As this procedure is referenced from the SAR, changes to the procedure are evaluated in accordance with the requirements of 10 CFR 50.59. These parameters are also performance criteria for the Unit 1 Rx System in the ANO 10 CFR 50.65 Maintenance Rule program. This provides assurance that adverse trends would be observed and corrected, prior reaching the performance criteria limit of 1 μC/gm. These proposed Frequencies are consistent with NUREG-1430.

CTS DISCUSSION OF CHANGES

L12 The RCS leakage evaluation Frequency required by CTS Table 4.1-2, item 6a, is revised from "daily" to once every 72 hours. This Frequency is also modified by a Surveillance column Note that indicates that ITS SR 3.4.13.1 is not required to be performed until 12 hours after establishment of steady state operation at or near operating pressure. An RCS water inventory balance is the primary method of determining leakage. However, steady state operation at near operating pressure is required to perform a proper water inventory balance; calculations during maneuvering may be useful to identify major problems, but they are not sufficient to accurately determine leakage. The 12 hours provides a reasonable period once the necessary operating conditions are established to perform the water inventory balance. The 72 hour Frequency is a reasonable interval to trend LEAKAGE and recognizes the importance of early leakage detection in the prevention of accidents. This change is consistent with NUREG-1430 as revised by TSTF-116, Rev. 2.

L13 CTS 3.1.6.7 requires three RCS leakage detection methods of different operating principles to be in operation. The Bases identify these as the sump level, radioactivity, and water inventory. Only the first two of these are required by ITS LCO 3.4.15. The water inventory balance is required to fulfill ITS SR 3.4.13.1. CTS 3.1.6.7 indicates that both a gaseous detector and an air particulate monitor are provided to fulfill the method which uses the radioactivity monitoring principle, and provides actions only when both monitors are out of service. This implies that one of the monitors may be out of service with no required actions. This interpretation is consistent with ITS. However, no time is provided in the CTS for either the sump level or the water inventory balance capabilities to be out of service. ITS 3.4.15 includes Required Action A.2 allowing the sump level monitor to be out of service for up to 30 days, and ITS SR 3.4.13.1 and ITS 3.4.15 Required Action A.1 requires the water inventory balance to be performed on specific intervals, i.e., instruments required for the water inventory balance may be out of service for short periods as long as the inventory balance Frequency is met. Also, the out of service time for both radioactivity monitors is extended from 72 hours to 30 days (ITS 3.4.15 Required Action B.2), and the required frequency for grab samples (while the radioactivity monitors are out of service) is revised from once per shift to once per 24 hours. Further, an alternative to taking grab samples is also provided as Required Action B.1.2, i.e., performing a water inventory balance on a more frequent interval. These ACTIONS continue to provide for adequate leakage monitoring capability, and are therefore, appropriate

Finally, an ACTIONS Note is provided in ITS LCO 3.4.15 which indicates that LCO 3.0.4 is not applicable. This exception will allow startup while depending on one of the Conditions. Since Required Actions A.2 and B.2 require restoration within 30 days, the Conditions do not allow unlimited continued operation and LCO 3.0.4 would not normally allow entry into any of the applicable MODES while operating within this Required Action. The Note is appropriate and acceptable because sufficient other equipment is available to provide the adequate leakage monitoring over the 30 days. This change is consistent with NUREG-1430 as modified by TSTF-060.

L14 CTS 3.1.6.1 allows only 10 gpm total RCS leakage which includes the identified, unidentified, and steam generator tube leakage. This is revised in ITS 3.4.13 to allow 10 gpm identified leakage, in addition to the 1 gpm unidentified leakage. This increase in the limit of 1 gpm is not significant since the allowed rate remains within the capability of the makeup system, the leakage is from a known source, and the capability of the leak detection systems is not impacted by the additional 1 gpm. This change is consistent with NUREG-1430.

L15 The CTS 3.1.1.3.B requirements for a single OPERABLE pressurizer safety valve when the reactor is subcritical are retained only for the conditions of MODE 3 and MODE 4 above the LTOP enable temperature. Above the LTOP enable temperature, the pressurizer safety valves provide the primary protection against overpressurization. At or below the LTOP enable temperature, the overpressure protection is provided by other equipment and controls, and the pressurizer safety valve is not credited. Therefore, the pressurizer safety valve(s) are not required to be OPERABLE at or below the LTOP enable temperature. CTS 3.1.1.3.B also contained a statement that only one code safety is required to be operable if all reactor coolant system openings are closed. This statement is deleted due to the change in specified MODE. These changes are consistent with NUREG-1430.

3.4B-23

L16 The testing requirements of CTS Table 3.1.6.9 and Table 4.1-2, item 6b are revised by the addition of a Note for ITS SR 3.4.14.1 which indicates the leakage testing is only required to be performed in MODES 1 & 2. This permits entry into MODES 3 and 4 to establish the necessary differential pressures and stable conditions to allow for performance of this Surveillance. The Note that allows this provision is complimentary to the need to not perform the test on the DHR System when the DHR System is aligned to the RCS in the decay heat removal mode of operation. PIVs contained in the DHR flow path must be leakage rate tested after DHR is secured and stable unit conditions and the necessary differential pressures are established. This change is consistent with NUREG-1430.

L17 The CTS 3.1.1.3 requirement for OPERABILITY of one or two pressurizer safety valves is retained in ITS 3.4.10. The ITS includes an additional Note (LCO 3.4.10, Note 1) which allows the OPERABILITY to be based on a preliminary cold lift setting made prior to heatup for operation in MODE 4 and up to 36 hours of operation in MODE 3. This permits testing and examination of the safety valves at high pressure and temperature near their normal operating range, but only after the valves have had a preliminary cold setting. The cold setting gives assurance that the valves are OPERABLE near their design condition. This change is consistent with NUREG-1430.

3.4B-03

L18 The CTS 3.1.4.1.c requirement to initiate immediate corrective action is omitted from ITS 3.4.12. The ITS requirements to complete Required Actions within a specified Completion Time permits appropriate evaluation of the situation, its causes, and the impact of the corrective action being considered while maintaining a limited time period during which the restoration of compliance must be achieved. This change is consistent with NUREG-1430.

L19 The relocation of CTS 3.1.6.8 moves a 30 gpm upper limit on returnable RCS leakage via the reactor coolant pump seals from the CTS to the TRM. This upper limit was an exception to the CTS 3.1.6.1 limitation on total leakage that is largely equivalent to the NUREG limitation on Identified LEAKAGE. Further, this CTS limitation is already provided as an exclusion to the NUREG definition for Identified Leakage. The basis for this limitation is well described in ANO-1 SAR Section 4.3.11.3, "Leakage." The relocation of this exception will be less restrictive in that no specific upper limit on RCP seal leakoff will be specified in the ITS. Appropriate administrative controls will still be in place via the TRM requirements and the Condition Reporting corrective action process should the value exceed the TRM and SAR established value. Because these controls are in place, the CTS actions shown as applicable when the exception is in effect have been marked as deleted. This DOC addresses only those CTS requirements associated the relocation of the 30 gpm limit. This change is consistent with NUREG-1430.

3.4B-25

LESS RESTRICTIVE -- ADMINISTRATIVE DELETION OF REQUIREMENTS

LA1 This information has been moved to the Bases. This information provides details of design or process which are not directly pertinent to the actual requirement, i.e., Definition, Limiting Condition for Operation or Surveillance Requirement, but rather describe additional unnecessary details such as an acceptable method of compliance. Since these details are not necessary to adequately describe the actual regulatory requirement, they can be moved to a licensee controlled document without a significant impact on safety. Placing these details in controlled documents provides adequate assurance that they will be maintained. The Bases will be controlled by the Bases Control Process in Chapter 5 of the proposed Technical Specifications. This change is consistent with NUREG-1430.

	CTS Location	New Location
	2.2.2	Bases 3.4.10, SR 3.4.10.1
	3.1.2.9	Bases 3.4.11, LCO
	3.1.2.10	Bases 3.4.11, LCO
3.4B-24	3.1.4.1.a	Bases 3.4.16, LCO
	3.1.6.3.b	Bases 3.4.13, LCO
	Table 3.1.6.9	Bases 3.4.14, SR 3.4.14.1
	Table 3.1.6.9, footnote (c)	Bases 3.4.14, SR 3.4.14.1
3.4B-22	Table 4.1-3, Note (1)	Bases 3.4.12, SR 3.4.12.1
	Table 4.1-3, Note (2)	Bases 3.4.12, SR 3.4.12.3
	Table 4.1-3, Note (4)	Bases 3.4.12, SR 3.4.12.3

LA2 This information has been moved to the Technical Requirements Manual (TRM). This information provides details of design or process which are not directly pertinent to the actual requirement, i.e., Definition, Limiting Condition for Operation or Surveillance Requirement, but rather describe additional unnecessary details such as an acceptable method of compliance. Since these details are not necessary to adequately describe the actual regulatory requirement, they can be moved to a licensee controlled document without a significant impact on safety. Placing these details in controlled documents provides adequate assurance that they will be maintained. The TRM will be controlled by 10 CFR 50.59 and 10 CFR 50.71, as applicable. This change is consistent with NUREG-1430.

CTS Location	New Location
3.1.6.2	TRM
3.1.6.5	TRM
3.1.6.8	TRM
3.4B-22 3.5.1.7	TRM
Table 3.5.1-1, OTHER... #1a	TRM
Table 3.5.1-1, OTHER... #1b	TRM
Table 4.1-1, #30	TRM
Table 4.1-1, #60 [a]	TRM
Table 4.1-2, #7	TRM
Table 4.1-2, Note (2)	TRM
Table 4.1-3, #1.a	TRM
Table 4.1-3, Note 7	TRM

3.4B-12 (a) CTS Table 4.1-1 #60 provides testing and calibration requirements for the Low Temperature Overpressure Protection Alarm Logic. This circuitry provides an alarm to the operators when the RCS temperature is below the LTOP temperature and the HPI motor operated valves are not disabled, the ERV isolation valve is closed, the ERV setpoint is not in LTOP the position, or the pressurizer level is not within limits. This circuit is not associated with the ERV opening circuitry, but is considered to be an operator tool in diagnosing potential LTOP inoperabilities.

LA3 This information has been moved to the Inservice Testing (IST) Program. This information provides details of design or process which are not directly pertinent to the actual requirement, i.e., Definition, Limiting Condition for Operation or Surveillance Requirement, but rather describe additional unnecessary details such as an acceptable method of compliance. Since these details are not necessary to adequately describe the actual regulatory requirement, they can be moved to a licensee controlled document without a significant impact on safety. Placing these details in controlled documents provides adequate assurance that they will be maintained. The IST Program will be controlled by 10 CFR 50.54a and 10 CFR 50.59. This change is consistent with NUREG-1430.

Table 4.1-2, #3	IST
Table 4.1-2, Note (1)	IST

3.4.10

2.2 SAFETY LIMITS - REACTOR SYSTEM PRESSURE

<LATER> (2.0)

LATER

~~**Applicability**~~

~~Applies to the limit on reactor coolant system pressure.~~

~~**Objective**~~

~~To maintain the integrity of the reactor coolant system and to prevent the release of significant amounts of fission product activity.~~

~~**Specification**~~

~~2.2.1 The reactor coolant system pressure shall not exceed 2750 psig when there are fuel assemblies in the reactor vessel.~~

3.4.10 LCO

2.2.2 The setpoint of the pressurizer code safety valves shall be in accordance with ASME, Boiler and Pressurizer Vessel Code, Section III, Article 9, Summer 1968.

OPERABLE

LA1 Bases

Bases

~~The reactor coolant system (1) serves as a barrier to prevent radionuclides in the reactor coolant from reaching the atmosphere. In the event of a fuel cladding failure, the reactor coolant system is a barrier against the release of fission products. Establishing a system pressure limit helps to assure the integrity of the reactor coolant system. The maximum transient pressure allowable in the reactor coolant system pressure vessel under the ASME code, Section III, is 110 percent of design pressure.(2) The maximum transient pressure allowable in the reactor coolant system piping, valves, and fittings under ANSI Section B31.7 is 110 percent of design pressure. Thus, the safety limit of 2750 psig (110 percent of the 2500 psig design pressure) has been established. (2) The settings for the reactor high pressure trip (2355 psig) and the pressurizer code safety valves (2500 psig ±1%)(3) have been established to assure that the reactor coolant system pressure safety limit is not exceeded. When testing the pressurizer code safety valves, the "as found" lift setpoint may be 2500 psig +1, -3%. However, if found outside of a ±1% tolerance band, they shall be reset to 2500 psig ±1%. The initial hydrostatic test is conducted at 3125 psig (125 percent of design pressure) to verify the integrity of the reactor coolant system. Additional assurance that the reactor coolant system pressure does not exceed the safety limit is provided by setting the pressurizer electromatic relief valve at 2450 psig.(4)~~

A2

~~**REFERENCES**~~

~~(1) FSAR, Section 4~~

~~(2) FSAR, Section 4.3.11.1~~

~~(3) FSAR, Section 4.2.4~~

~~(4) FSAR, Table 4-1~~

<ADD SR 3.4.10.1 as-left lift setting criterion> A8

3.4B-03

<Add 3.4.10 LCO Note 2> L17

3.1 REACTOR COOLANT SYSTEM

A1

Applicability

Applies to the operating status of the reactor coolant system.

Objective

To specify those limiting conditions for operation of the reactor coolant system which must be met to ensure safe reactor operations.

3.1.1 Operational Components

Specification

<LATER> (3.4A)

3.1.1.1 Reactor Coolant Pumps

A. Pump combinations permissible for given power levels shall be as shown in Table 2.3-1. Operation with one Reactor Coolant Pump operating in each loop is limited to 24 hours with the reactor critical. — LATER

<LATER> (3.4A & 3.9)

B. The boron concentration in the reactor coolant system shall not be reduced unless at least one reactor coolant pump or one decay heat removal pump is circulating reactor coolant. With no reactor coolant pumps or decay heat removal pumps running, immediately suspend all operations involving a reduction of boron concentration in the reactor coolant system. — LATER

<LATER> (3.4 A)

3.1.1.2 Steam Generator — LATER

A. Two steam generators shall be operable whenever the reactor coolant average temperature is above 280°F.

3.1.1.3 Pressurizer Safety Valves

3.4.10 LCO & APPL
3.4.10 RA A.1
3.4.10 RA B.1

A. Both pressurizer code safety valves in MODES 1 & 2 shall be operable ~~when the reactor is critical.~~ With one pressurizer code safety valve inoperable, either restore the valve to operable status within 15 minutes or be in ~~HOT SHUTDOWN~~ MODE 3 within ~~12~~ 6 hours. M2 A1

3.4B-23

3.4.10 LCO Note 1
3.4.10 LCO Note 3
3.4.10 LCO Note 4

B. When the reactor is ~~subcritical~~ in MODE 3 or MODE 4 @ > LTOP enable temp at least one pressurizer code safety valve shall be operable if all ~~reactor coolant system openings are closed,~~ except for hydrostatic tests in accordance with ASME Boiler and Pressure Vessel Code, Section III. The provisions of Specification 3.0.3 are not applicable. L15

<LATER> (3.4A)

3.1.1.4 Reactor Internals Vent Valves — LATER

The structural integrity and operability of the reactor internals vent valves shall be maintained at a level consistent with the acceptance criteria in Specification 4.1. The provisions of Specification 3.0.3 are not applicable.

<LATER> (3.4A)

3.1.1.5 Reactor Coolant Loops — LATER

A. With the reactor coolant average temperature above 280°F, the reactor coolant loops listed below shall be operable:

<Add 3.4.10 RA C.1 & Cond.B - second entry condition> M2

BASES:

The plant is designed to operate with both reactor coolant loops and at least one reactor coolant pump per loop in operation, and maintain DNBR above 1.30 (for the BAW-2 correlation) and 1.18 (for the BWC correlation) during all normal operations and anticipated transients. (1)

Whenever the reactor coolant average temperature is above 280°F, single failure considerations require that two loops be operable.

(A2)

The decay heat removal system suction piping is designed for 300°F thus, the system can remove decay heat when the reactor coolant system is below this temperature. (2,3)

One pressurizer code safety valve is capable of preventing overpressurization when the reactor is not critical since its relieving capacity is greater than that required by the sum of the available heat sources which are pump energy, pressurizer heaters, and reactor decay heat. (4) Both pressurizer code safety valves are required to be in service prior to criticality to conform to the system design relief capabilities. The code safety valves prevent overpressure for a rod withdrawal accident. (5) The pressurizer code safety valve lift setpoint shall be 2,500 psig ±1 percent allowance for error and each valve shall be capable of relieving 300,000 lb/h of saturated steam at a pressure not greater than 3 percent above the set pressure. When testing the pressurizer code safety valves, the "as found" lift setpoint may be 2500 psig +1, -3 percent. However, if found outside the ±1 percent tolerance band, they shall be reset to 2500 psig ±1 percent.

The internals vent valves are provided to relieve the pressure generated by steaming in the core following a LOCA so that the core remains sufficiently covered. Inspection and manual actuation of the internal vent valves (1) ensure operability, (2) ensure that the valves are not open during normal operation, and (3) demonstrate that the valves begin to open and are fully open at the forces equivalent to the differential pressures assumed in the safety analysis.

The reactor coolant vents are provided to exhaust noncondensible gases and/or steam from the primary system that could inhibit natural circulation core cooling. The operability of at least one reactor coolant system vent path from the reactor vessel head, the reactor coolant system highpoints, and the pressurizer steam space ensures the capability exists to perform this function. The valve redundancy of the vent paths serves to minimize the probability of inadvertent actuation and breach of reactor coolant pressure boundary while ensuring that a single failure of a vent valve, power supply, or control system does not prevent isolation of the vent path. Testing requirements are covered in Section 4.0 for the class 2 valves and Table 4.1-2 for the vent paths. These are consistent with ASME Section XI and Item II.B.1 of NUREG-0737, "Clarification of TMI Action Plan Requirements," 11/80.

REFERENCES

(1) FSAR, Tables 9-10 and 4-3 through 4-7
(2) FSAR, Section 4.2.5.1 and 9.5.2.3
(3) FSAR, Section 4.2.5.4
(4) FSAR, Section 4.3.10.4 and 4.2.4
(5) FSAR, Section 4.3.7

<LATER> (3.4A)

3.1.2.7 ~~Prior to reaching thirty one effective full power years of operation, Figures 3.1.2-1, 3.1.2-2 and 3.1.2-3 shall be updated for the next service period in accordance with 10CFR50, Appendix G. The service period shall be of sufficient duration to permit the scheduled evaluation of a portion of the surveillance data scheduled in accordance with the latest revision of Topical Report BAW-1543(5). The highest predicted adjusted reference temperature of all the beltline region materials shall be used to determine the adjusted reference temperature at the end of the service period. The basis for this prediction shall be submitted for NRC staff review in accordance with Specification 3.1.2.8. The provisions of Specification 3.0.3 are not applicable.~~

3.1.2.8 ~~The updated proposed technical specifications referred to in 3.1.2.7 shall be submitted for NRC review at least 90 days prior to the end of the service period.~~

LATER

M7

3.4B-05

3.4.11 LCO w/ Notes 1 and 5

3.1.2.9 With the exception of ASME Section XI testing and when the core flood tank is depressurized, ~~during a plant cooldown~~ the core flood tank ~~discharge valves~~ shall be ~~closed and the circuit breakers for the motor operators opened before depressurizing the reactor coolant system below 600 psig.~~ isolated.

LA1 Bases

L2

3.4B-05

3.4.11 LCO w/ Notes 1-4

3.4.11 APPL

≤

3.1.2.10 With the exception of ASME Section XI testing, fill and vent of the reactor coolant system, emergency RCS makeup and to allow maintenance of the valves, when the reactor coolant temperature is less than 262°F, the High Pressure Injection ~~motor operated valves~~ shall be ~~closed with their opening control circuits for the motor operators disabled.~~ deactivated.

A12

LA1 Bases

3.4B-05

3.4.11 LCO a

3.4.11 LCO b

3.4.11 LCO a Notes 1 and 2

3.1.2.11 The plant shall not be operated in a water solid condition when the RCS pressure boundary is intact ~~except as allowed by Emergency Operating Procedures and during System Hydrotest.~~

A12

3.4B-05

<Add 3.4.11 LCO b> A12

<Add 3.4.11 LCO a (ERV Requirements)> A12

<Add 3.4.11 Appl> A12

<Add 3.4.11 ACTIONS> M3

3.4B-10

3.4B-11

Add SR 3.4.11.1
Add SR 3.4.11.2
Add SR 3.4.11.3
Add SR 3.4.11.4 with Note

M3

BASES

All reactor coolant system components are designed to withstand the effects of cyclic loads due to system temperature and pressure changes.[1] These cyclic loads are introduced by unit load transients, reactor trips, and unit heatup and cooldown operations. The number of thermal and loading cycles used for design purposes are shown in Table 4-8 of the FSAR. The maximum unit heatup and cooldown rates satisfy stress limits for cyclic operation.[2] The 200 psig pressure limit for the secondary side of the steam generator at a temperature less than 100°F satisfies stress levels for temperatures below the DTT.[3]

R TRM

The major components of the reactor coolant pressure boundary have been analyzed in accordance with Appendix G to 10CFR50. Results of this analysis, including the actual pressure-temperature limitations of the reactor coolant pressure boundary, are given in FTI Document 77-1258569-01 [4]. The limiting weld material is being irradiated as part of the B&W Owners Group Integrated Reactor Vessel Material Surveillance Program and the identification and locations of the capsules containing the limiting weld material is discussed in the latest revision to B&W report, BAW-1543. [5] The chemical composition of the limiting weld material is reported in the B&W Report, BAW-2121P [6]. The effect of neutron irradiation on the RT_{NDT} of the limiting weld material is reported in FTI Calculations 32-1245917-00 and 32-1257716-00 [7].

A2

Figures 3.1.2-1, 3.1.2-2, and 3.1.2-3 present the pressure-temperature limit curves for hydrostatic test, normal heatup, and normal cooldown respectively. The limit curves are applicable through the thirty first effective full power year of operation. The service period was reduced by one effective full power year from that assumed in FTI Document 77-1258569-01 to be conservative with respect to independent calculations performed by the NRC staff. The pressure limit is also adjusted for the pressure differential between the point of system pressure measurement and the limiting component for all allowed operating reactor coolant pump combinations.

The pressure-temperature limit lines shown on Figure 3.1.2-2 for reactor criticality and on Figure 3.1.2-1 for hydrostatic testing have been provided to assure compliance with the minimum temperature requirements of Appendix G to 10CFR50 for reactor criticality and for inservice hydrostatic testing.

The actual shift in RT_{NDT} of the beltline region material will be established periodically during operation by removing and evaluating, in accordance with Appendix H to 10CFR50, reactor vessel material irradiation surveillance specimens which are installed near the inside wall of this or a similar reactor vessel in the core region.

The spray temperature difference restriction based on a stress analysis of the spray line nozzle is imposed to maintain the thermal stresses at the pressurizer spray line nozzle below the design limit. Temperature requirements for the steam generator correspond with the measured NDTT for the shell.

R TRM

~~The heatup and cooldown rates stated in this specification are intended as the maximum changes in temperature in one direction in a one hour period. The actual temperature linear ramp rate may exceed the stated limits for a time period provided that the maximum total temperature difference does not exceed the limit and that a temperature hold is observed to prevent the total temperature difference from exceeding the limit for the one hour period.~~

~~Specification 3.1.2.9 is to ensure that the core flood tanks are not the source for pressurizing the reactor coolant system when in cold shutdown.~~

~~Specification 3.1.2.10 is to ensure that high pressure injection is not the source of pressurizing the reactor coolant system when in cold shutdown. The LTOP enable temperature has been calculated in accordance with Code Case N-514. Instrument error is not included in the reactor coolant temperature of 262°F.~~

~~Specification 3.1.2.11 is to ensure that the reactor coolant system is not operated in a manner which would allow overpressurization due to a temperature transient.~~

A2

~~REFERENCES~~

~~(1) FSAR, Section 4.1.2.4~~

~~(2) ASME Boiler and Pressure Code, Section III, N-415~~

~~(3) FSAR, Section 4.3.11.5~~

R TRM

~~(4) FTI Document Number 77-1258569-01~~

~~(5) BAW-1543, latest revision~~

~~(6) BAW-2121P~~

~~(7) FTI Calculation Numbers 32-1245917-00 and 32-1257716-00~~

A2

<Add 3.4.9 RA D.1> — M9
<Add SR 3.4.9.1>

<LATER> (3.4A)

3.1.3 Minimum Conditions for Criticality

Specification

<LATER> (3.1, 3.4A)

3.1.3.1 The reactor coolant temperature shall be above 525F except for portions of low power physics testing when the requirements of Specification 3.1.8 shall apply. — LATER

<LATER> (3.4A)

3.1.3.2 Reactor coolant temperature shall be to the right of the criticality limit of Figure 3.1.2-2. — LATER

<LATER> (3.1, 3.4A)

3.1.3.3 When the reactor coolant temperature is below the minimum temperature specified in 3.1.3.1 above, except for portions of low power physics testing when the requirements of Specification 3.1.8 shall apply, the reactor shall be subcritical by an amount equal to or greater than the calculated reactivity insertion due to depressurization. — LATER

In MODES 1, 2, 3 & MODE 4 @ ≥ 262°F — A1

3.4.9 APPL LCO a

3.1.3.4 ~~The reactor shall be maintained subcritical by at least 1 percent Δk/k until a steam bubble is formed~~ and ~~an indicated~~ water level between 45 and ~~305~~ 320 inches is established in the pressurizer. — M9 — A9

<LATER> (3.1, 3.2)

3.1.3.5 Except for physics tests and as limited by 3.5.2.1, safety rod groups shall be fully withdrawn and the regulating rods shall be positioned within their position limits as defined by Specification 3.5.2.5 prior to any other reduction in shutdown margin by deboration or regulating rod withdrawal during the approach to criticality. — LATER — M9

3.4.9 APPL LCO b RA C.1 RA D.2

In MODES 1, 2, + 3 — A3

3.1.3.6 ~~The reactor shall not be made critical~~ until at least 2 of the 3 emergency-powered pressurizer heater groups are operable. With less than 2 of the 3 required heater groups operable, restore the required heater groups to operable status within 72 hours. If the required heater groups are not restored to operable status within 72 hours, be in ~~hot shutdown~~ MODE 4 within the following 12 hours. — M9 — A1

3.4.9 RA A.1/B.1 + <LATER> (3.1 / 3.4A)

3.1.3.7 With any of the above limits violated, restore the reactor to within the limit in ~~15 minutes~~ 1 hour or be in at least ~~Hot Shutdown~~ MODE 3 within the next ~~15 minutes~~ 6 hours — + LATER — L6

Bases — A2

At the beginning of life of the initial fuel cycle, the moderator temperature coefficient is expected to be slightly positive at operating temperatures with the operating configuration of control rods. (1) Calculations show that above 525F the positive moderator coefficient is acceptable.

Since the moderator temperature coefficient at lower temperatures will be less negative or more positive than at operating temperature, (2) startup and operation of the reactor when reactor coolant temperature is less than 525F is prohibited except where necessary for low power physics tests.

The potential reactivity insertion due to the moderator pressure coefficient (2) that could result from depressurizing the coolant from 2100 psia to saturation pressure of 900 psia is approximately 0.1 percent Δk/k.

During physics tests, special operating precautions will be taken. In addition, the strong negative Doppler coefficient (1) and the small integrated Δk/k would limit the magnitude of power excursion resulting from a reduction of moderator density.

<Add 3.4.9 RA B.2> — M9
<Add 3.4.9 LCO NOTE>

A2

The requirement that the reactor is not to be made critical below the limits of Figure 3.1.2-2 provides increased assurances that the proper relationship between primary coolant pressure and temperatures will be maintained relative to the NDTT of the primary coolant system. Heatup to this temperature will be accomplished by operating the reactor coolant pumps.

If the shutdown margin required by Specification 3.5.2 is maintained, there is no possibility of an accidental criticality as a result of a decrease of coolant pressure.

The requirement for pressurizer bubble formation and specified water level when the reactor is less than one (1) percent subcritical will assure that the reactor coolant system cannot become solid in the event of a rod withdrawal accident or a start-up accident and that the water level is above the minimum detectable level.

The requirement that 2 of the 3 emergency-powered pressurizer heaters be operable provides assurance that sufficient heater capacity (≥126 kw) is available to provide reactor coolant system pressure control during a loss of off-site power.

The requirement that the safety rod groups be fully withdrawn before criticality ensures shutdown capability during startup. This does not prohibit rod latch confirmation, i.e., withdrawal by group to a maximum of 3 inches withdrawn of all seven groups prior to safety rod withdrawal.

The requirement for regulating rods being within their rod position limits ensures that the shutdown margin and ejected rod criteria at hot zero power are not violated.

REFERENCES

(1) FSAR, Section 3

(2) FSAR, Section 3.2.2.1.5

3.1.4 Reactor Coolant System Activity

Specification

MODES 1&2 & MODE 3 w/ RCS T ≥ 500F.

3.4.12 LCO & APPL

3.1.4.1 ~~Whenever the reactor is operating under steady-state conditions,~~ the following conditions shall be met. (L7)

LCO b <LATER (1,0)

a. The total specific activity of the primary coolant shall not exceed 72/E μCi/gm ~~where E is the sum of the average beta energy and average gamma energy per disintegration in MEV/disintegration.~~ — LATER & (LA1) Bases

LCO a

b. The I-131 dose equivalent of the radioiodine activity in the primary coolant shall not exceed 3.5 μCi/gm.

3.4.12 RA A.1 RA B.1

c. If the radioactivity in the primary coolant exceeds the limits given above, ~~corrective action shall be taken immediately to return the coolant activity to within these specifications.~~ (L18) If the specific activity limits given above cannot be achieved within 24 hours, the reactor shall be ~~brought to a hot shutdown condition using normal operating procedures.~~ (M12) In MODE 3 with Tavg <500°F within 6 hours ~~If the coolant radioactivity is not reduced to acceptable limits within an additional 48 hours, the reactor shall be brought to a cold shutdown condition and the cause of the out-of-specification operation ascertained.~~ (L7)

Bases

(A2)

~~Rupture of a steam generator tube would allow primary coolant activity to enter the secondary coolant. The major portion of this activity is noble gases and would be released to the atmosphere from the condenser vacuum pump or a relief valve. Activity would continue to be released until the operator could reduce the primary system pressure below the setpoint of the secondary relief valves and could isolate the faulty steam generator. The worst credible set of circumstances is considered to be a double-ended break of a single steam generator tube, followed by isolation of the faulty steam generator within 34 minutes after the tube break. Assuming the full differential pressure across the steam generator, no more than one-quarter of the total primary coolant could be released to the secondary coolant in this period. The decay heat during this period of 1 hour for pressure reduction will generate steam in the secondary system representing less than 15 weight percent of the secondary system.~~

~~The parameters assumed in the dose analysis for the single steam generator tube failure included the following values:~~

1) ~~total primary coolant volume (mass) = 5.2×10^5 lbs.~~

2) ~~total secondary coolant volume (mass) = 2×10^6 lbs.~~

3) ~~leakage rate from primary to secondary system = 1 gpm.~~

4) ~~fission product decay heat energy for 1 hour = 1.56×10^8 BTU.~~

A2

5) steam mass released to environs = 2.84 x 10^5 lbs.

6) primary coolant released to secondary (34 minutes) = 8.7 x 10^4 lbs.

7) minimum primary to secondary iodine equilibrium activity ratio = 20 to 1 (for 1 gpm leakage).

8) specific I-131 dose equivalent activity = 3.5 μCi/gm (Primary)
= 0.17 μCi/gm (Secondary).

9) gross specific activity in primary = $72/\bar{E}$ μCi/gm.

10) X/Q = 7.0 x 10^{-4} sec/m³ at limiting point beyond site boundary of 1046 meters for 30 m release height - equivalent to ground level release due to topography including building wake effect for 5 percentile meteorology.

11) total gross radioactivity in primary coolant released to secondary coolant released to environs.

12) ten percent of the combined radioiodine activity from primary activity in secondary coolant and secondary activity present in steam mass (released to environs) assumed released to environs.

The whole body dose resulting from immersion in the cloud containing the released activity would include both gamma and beta radiation. The gamma dose is dependent on the finite size and configuration of the cloud. However, the analysis employed the simple model of the semi-infinite cloud, which gives an upper limit to the potential gamma dose. The semi-infinite cloud model is applicable to the beta dose, because of the short range of beta radiation in air. The resulting whole body dose was determined to be less than 0.5 Rem for this accident.

The thyroid dose from the steam generator tube rupture accident has been analyzed assuming a tube rupture at full load and loss of offsite power at the time of the reactor trip, which results in steam release through the relief valves in the period before the faulty steam generator is isolated and primary system pressure is reduced. The limiting iodine activities for the primary and secondary systems are used in the initial conditions. One-tenth of the iodine contained in the liquid which is converted to steam and passed through the relief valves is assumed to reach the site boundary. The resulting thyroid dose from the combined primary and secondary iodine activity released to the environs was determined to be 1.5 Rem for this accident.

The limit for secondary iodine activity is consistent with the limits on primary system iodine activity and primary-to-secondary leakage of 1 gpm. If the activity should exceed the specified limits following a power transient, the major concern would be whether additional fuel defects had developed bringing the total to above expected levels. From the observed removal of excess activity by decay and cleanup, it should be apparent whether activity is returning to a level below the specification limit. Appropriate action to be taken to bring the activity within specification include one or more of the following: gradual decrease in power to a lower base power, increase in letdown flow rate, and venting of the makeup tank gases to the waste gas decay tanks.

3.4B-25

3.1.6 **Leakage** <Add 3.4.13 Appl.> M13

Specification

identified L14

3.4.13 LCO c

3.4.13 RA B.1, C.1, C.2

3.1.6.1 If the ~~total~~ reactor coolant leakage rate exceeds 10 gpm, the reactor shall be ~~shutdown within 24 hours of detection.~~ restored in 18 hours or in MODE 3 in 6 hours and in MODE 5 in 36 hours M13

3.4.13 LCO b

See page 27-2

3.4.13 RA B.1, C.1, C.2

3.1.6.2 If unidentified reactor coolant leakage (exceeding normal evaporative losses) exceeds 1 gpm or if any reactor coolant leakage is evaluated as unsafe, the reactor shall be ~~shut-down within 24 hours of detection.~~ restored in 18 hours or in MODE 3 in 6 hours and in MODE 5 in 36 hours A11 M13

3.4B-24

3.4.13 LCO a

<LATER> (1.0)

3.4.13 RA C.1, C.2

3.1.6.3.a If it is determined that any reactor coolant leakage exists through a non-isolable fault in a reactor coolant system strength boundary (such as the reactor vessel, piping, valve body, etc., except steam generator tubes), the reactor shall be shutdown and a cooldown to the cold shutdown condition shall be initiated within 24 hours of detection. in MODE 3 in 6 hours and in MODE 5 in 36 hours. <LATER> M13 LA1 BASES

3.4.13 LCO d

3.4.13 RA A.1, C.1, C.2

3.1.6.3.b If the leakage through the tubes of any one steam generator equals or exceeds 150 gallons per day ~~(0.104 gpm)~~, a reactor shutdown shall be initiated within 4 hours and the reactor shall be in the cold shutdown condition within the next 30 hours. Restore in 4 hours or be in MODE 3 in 6 hours and in MODE 5 in 36 hours L3 M13

3.1.6.4 ~~Deleted~~ A1

3.4B-25 See page 27-2

3.1.6.5 Action to evaluate the safety implication of reactor coolant leakage shall be initiated within 4 hours of detection. The nature, as well as the magnitude of the leak, shall be considered in this evaluation. The safety evaluation shall assure that the exposure of offsite personnel to radiation is within the guidelines of 10CFR20.

3.1.6.6 ~~If reactor shutdown is required per Specification 3.1.6.1, 3.1.6.2, or 3.1.6.3 the reactor shall not be restarted until the leak is repaired or until the problem is otherwise corrected.~~ A4

In MODES 1, 2, 3 & 4, M13

3.4.15 LCO

3.4.15 Appl

3.4.15 RA B.2

3.4.15 RA B.1.1

3.4.15 RA C.1

3.4.15 RA C.2

3.1.6.7 ~~When the reactor is at power operation~~ ~~three~~ reactor coolant leak detection systems of different operating principles shall be in operation. One of these systems is sensitive to radioactivity and consists of a radioactive gas detector ~~and~~ or an air particulate activity detector. Both of these instruments may be out-of-service simultaneously for a period of no more than 72 hours provided two other means are available to detect leakage and reactor building air samples are taken and analyzed in the laboratory at least once ~~per shift~~; otherwise, be in at least ~~Hot Standby~~ MODE 3 within the next 6 hours and in ~~Cold Shutdown~~ MODE 5 within the following 30 hours. L13 L13 M6

See page 27-2

3.1.6.8 Loss of reactor coolant through reactor coolant pump seals and system valves to connecting systems which

<Add 3.4.15 Cond. A & RA B.1.2 with Note> L13

<Add 3.4.15 Actions Note> L13

<Add 3.4.15 Cond. D> M15

3.1.6 **Leakage**

Specification

3.1.6.1 If the total reactor coolant leakage rate exceeds 10 gpm, the reactor shall be shutdown within 24 hours of detection. L19

3.4B-25

3.1.6.2 If unidentified reactor coolant leakage (exceeding normal evaporative losses) exceeds 1 gpm or if any reactor coolant leakage is evaluated as unsafe, the reactor shall be shut-down within 24 hours of detection. LA2 TRM

3.1.6.3.a If it is determined that any reactor coolant leakage exists through a non-isolable fault in a reactor coolant system strength boundary (such as the reactor vessel, piping, valve body, etc., except steam generator tubes), the reactor shall be shutdown and a cooldown to the cold shutdown condition shall be initiated within 24 hours of detection.

See page 27-1

3.1.6.3.b If the leakage through the tubes of any one steam generator equals or exceeds 150 gallons per day (0.104 gpm), a reactor shutdown shall be initiated within 4 hours and the reactor shall be in the cold shutdown condition within the next 30 hours.

3.1.6.4 Deleted

3.4B-25

3.1.6.5 Action to evaluate the safety implication of reactor coolant leakage shall be initiated within 4 hours of detection. The nature, as well as the magnitude of the leak, shall be considered in this evaluation. The safety evaluation shall assure that the exposure of offsite personnel to radiation is within the guidelines of 10CFR20. LA2 TRM

See page 27-1

3.1.6.6 If reactor shutdown is required per Specification 3.1.6.1, 3.1.6.2, or 3.1.6.3 the reactor shall not be restarted until the leak is repaired or until the problem is otherwise corrected. L19

See page 27-1

3.1.6.7 When the reactor is at power operation, three reactor coolant leak detection systems of different operating principles shall be in operation. One of these systems is sensitive to radioactivity and consists of a radioactive gas detector and an air particulate activity detector. Both of these instruments may be out-of-service simultaneously for a period of no more than 72 hours provided two other means are available to detect leakage and reactor building air samples are taken and analyzed in the laboratory at least once per shift; otherwise, be in at least Hot Standby within the next 6 hours and in Cold Shutdown within the following 30 hours.

3.1.6.8 Loss of reactor coolant through reactor coolant pump seals and system valves to connecting systems which LA2 TRM M5

3.4.13
3.4.14
3.4.15

<Add 3.4.14 Appl> A7

<Add 3.4.14 ACTIONS Note 1> A7

~~vent to the gas vent header and from which coolant can be returned to the reactor coolant system shall not be considered as reactor coolant leakage and shall not be subject to the consideration of Specifications 3.1.6.1 and 3.1.6.6 except that such losses when added to leakage shall not exceed 30 gpm.~~ M5 LA2 TRM L19

3.1.6.9 — 3.4.14 RA A.1 — ACTIONS Note 2 — RA C.1/C.2

If the reactor coolant system pressure isolation valve leakage is greater than the values given in Table 3.1.6.9, isolate (by having at least two valves in the high pressure piping closed*) the high pressure portion of the affected system from the low pressure portion within 4 hours and apply Specification 3.3.6, or be in at least ~~hot shutdown~~ MODE 3 within the next 6 hours and in ~~cold shutdown~~ MODE 5 within the following 30 hours. A1

Bases

A2

~~Every reasonable effort will be made to reduce reactor coolant leakage, including evaporative losses (which may be on the order of 0.5 gpm), to prevent a large leak from masking the presence of a smaller leak. Reactor building sump level, water inventory balances, radiation monitoring equipment, boric acid crystalline deposits, and physical inspections can disclose reactor coolant leaks. Any leak of radioactive fluid, whether from the reactor coolant system primary boundary or not can be a serious problem with respect to in-plant radioactive contamination and cleanup or it could develop into a still more serious problem; and therefore, the first indication of such leakage will be followed up as soon as practicable.~~

~~Although some leak rates on the order of GPM may be tolerable from a dose point of view, especially if they are to closed systems, it must be recognized that leaks on the order of drops per minute through any of the walls of the primary system could be indicative of materials failure such as by stress corrosion cracking. If depressurization, isolation and/or other safety measures are not taken promptly, these small leaks could develop into much larger leaks, possibly into a gross pipe rupture. Therefore, the nature of the leak, as well as the magnitude of the leakage must be considered in the safety evaluation.~~

~~When the source of leakage has been identified, the situation can be evaluated to determine if operation can safely continue. This evaluation will be performed by the Operating Staff and will be documented in writing and approved by the Superintendent. Under these conditions, an allowable reactor coolant system leakage rate of 10 gpm has been established. This explained leakage rate of 10 gpm is also available even during a loss of off-site power.~~

~~If leakage is to the reactor building it may be identified by one or more of the following methods:~~

a. ~~Leakage is monitored by a level indicator in the reactor building sump. Changes in normal sump level may be indicative of leakage from any of the systems located inside the reactor building such as the reactor coolant system, service water system, intermediate cooling system and steam and feedwater lines or condensation of humidity within the reactor building atmosphere. The reactor building sump contains 63.6 gallons per inch of height. A 1 gpm leak would be detected in less than 1 hour.~~

3.4.14 RA A.1 — *The motor operated valve shall remain closed and ~~power supplies deenergized~~ deactivated. A1

3.4.13
3.4.14
3.4.15

A2

b. Total reactor coolant system leakage rate is periodically determined by comparing indications of reactor power, reactor coolant temperature, pressurizer water level and reactor coolant makeup tank level over a time interval. All of these indications are recorded. Since the pressurizer level is maintained essentially constant by the pressurizer level controller, any coolant leakage is replaced by coolant from the reactor coolant makeup tank resulting in a tank level decrease. The reactor coolant makeup tank capacity is 31 gallons per inch of height and each graduation on the level recorder represents 2 inches of tank height. This inventory monitoring method is capable of detecting changes on the order of 62 gallons. A 1 gpm leak would therefore be detectable within approximately 1.1 hours.

As described above, in addition to direct observation, the means of detecting reactor coolant leakage are based on different principles, i.e., activity, sump level and reactor coolant inventory measurements. Two systems of different principles provide, therefore, diversified ways of detecting leakage to the reactor building.

c. The reactor building gaseous monitor is sensitive to low leak rates if expected values of failed fuel exist. The rates of reactor coolant leakage to which the instrument is sensitive are discussed in FSAR Section 4.2.3.8.

The upper limit of 30 gpm is based on the contingency of a hypothetical loss of all AC power. A 30 gpm loss of water in conjunction with a hypothetical loss of all AC power and subsequent cooldown of the reactor coolant system by the atmospheric dump system and steam driven emergency feedwater pump would require more than 60 minutes to empty the pressurizer from the combined effect of system leakage and contraction. This will be ample time to restore both electrical power to the station and makeup flow to the reactor coolant system.

The steam generator tube leakage limit (i.e., primary to secondary leakage limit) in Specification 3.1.6.3 is intended to assure timely shutdown of the plant for appropriate corrective action before rupture of the steam generator tubes occurs under normal operating or postulated accident conditions. These limits also serve to provide added assurance that the dosage contribution from tube leakage will be limited to a small fraction of 10CFR100 limits for a design basis steam generator tube rupture or main steam line break event.

<u>References</u>

FSAR Section 4.2.3.8

< Add SR 3.4.14.1, Note > L16

SR 3.4.14.1

LA1 Bases

TABLE 3.1.6.9

PRIMARY COOLANT SYSTEM PRESSURE ISOLATION VALVES

System	Valve No.	Maximum (a)(b)(c) Allowable Leakage
Decay Heat Removal Train A	DH-14A	≦ 5.0 GPM
	DH-13A) AND) DH-17)	≦ 5.0 GPM (~~both valves together~~ total)
Decay Heat Removal Train B	DH-14B	≦ 5.0 GPM
	DH-13B) AND) DH-18)	≦ 5.0 GPM (~~both valves together~~ total)

A1

Footnote:

(a) 1. Leakage rates less than or equal to 1.0 gpm are considered acceptable.

2. Leakage rates greater than 1.0 gpm but less than or equal to 5.0 gpm are considered acceptable if the latest measured rate has not exceeded the rate determined by the previous test by an amount that reduces the margin between measured leakage rate and the maximum permissible rate of 5.0 gpm by 50% or greater.

3. Leakage rates greater than 1.0 gpm but less than or equal to 5.0 gpm are considered unacceptable if the latest measured rate exceeded the rate determined by the previous test by an amount that reduces the margin between measured leakage rate and the maximum permissible rate of 5.0 gpm by 50% or greater.

L5

SR 3.4.14.1 4. Leakage rates greater than 5.0 gpm are considered unacceptable.

SR 3.4.14.1 (b) Minimum differential test pressure shall not be less than 150 psig.

(c) To satisfy ALARA requirements, leakage may be measured indirectly (as from the performance of pressure indicators) if accomplished in accordance with approved procedures and supported by computations showing that the method is capable of demonstrating valve compliance with the leakage criteria.

LA1 Bases

3.5 INSTRUMENTATION SYSTEMS

3.5.1 Operational Safety Instrumentation

Applicability

Applies to unit instrumentation and control systems.

Objectives

To delineate the conditions of the unit instrumentation and safety circuits necessary to assure reactor safety.

Specifications

A1

<LATER> (3.3A, 3.3B, 3.3C, 3.3D)

3.5.1.1 Startup and operation are not permitted unless the requirements of Table 3.5.1-1, columns 3 and 4 are met.

3.5.1.2 In the event the number of protection channels operable falls below the limit given under Table 3.5.1-1, Columns 3 and 4, operation shall be limited as specified in Column 5.

A10 & LATER

<LATER> (3.3A)

3.5.1.3 For on-line testing or in the event of a protection instrument or channel failure, a key operated channel bypass switch associated with each reactor protection channel may be used to lock the channel trip relay in the untripped state as indicated by a light. Only one channel shall be locked in the untripped state or contain inoperable functions in the untripped state at any one time. In the event more than one protection channel contains inoperable functions in the untripped state, or a protection channel or function becomes inoperable concurrent with another protection channel locked in the untripped state, within 1 hour implement the actions required by Table 3.5.1-1 Note 6. Only one channel bypass key shall be accessible for use in the control room. While operating with an inoperable function unbypassed in the untripped state, the remaining RPS key operated channel bypass switches shall be tagged to prevent their operation.

LATER

3.5.1.4 The key operated shutdown bypass switch associated with each reactor protection channel shall not be used during reactor power operation except during channel testing.

R TRM

<LATER> (3.3A)

3.5.1.5 During startup when the intermediate range instruments come on scale, the overlap between the intermediate range and the source range instrumentation shall not be less than one decade. If the overlap is less than one decade, the flux level shall be maintained in the source range until the one decade overlap is achieved.

3.5.1.6 In the event that one of the trip devices in either of the sources supplying power to the control rod drive mechanisms fails in the untripped state, the power supplied to the rod drive mechanisms through the failed trip device shall be manually removed within 30 minutes following detection. The condition will be corrected and the remaining trip devices shall be tested within eight hours following detection. If the condition is not corrected and the remaining trip devices are not tested within the eight-hour period, the reactor shall be placed in the hot shutdown condition within an additional four hours.

LATER

3/B-22

3.5.1.7 SR 3.4.14.3 The Decay Heat Removal System isolation valve closure setpoints shall be equal to or less than 340 psig for one valve and equal to or less than 400 psig for the second valve in the suction line. ~~The relief valve setting for the DHR system shall be equal to or less than 450 psig.~~ LA1 TRM

<LATER> (3.3D) ~~3.5.1.8 The degraded voltage monitoring relay settings shall be as follows:~~ LATER

~~a. The 4.16 KV emergency bus undervoltage relay setpoints shall be >3115 VAC but <3177 VAC.~~

~~b. The 460 V emergency bus undervoltage relay setpoints shall be >423 VAC but <431 VAC with a time delay setpoint of 8 seconds ±1 second.~~

<LATER> (3.3 A) ~~3.5.1.9 The following Reactor Trip circuitry shall be operable as indicated:~~ LATER

~~1. Reactor trip upon loss of Main Feedwater shall be operable (as determined by Specification 4.1.a and item 35 of Table 4.1-1) at greater than 5% reactor power. (May be bypassed up to 10% reactor power.)~~

~~2. Reactor trip upon Turbine Trip shall be operable (as determined by Specification 4.1.a and item 41 of Table 4.1-1) at greater than 5% reactor power. (May be bypassed up to 45% reactor power.)~~

~~3. If the requirements of Specifications 3.5.1.9.1 or 3.5.1.9.2 cannot be met, restore the inoperable trip within 12 hours or bring the plant to a hot shutdown condition.~~

3.5.1.10 ~~Deleted~~ A1

<LATER> (3.3C) ~~3.5.1.11 For on-line testing of the Emergency Feedwater Initiation and Control (EFIC) system channels during power operation only one channel shall be locked into "maintenance bypass" at any one time. If one channel of the NI/RPS is in maintenance bypass, only the corresponding channel of EFIC may be bypassed.~~ LATER

<LATER> (3.3 D) ~~3.5.1.12 The Containment High Range Radiation Monitoring instrumentation shall be operable with a minimum measurement range from 1 to 10^7 R/hr.~~ LATER

A2

Removal of a module required for protection from a RPS channel will cause that channel to trip, unless that channel has been bypassed, so that only one channel of the other three must trip to cause a reactor trip. Thus, sufficient redundancy has been built into the system to cover this situation.

Removal of a module required for protective action from an analog ESAS channel will cause that channel to trip, so that only one of the other two must trip to actuate the safeguards systems. Removal of a module required for protective action from a digital ESAS subsystem will not cause that subsystem to trip. The fact that a module has been removed will be continuously annunciated to the operator. The redundant digital subsystem is still sufficient to indicate complete ESAS action.

The testing schemes of the RPS, the ESAS, and the EFIC enables complete system testing while the reactor is operating. Each channel is capable of being tested independently so that operation of individual channels may be evaluated.

The EFIC is designed to allow testing during power operation. One channel may be placed in key locked "maintenance bypass" prior to testing. This will bypass only one channel of EFW initiate logic. An interlock feature prevents bypassing more than one channel at a time. In addition, since the EFIC receives signals from the NI/RPS, the maintenance bypass from the NI/RPS is interlocked with the EFIC. If one channel of the NI/RPS is in maintenance bypass, only the corresponding channel of EFIC may be bypassed. Prior to placing a channel of EFIC in maintenance bypass, any NI/RPS channel containing inoperable functions in the untripped state is evaluated for its effect on EFIC. Only the EFIC channel corresponding to the NI/RPS channel containing the inoperable function may be placed in maintenance bypass unles it can be shown that the failure in the NI/RPS channel has no effect on EFIC actuation, actions are taken to ensure EFIC actuation when required, or the appropriate actions of Table 3.5.1-1 are implemented. The EFIC can be teste from its input terminals to the actuated device controllers. A test of the EFIC trip logic will actuate one of two relays in the controllers. Activati of both relays is required in order to actuate the controllers. The two relays are tested individually to prevent automatic actuation of the compone The EFIC trip logic is two (one-out-of-two).

Reactor trips on loss of all main feedwater and on turbine trips will sense the start of a loss of OTSG heat sink and actuate earlier than other trip signals. This early actuation will provide a lower peak RC pressure during the initial over pressurization following a loss of feedwater or turbine trip event. The LOFW trip may be bypassed up to 10% to allow sufficient margin for bringing the MFW pumps into use at approximately 7%. The Turbine Trip may be bypassed up to 45% based on BAW-1893, "Basis for Raising Arming Threshold for Anticipatory Reactor Trip on Turbine Trip," October 1985 and the NRC Safety Evaluation Report for BAW-1893 issued from Mr. D. M. Crutchfield to Mr. J. H. Taylor via letter dated April 25, 1986.

The Automatic Closure and Isolation System (ACI) is designed to close the Decay Heat Removal System (DHRS) return line isolation valves when the Reactor Coolant System (RCS) pressure exceeds a selected fraction of the DHRS design pressure or when core flooding system isolation valves are opened. The ACI is designed to permit manual operation of the DHRS return line isolation valves when permissive conditions exist. In addition, the ACI is designed to disallow manual operation of the valves when permissive conditions do not exist.

TABLE 3.5.1-1 (cont'd)

EMERGENCY FEEDWATER INITIATION
AND CONTROL SYSTEM (cont'd)

FUNCTIONAL UNIT	1 No. of channels	2 No. of channels for system trip	3 Min. operable channels	4 Min. degree of redundancy	5 Operator action if conditions of column 3 or 4 cannot be met
OTHER SAFETY RELATED SYSTEMS					
3.4.14 LCO 1. Decay heat removal system isolation valve automatic closure and interlock system					LA2 TRM A1
a. ~~Reactor coolant pressure instrument channels~~	~~2~~	~~1~~	~~2~~	~~1~~	Notes 1, 5
b. ~~Core flood isolation valve interlocks~~	~~2~~	~~1~~	~~2~~	~~1~~	Notes 1, 5

<Add 3.4.14 LCO Note> L8

3.4.14

TABLE 3.5.1-1 (Cont'd)

3.4.14 Cond. B&C & <LATER> (3.3A, 3.3B, 3.3C, 3.3D)

Notes:

1. Initiate a shutdown using normal operating instructions and place the reactor in the hot shutdown condition within 12 hours if the requirements of Columns 3 and 4 are not met. {MODE 3} L8 & LATER M 14 A1

<LATER> (3.3A)

2. When 2 of 4 power range instrument channels are greater than 10% rated power, hot shutdown is not required.

3. When 1 of 2 intermediate range instrument channels is greater than 10-10 amps, hot shutdown is not required.

4. For channel testing, calibration, or maintenance, the minimum number of operable channels may be two and a degree of redundancy of one for a maximum of four hours, after which Note 1 applies. LATER

3.4.14 Cond B&C <LATER> (3.3 B)

5. If the requirements of Columns 3 or 4 cannot be met within an additional 48 hours, place the reactor in the cold shutdown condition within 24 hours. {MODE 5} L8 M 14 & LATER A1

<LATER> (3.3A, 3.3B & C)

6. The minimum number of operable channels may be reduced to 2, provided that the system is reduced to 1 out of 2 coincidence by tripping the remaining channel. Otherwise, the actions required by Column 5 shall apply. LATER

<LATER> (3.3A)

7. These channels initiate control rod withdrawal inhibits not reactor trips at ~10% rated power. Above 10% rated power, those inhibits are bypassed. LATER

<LATER> (3.3 B)

8. If any one component of a digital subsystem is inoperable, the entire digital subsystem is considered inoperable. Hence, the associated safety features are inoperable and Specification 3.3 applies. LATER

<LATER> (3.3 D)

9. The minimum number of operable channels may be reduced to one and the minimum degree of redundancy to zero for a maximum of 24 hours, after which Note 1 applies.

10. With the number of operable channels less than required, either restore the inoperable channel to operable status within 30 days, or be in hot shutdown within 12 hours.

11. With the number of operable channels less than required, isolate the electromatic relief valve within 4 hours, otherwise Note 9 applies. LATER

3.4.14

Table 4.1-1 (Cont'd)

	Channel Description	Check	Test	Calibrate	Remarks	
⟨LATER⟩ (3.3B)	20. Reactor Building Spray System System Logic Channels	NA	M(1)	NA	(1) Including RB spray pump, spray valves, and chem. add. valve logic channels.	LATER
	21. Reactor Building Spray System Analog Channels					
	a. Reactor Building Pressure Channels	NA	M	R		LATER
⟨LATER⟩ (3.3D)	22. Pressurizer Temperature Channels	S	NA	R		
⟨LATER⟩ (3.1)	23. Control Rod Absolute Position	S(1)	NA	R	(1) Compare with Relative Position Indicator.	LATER
	24. Control Rod Relative Position	S(1)	NA	R	(1) Check with Absolute Position Indicator	
⟨LATER⟩ (3.5)	25. Core Flooding Tanks					LATER
	a. Pressure Channels	S	NA	R		
	b. Level Channels	S	NA	R		
⟨LATER⟩ (3.3D)	26. Pressurizer Level Channels	S	NA	R		LATER
	27. Makeup Tank Level Channels	D	NA	R		
	28. Radiation Monitoring Systems other than containment high range monitors (item 57)				(1) Check functioning of self-checking feature on each detector.	
3.4.15 & LATER (3.3D)	a. Process Monitoring System (RCS Leakage monitors only)	S SR3.4.15.1	Q SR3.4.15.2	R SR 3.4.15.3		& LATER
⟨LATER⟩ (3.3D)	b. Area Monitoring System	S	M(1)	R		LATER
	c. Main Steam Line Radiation Monitors	S	M	R		

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3.4.15

Table 4.1-1 (Cont.)

Channel Description	Check	Test	Calibrate	Remarks
29. High and Low Pressure Injection Systems: Flow Channels	NA	NA	R	
30. Decay heat removal system isolation valve automatic closure and interlock system	S(1)(2)	M(1)(3)	R	(1) Includes RCS Pressure Analog Channel (2) Includes CFT Isolation Valve Position (3) At least once every refueling shutdown, with Reactor Coolant System Pressure greater than or equal to 200 psig, but less than 300 psig, verify automatic isolation of the decay heat removal system from the Reactor Coolant System on high Reactor Coolant System pressure.
31. Deleted				
32. Diesel generator protective relaying starting interlocks and circuitry	M	Q	NA	
33. Off-site power undervoltage and protective relaying interlocks and circuitry	W	R(1)	R(1)	(1) Shall be tested during refueling shutdown to demonstrate selective load shedding interlocks function during manual or automatic transfer of Unit 1 auxiliary load to Startup Transformer No. 2
34. Borated water storage tank level indicator	W	NA	R	
35. Reactor trip upon loss of main feedwater circuitry	M	PC	R	

Handwritten annotations: ⟨LATER⟩ (3.3 D) beside item 29; LATER; L4; ⟨SR 3.4.14.2 SR 3.4.14.4 & SR 3.4.14.5⟩; 3.4.14 beside item 30; ⟨SR 3.4.14.3⟩; LA2 TRM; A1 beside item 31; ⟨LATER⟩ (3.8) beside items 32–33; LATER; ⟨LATER⟩ (3.3 D) LATER beside item 34; ⟨LATER⟩ (3.3A) LATER beside item 35; 3.4.14 in right margin.

Table 4.1-1 (Cont.)

	Channel Description	Check	Test	Calibrate	Remarks
<LATER> (3.3 B)	43. ~~ESAS Manual Trip Functions~~				
	a. ~~Switches & Logic~~	~~NA~~	~~R~~	~~NA~~	LATER
	b. ~~Logic~~	~~NA~~	~~M~~	~~NA~~	
<LATER> (3.3 A)	44. ~~Reactor Manual Trip~~	~~NA~~	~~P~~	~~NA~~	LATER
3.4.15	45. Reactor Building Sump Level	NA	NA	R SR 3.4.15.4	
<LATER> (3.3 D)	46. ~~EFW Flow Indication~~	~~M~~	~~NA~~	~~R~~	LATER

3.4.15

Table 4.1-1 (Cont.)

Channel Description	Check	Test	Calibrate	Remarks
d. SG A High Range Level High-high	S	M	R	LATER
e. SG B High Range Level High-high	S	M	R	
57. Containment High Range Radiation Monitors	D	M	R	LATER
58. Containment Pressure-High	M	NA	R	
59. Containment Water Level-Wide Range	M	NA	R	
60. Low Temperature Overpressure Protection Alarm Logic	NA	R	R	LA2 TRM
61. Core-exit Thermocouples	M	NA	R	LATER
62 Electronic (SCR) Trip Relays	NA	Q	NA	LATER
63 RVLMS	M	NA	R	LATER
64 HLLMS	M	NA	R	

<LATER> (3.3C)
<LATER> (3.3D)
<LATER> (3.3D)
<LATER> (3.3A)
<LATER> (3.3D)

NOTE:

S - Each Shift
W - Weekly
M - Monthly
D - Daily

T/W - Twice per Week
Q - Quarterly
P - Prior to each startup if not done previous week
B/M - Every 2 months

R - Once every 18 months
PC - Prior to going Critical if not done within previous 31 days
NA - Not Applicable
SA - SA Twice per Year

+<LATER> 3.3A 3.3B 3.3C 3.3D

A13 + LATER + R TRM

Amendment No. ~~91~~, ~~94~~, ~~95~~, ~~116~~, ~~117~~, ~~135~~, ~~151~~, 194

3.4.11

3.4.9
3.4.10
3.4.13
3.4.14

Table 4.1-2
Minimum Equipment Test Frequency

	Item	Test	Frequency	
⟨LATER⟩ (3.1)	~~1. Control Rods~~	~~Rod Drop Times of all Full Length Rods 1/~~	~~Each Refueling Shutdown~~	LATER
	~~2. Control Rod Movement~~	~~Movement of Each Rod~~	~~Every Two Weeks Above Cold Shutdown Conditions~~	
SR 3.4.10.1	3. Pressurizer Code Safety Valves	Setpoint	~~One Valve Every 18 Month~~	LA3 IST
⟨LATER⟩ (3.7)	~~4. Main Steam Safety Valves~~	~~Setpoint~~	~~Four Valves Every 18 Months~~	LATER
	~~5. Refueling System Interlocks~~	~~Functioning~~	~~Start of Each Refueling Shutdown~~	R TRM
SR 3.4.13.1	6a. Reactor Coolant System Leakage	Evaluate ⟨Add SR 3.4.13.1 Note⟩	Daily	L12 M1
SR 3.4.14.1	b. Reactor Coolant System Pressure Isolation Valves	Leakage Test Per Table 3.1.6.9	See Notes 1 & 2	A1 L16
	7. Emergency-powered Pressurizer Heaters	~~Power availability~~	~~Daily~~	LA2 TRM
SR 3.4.9.2		Heater capacity functional test	Every 18 Months	
⟨LATER⟩ (3.6)	~~8. Reactor Building Isolation Trip~~	~~Functioning~~	~~Every 18 Months~~	LATER
⟨LATER⟩ (3.7)	~~9. Service Water Systems~~	~~Functioning~~	~~Every 18 Months~~	LATER
	~~10. Spent Fuel Cooling System~~	~~Functioning~~	~~Every 18 Months when irradiated fuel is in the pool~~	R TRM

⟨LATER⟩ (3.1) ~~1/ Same as tests listed in Section 4.7~~ LATER

LA3 IST

Notes:

SR 3.4.14.1 (1) Leak testing for each valve shall be individually accomplished to demonstrate operability following each refueling, following each time the plant is placed in a cold shutdown condition if testing has not been accomplished in the preceding 9 months, and prior to returning the valve to service after maintenance, repair or replacement. L1 LA3 IST

~~(2) Whenever integrity of a pressure isolation valve listed in Table 3.1.6.9 cannot be demonstrated the integrity of the remaining valve in each high pressure line having a leaking valve shall be determined and recorded daily. In addition, the position of one other valve located in the high pressure piping shall be recorded daily.~~ LA2 TRM

3.4.14

Table 4.1-2 (Cont.)

Minimum Equipment Test Frequency

	Item	Test	Frequency	
SR 3.4.14.2 SR 3.4.14.3 SR 3.4.14.4 SR 3.4.14.5	11. Decay heat removal system isolation valve automatic closure and isolation system	Functioning	~~Each Refueling Shutdown~~ 18 months	L9 M9
<LATER> (5.0)	~~12. Flow limiting annulus on main feedwater line at reactor building penetration~~	~~Verify, at normal operating conditions, that a gap of at least 0.025 inches exists between the pipe and the annulus.~~	~~One year, two years, three years, and every five years thereafter measured from date of initial test.~~	Later
<LATER> (3.7)	~~13. Main steam isolation valves~~	~~a. Exercise through approximately 10% travel~~ ~~b. Cycle~~	~~a. Quarterly~~ ~~b. Every 18 months~~	Later
	~~14. Main feedwater isolation valves~~	~~a. Exercise through approximately 5% travel~~ ~~b. Cycle~~	~~a. Quarterly~~ ~~b. Every 18 months~~	
<LATER> (3.4A)	~~15. Reactor internals vent valves~~	~~Demonstrate operability by:~~ ~~a. Conducting a remote visual inspection of visually accessible surfaces of the valve body and disc sealing faces and evaluating any observed surface irregularities.~~ ~~b. Verifying that the valve is not stuck in an open position, and~~ ~~c. Verifying through manual actuation that the valve is fully open with a force of ≤ 400 lbs (applied vertically upward).~~	~~Each refueling shutdown~~	LATER

Table 4.1-2 (Cont.)

Minimum Equipment Test Frequency

	Item	Test	Frequency	
⟨LATER⟩ (3.4A)	~~16. RCS Vent Paths~~	~~Demonstrate operability by flow verification~~	~~At least once per 18 months during cold shutdown~~	LATER
SR 3.4.11.5	17. PORV	Exercise	~~End of each refueling outage~~ 18 months	L9 M4

3.4B-09

⟨Add SR 3.4.12.2 NOTE⟩ L11

⟨Add SR 3.4.12.3 NOTE⟩ L11

Table 4.1-3

MINIMUM SAMPLING AND ANALYSIS FREQUENCY

	Item	Test	Frequency	
	1. Reactor Coolant Samples	a. ~~Gamma Isotopic Analysis~~	~~a. Bi-weekly (7)~~	LA2 TRM
SR 3.4.12.1		b. Gross Activity Determination	~~b. 3 times/week and at least every third day (1)(6)(7)~~	L11
SR 3.4.12.2		c. Gross Radioiodine Determination	c. ~~Weekly (3)(6)(7)~~	L11
⟨LATER⟩ (3.1)		~~d. Dissolved Gases~~	~~d. Weekly (7)~~	LATER
⟨LATER⟩ (3.4A)		~~e. Chemistry (Cl, F, and O_2)~~	~~e. 3 times/week (8)~~	LATER
⟨LATER⟩ (3.9)		~~f. Boron Concentration~~	~~f. 3 times/week~~	LATER & (R) TRM
SR 3.4.12.3		g. Radiochemical Analysis for Ē Determination (2)(4)	~~g. Monthly (7)~~	L11
⟨LATER⟩ (3.5)	~~2. Borated Water Storage Tank Water Sample~~	~~Boron Concentration~~	~~Weekly and after each makeup~~	LATER
	~~3. Core Flooding Tank Sample~~	~~Boron Concentration~~	~~Monthly and after each makeup~~	
⟨LATER⟩ (3.7)	~~4. Spent Fuel Pool Water Sample~~	~~Boron Concentration~~	~~Monthly and after each makeup (9)~~	LATER
	~~5. Secondary Coolant Samples~~	~~a. Gross Radioiodine Concentration~~	~~a. Weekly (5)(7)(10)~~	(R) TRM
		~~b. Isotopic Radioiodine Concentration (4)~~	~~b. Monthly (7)(10)~~	LATER
⟨LATER⟩ (3.6)	~~6. Sodium Hydroxide Tank Sample~~	~~Sodium Hydroxide Concentration~~	~~Quarterly and after each makeup~~	LATER

Notes:

(1) ~~A gross radioactivity analysis shall consist of the quantitative measurement of the total radioactivity of the primary coolant in units of μCi/gm. The total primary coolant activity shall be the sum of the degassed beta-gamma activity and the total of all identified gaseous activities 15 minutes after the primary system is sampled.~~ LA1 Bases ~~Whenever the gross radioactivity concentration exceeds 10% of the limit specified in the Specification 3.1.4.1 or increases by 10 μCi/gm from the previous measured level, the frequency of sampling and analyzing shall be increased to a minimum of once/day until a steady activity level is established.~~ L11

(2) A radiochemical analysis shall consist of the quantitative measurement the activity for each radionuclide which is identified in the primary coolant 15 minutes after the primary system is sampled. The activities for the individual isotopes shall be used in the determination of $\bar{E}$. A radiochemical analysis and calculation of $\bar{E}$ and iodine isotopic activity shall be performed if the measured gross activity changes by more than 10 μCi/gm from the previous measured level. The gamma energy per disintegration for those radioisotopes determined to be present shall be as given in "Table of Isotopes" (1967) and beta energy per disintegration shall be as given in USNRDL-TR-802 (Part II) or other references using the equivalent values for the radioisotopes. LA1 Bases L11 LA1 Bases

(3) In addition to the weekly measurement, the radioiodine concentration shall be determined if the measured gross radioactivity concentration changes by more than 10 μCi/gm from the previous measured level. L11

& <LATER> (3.7) (4) Iodine isotopic activities shall be weighted to give I-131 dose equivalent activity. LA1 Bases & LATER

(5) In addition to the weekly measurement, the radioiodine concentration shall be determined if there are indications that the primary to secondary coolant leakage rate has increased by a factor of 2. R TRM

(6) Whenever the steady state radioiodine or gross radioactivity concentration of prior operation is greater than 1 percent but less than 10 percent of Specification 3.1.4.1, a sample of reactor coolant shall taken within 24 hours of any reactor criticality and analyzed for radioactive iodines of I-131 through I-135 and gross radioactivity as well as the coolant sample and analyses required by the above. L11

Whenever the steady state radioiodine or gross radioactivity concentration of prior operation is greater than 10 percent of Specification 3.1.4.1, a sample of reactor coolant shall be taken prior to any reactor criticality and analyzed for radioactive iodines of I-131 through I-135 and gross radioactivity as well as the coolant sample and analyses required by above. & LA2 TRM

3.4.12 Appl. & <LATER> (3.1, 3.7) (7) ~~Not required when plant is in the cold shutdown condition or refueling shutdown condition.~~ MODES 1 & 2 & 3 w/ RCS T ≥ 500 F. L7 & LATER & R TRM

<LATER> 3.4A (8) O_2 analysis is not required when plant is in the cold shutdown condition or refueling shutdown condition. LATER

<LATER> (3.7) (9) Required only when fuel is in the pool and prior to transferring fuel to the pool. LATER

<LATER> (3.7) (10) Not required when not generating steam in the steam generators. LATER & R TRM

(11) The following shall be required until the end of Cycle 2 operation: A6

a. Gross radioiodine shall be determined at least three times per week during power operation.

b. ~~If the steady state gross radioiodine concentration increases by a factor of ten or more, the NRC shall be promptly notified with a written followup per Specification 6.12.3.1.~~ A6

SR 3.4.13.2 & <LATER> (5.0)

A5 & LATER

b. The steam generator shall be determined operable after completing the corresponding actions (plug, reroll, or sleeve all tubes exceeding the plugging limit and all tubes containing non-TEC through-wall cracks) required by Table 4.18-2.

Tube 110/60 in the "A" steam generator contains indications in the upper roll transition that exceed the plugging limit. Tube 110/60 may remain in service with these indications for the duration of cycle 16 without rendering the "A" steam generator inoperable.

4.18.6 Reports

Following each inservice inspection of steam generator tubes, the complete results of the inspection shall be reported to the NRC. This report, to be submitted within 90 days of inspection completion, shall include:

a. Number and extent of tubes inspected;

b. Location and percent of wall-thickness penetration for each indication of an imperfection;

c. Identification of tubes plugged and tubes sleeved;

d. Number of tubes repaired by rerolling and number of indications detected in the new roll area of the repaired tubes;

e. Summary of the condition monitoring and operational assessment results when applying TEC alternate repair criteria; and

f. Summary of the condition monitoring and the operational assessment results (including growth) when applying the upper tubesheet ODIGA alternate repair criteria.

This report shall be in addition to a Special Report (per Specification 6.12.5.d) required for the results of steam generator tube inspections which fall into Category C-3 as denoted in Table 4.18-2. The Commission shall be notified of the results of steam generator tube inspections which fall into Category C-3 prior to resumption of plant operation. The written Special Report shall provide a description of investigations conducted to determine cause of the tube degradation and corrective measures taken to prevent recurrence.

"R" - Relocation of requirements:

Relocating requirements which do not meet the Technical Specification selection criteria to documents with an established control program allows the Technical Specifications to be reserved only for those conditions or limitations upon reactor operation which are necessary to adequately limit the possibility of an abnormal situation or event giving rise to an immediate threat to the public health and safety, thereby focusing the scope of Technical Specifications.

Therefore, requirements which do not meet the Technical Specification selection criteria in 10 CFR 50.36 have been relocated to other controlled license basis documents. This regulation addresses the scope and purpose of Technical Specifications. In doing so, it establishes a specific set of objective criteria for determining which regulatory requirements and operating restrictions should be included in Technical Specifications. These criteria are as follows:

Criterion 1: Installed instrumentation that is used to detect and indicate in the control room a significant abnormal degradation of the reactor coolant pressure boundary.

Criterion 2: A process variable that is an initial condition of a design basis accident (DBA) or transient analysis that either assumes the failure of or presents a challenge to the integrity of a fission product barrier.

Criterion 3: A structure, system or component that is part of the primary success path and which functions or actuates to mitigate a design basis accident or transient that either assumes the failure of or presents a challenge to the integrity of a fission barrier.

Criterion 4: A structure, system or component which operating experience or probabilistic safety assessment has shown to be significant to public health and safety.

The application of these criteria is provided in the "Application of Selection Criteria to the ANO-1 Technical Specifications." Requirements which met the criteria have been included in the proposed improved Technical Specifications. Entergy Operations proposes to remove the requirements which do not meet the criteria from the Technical Specifications and relocate the requirements to a suitable owner controlled document. The requirements in the relocated Specifications will not be affected by this Technical Specification change. Entergy Operations will initially continue to perform the required operation and maintenance to assure that the requirements are satisfied. Relocating specific requirements for systems or variables will have no impact on the system's operability or the variable's maintenance, as applicable.

NO SIGNIFICANT HAZARDS CONSIDERATIONS
GENERIC EVALUATIONS

License basis document control mechanisms, such as 10 CFR 50.59, 10 CFR 50.54(a)(3), and ITS Section 5, "Administrative Controls," will be utilized for the relocated Specifications as they will be placed in other controlled license basis documents. This would allow Entergy Operations to make changes to these requirements, without NRC approval, as allowed by the applicable regulatory requirements. These controls are considered adequate for assuring structures, systems and components in the relocated Specifications are maintained operable and variables in the relocated Specifications are maintained within limits.

Entergy Operations has evaluated this proposed Technical Specification change and has determined that it involves no significant hazards consideration. This determination has been performed in accordance with the criteria set forth in 10 CFR 50.92(c) as indicated below:

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 which did not meet the criteria for inclusion in Technical Specifications as identified in the Application of Selection Criteria to the ANO-1 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 license basis document and maintained pursuant to the applicable regulatory requirements. 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 necessitate a physical alteration of the plant (no new or different type of equipment will be installed) or change in parameters governing normal plant operation. The proposed change will not impose any different requirements and adequate control of 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 affected requirement will be relocated to an owner controlled license basis document for which future changes will be evaluated pursuant to the requirements of the applicable regulatory requirements. Therefore, this change does not involve a significant reduction in a margin of safety.

NO SIGNIFICANT HAZARDS CONSIDERATIONS
GENERIC EVALUATIONS

"A" - Administrative changes to requirements:

Reformatting and rewording the remaining requirements in accordance with the style of the improved Babcock & Wilcox Standard Technical Specifications in NUREG-1430 will make the Technical Specifications more readily understandable to plant operators and other users. Application of the format and style will also assure consistency is achieved between specifications. As a result, the reformatting and rewording of the Technical Specifications has been performed to make them more readily understandable by plant operators and other users. During this reformatting and rewording process, no technical changes (either actual or interpretational) to the Technical Specifications were made unless they were identified and justified.

Entergy Operations has evaluated this proposed Technical Specification change and has determined that it involves no significant hazards consideration. This determination has been performed in accordance with the criteria set forth in 10 CFR 50.92(c) as indicated below:

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

   The proposed change involves reformatting and rewording of the existing Technical Specifications. The reformatting and rewording process involves no technical changes to existing requirements. 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 necessitate a physical alteration of the plant (no new or different type of equipment will be installed) or changes in parameters governing normal plant operation. The proposed change will not impose any different 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 significantly reduce the margin of safety because it has no impact on any safety analysis assumptions. This change is administrative in nature. As such, there is no technical change to the requirements and therefore, there is no significant reduction in the margin of safety.

"LA" - Less restrictive, Administrative deletion of requirements:

Portions of some Specifications provide information that is descriptive in nature regarding the equipment, system(s), actions or surveillances. This information is proposed to be deleted from the specification and relocated to other license basis documents which are under licensee control. These documents include the TS Bases, Safety Analysis Report (SAR), Technical Requirements Manual, and Programs and Manuals identified in ITS Section 5, "Administrative Controls." The removal of descriptive information is permissible, because the documents containing the relocated information will be controlled through the applicable process provided by the regulatory requirements, e.g., 10 CFR 50.59, 10 CFR 50.54(a)(3), and ITS Section 5, "Administrative Controls." This will not impact the actual requirements but may provide some flexibility in how the requirement is conducted. Therefore, the descriptive information that has been moved continues to be maintained in an appropriately controlled manner.

Entergy Operations has evaluated this proposed Technical Specification change and has determined that it involves no significant hazards consideration. This determination has been performed in accordance with the criteria set forth in 10 CFR 50.92(c) as indicated below:

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

   The proposed change relocates requirements from the Technical Specifications to other license basis documents which are under licensee control. The documents containing the relocated requirements will be maintained using the provisions of applicable regulatory requirements. 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 necessitate a physical alteration of the plant (no new or different type of equipment will be installed) or changes in parameters governing normal plant operation. The proposed change will not impose any different 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 requirements to be transposed from the Technical Specifications to other license basis documents, which are under licensee control, are the same as the existing Technical Specifications. The documents containing the relocated requirements will be maintained using the provisions of applicable regulatory requirements. Therefore, this change does not involve a significant reduction in a margin of safety.

"M" - More restrictive changes to requirements:

The ANO-1 Technical Specifications are proposed to be modified in some areas to impose more stringent requirements than previously identified. These more restrictive modifications are being imposed to be consistent with the improved Babcock & Wilcox Standard Technical Specifications. Such changes have been made after ensuring the previously evaluated safety analysis was not affected. Also, other more restrictive technical changes have been made to achieve consistency, correct discrepancies, and remove ambiguities from the specification.

The modification of the ANO-1 Technical Specifications and the changes made to achieve consistency within the specifications have been performed in a manner such that the most stringent requirements are imposed, except in cases which are individually evaluated.

Entergy Operations has evaluated this proposed Technical Specification change and has determined that it involves no significant hazards consideration. This determination has been performed in accordance with the criteria set forth in 10 CFR 50.92(c) as quoted below:

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 the ANO-1 Technical Specifications. These more stringent requirements are not assumed to be initiators of analyzed events and will not alter assumptions relative to mitigation of accident or transient events. The change has been confirmed to ensure no previously evaluated accident has been adversely affected. The more stringent requirements are imposed 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 necessitate a physical alteration of the plant (no new or different type of equipment will be installed) or changes in parameters governing normal plant operation. The proposed change does impose different requirements. However, these changes do not impact the safety analysis and licensing basis. Thus, this change does not create the possibility of a new or different kind of accident from any accident previously evaluated for ANO-1.

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

The imposition of more stringent requirements prevents a reduction in the margin of plant safety by:

a) Increasing the analytical or safety limit,
b) Increasing the scope of the specification to include additional plant equipment,
c) Increasing the applicability of the specification,
d) Providing additional actions,
e) Decreasing restoration times,
f) Imposing new surveillances, or
g) Decreasing surveillance intervals.

The change is consistent with the safety analysis and licensing basis. Therefore, this change does not involve a reduction in a margin of safety.

NO SIGNIFICANT HAZARDS CONSIDERATIONS STATEMENTS

ITS Section 3.4B: Reactor Coolant System

Entergy Operations has evaluated these proposed Technical Specification changes and has determined that they involve no significant hazards consideration. This determination has been performed in accordance with the criteria set forth in 10 CFR 50.92(c) as indicated below:

3.4B L1

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

This change revises the Frequency for testing RCS Pressure Isolation Valves (PIVs). This change allows less frequent performances of this Surveillance Requirement. A less frequent performance of a Surveillance Requirement does not result in any hardware changes. The Frequency of performance also does not significantly increase the probability of occurrence for initiation of any analyzed event since the function of the equipment does not change (and therefore any initiation scenarios are not changed). Further, the Frequency of performance of a surveillance does not significantly increase the consequences of an accident because a change in Frequency does not change the assumed response of the equipment in performing its specified mitigation functions from that considered with the original Frequency. Therefore, this change does not involve a significant increase in 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 necessitate a physical alteration of the plant (no new or different type of equipment will be installed) or changes in parameters governing normal plant operation. The proposed change will still ensure proper surveillances are required for the leakage parameter as considered in the safety analysis. 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?**

Changes in the surveilled parameter have been determined to be relatively slow during the proposed intervals, and the proposed Frequency has been determined to be sufficient to identify significant impact on compliance with the assumed conditions of the safety analysis. In addition, other indications continue to be available to indicate potential noncompliance. Therefore, an extended surveillance interval does not involve a significant reduction in the margin of safety.

NO SIGNIFICANT HAZARDS CONSIDERATIONS STATEMENTS

3.4B L2

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

The Applicability for the RCS low temperature overpressure protection (LTOP) requirements is revised to reflect only those conditions under which LTOP is required: MODE 4 with RCS temperature ≤ 262°F, MODE 5, and MODE 6 when the reactor vessel head is on. LTOP is only of concern during low temperature, high pressure conditions when normal overpressure protection capabilities are not available or adequate. Under the excluded conditions, low temperature overpressurization is unlikely since the unit is not in a low temperature condition. As such the proposed change does not significantly 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 necessitate a physical alteration of the plant (no new or different type of equipment will be installed) or changes in parameters governing normal plant operation. The proposed change will continue to ensure the assumed limits for RCS pressurization are met when low temperature conditions exist during which there is potential for the analyzed events. 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 continue to ensure the protection for RCPB integrity is provided when conditions exist during which there is potential for the analyzed events. Therefore, this change does not involve a significant reduction in a margin of safety.

3.4B L3

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

Reactor coolant primary to secondary leakage is an input assumption for dose consequence analyses and is an indicator of increased potential for a steam generator tube rupture event. However, a short extension of the Completion Time for a Required Action does not result in any hardware changes. The Completion Time for performance also does not significantly increase the probability of occurrence for initiation of any analyzed event since the limit for the parameter does not change (and therefore any initiation scenarios are not changed) and the proposed Completion Time extension is short (and therefore does not significantly affect probability). Further, the Completion Time for performance of Required Actions does not significantly increase the consequences of an accident because there is no change in the response of the core parameters to assumed scenarios from that considered during the original Completion Time. Therefore, this change does not involve an increase in 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 necessitate a physical alteration of the plant (no new or different type of equipment will be installed) or changes in parameters governing normal plant operation. The proposed change will still ensure prompt restoration of compliance with the limiting condition for operation, or prompt and appropriate compensatory actions are taken. 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?

This change does not involve a significant reduction in a margin of safety since the parameter (reactor coolant leakage) continues to be evaluated in the same manner. The increase in time allowed for such a evaluation and restoration is minimal and provides additional potential for the preferred action of restoration of the appropriate temperature rather than requiring a shutdown with increased potential for a transient.

3.4B L4

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

This change revises the Frequency for testing the decay heat removal system isolation valve automatic closure and interlock system. This change allows less frequent performances of this Surveillance Requirement. A less frequent performance of a Surveillance Requirement does not result in any hardware changes. The Frequency of performance also does not significantly increase the probability of occurrence for initiation of any analyzed event since the function of the equipment does not change (and therefore any initiation scenarios are not changed) and since the proposed Frequency has been determined to be appropriate to determine accurate leakage results. Further, the Frequency of performance of a surveillance does not significantly increase the consequences of an accident because a change in Frequency does not change the assumed response of the equipment in performing its specified mitigation functions from that considered with the original Frequency. Therefore, this change does not involve a significant increase in 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 necessitate a physical alteration of the plant (no new or different type of equipment will be installed) or changes in parameters governing normal plant operation. The proposed change will still ensure proper surveillances are required for the RCS isolation function as considered in the safety analysis. 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 margin of safety for the RCS isolation is provided through dual barriers, indication of leakage and small leakage limits. None of these are affected by this change. Further, the testing Frequency has been determined to be adequate based on the high reliability of the equipment and on the preference to perform the testing under unit conditions that apply during a unit outage to reduce the potential for an unplanned transient. Therefore, the change of Frequency for this surveillance is not considered to involve a significant reduction in a margin of safety.

3.4B L5

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

Reactor coolant pressure isolation valve leakage is an input assumption for dose consequence analyses and is an indicator of increased potential for a design basis event. However, this change of acceptance criteria does not result in any hardware changes, and also does not significantly increase the probability of occurrence nor significantly increase the consequences of any analyzed event since the overall leakage limit for the penetration does not change (and therefore any initiation scenarios are not changed). Therefore, this change does not involve an increase in 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 necessitate a physical alteration of the plant (no new or different type of equipment will be installed) or changes in parameters governing normal plant operation. The proposed change will still ensure prompt restoration of compliance with the limiting condition for operation, or prompt and appropriate compensatory actions are taken. 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?

This change does not involve a significant reduction in a margin of safety since the parameter (reactor coolant pressure isolation valve leakage) continues to be evaluated in the same manner. The increase in time allowed for such a evaluation and restoration is minimal and provides additional potential for the preferred action of restoration of the appropriate temperature, rather than requiring a shutdown with increased potential for a transient.

3.4B L6

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

3.4B-28

Pressurizer level is an input assumption for many analyses. However, it is not considered as the initiator for any previously analyzed accident. Emergency-powered pressurizer heaters are one method required to maintain RCS pressure control in hot standby conditions during a loss of offsite power. However, they are not considered as the initiator for any previously analyzed accident. A short extension of the Completion Time for a Required Action does not result in any hardware changes. The Completion Time for performance also does not significantly increase the probability of occurrence for initiation of any analyzed event since the limit for the parameter does not change (and therefore any initiation scenarios are not changed) and the proposed Completion Time extension is short (and therefore does not significantly affect probability). Further, the Completion Time for performance of Required Actions does not significantly increase the consequences of an accident because there is no change in the response of the core parameters to assumed scenarios from that considered during the original Completion Time. Therefore, this change does not involve an increase in 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 necessitate a physical alteration of the plant (no new or different type of equipment will be installed) or changes in parameters governing normal plant operation. The proposed change will still ensure prompt restoration of compliance with the limiting condition for operation, or prompt and appropriate compensatory actions are taken. 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?**

This change does not involve a significant reduction in a margin of safety since the parameters (pressurizer level and emergency-powered pressurizer heater capacity) continue to be evaluated in the same manner. The increase in time allowed for such an evaluation and restoration is minimal and provides additional potential for the preferred action of restoration of the appropriate temperature, rather than requiring a shutdown with increased potential for a transient.

3.4B L7

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

The Applicability for the RCS activity limits is revised to omit MODE 3 below 500°F and MODE 4. The RCS activity is primarily of concern for a steam generator tube rupture event, which is unlikely in these conditions since the saturation pressure of the reactor coolant is below the open setting of the atmospheric dump valves and the lift pressure settings of the main steam safety valves. RCS activity is not considered as the initiator of any previously analyzed accident, and the limits for RCS activity are not changed for the previously analyzed accident. As such the proposed change does not significantly 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 necessitate a physical alteration of the plant (no new or different type of equipment will be installed) or changes in parameters governing normal plant operation. The proposed change will continue to ensure the assumed limits for RCS activity are met when conditions exist during which there is potential for the analyzed events. 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 continue to ensure the assumed limits for RCS activity are met when conditions exist during which there is potential for the analyzed events. Therefore, this change does not involve a significant reduction in a margin of safety.

NO SIGNIFICANT HAZARDS CONSIDERATIONS STATEMENTS

3.4B L8

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

The RCS isolation function is provided to reduce the probability of an intersystem LOCA. The proposed change will lengthen the Completion Time for an inoperable decay heat removal system isolation valve automatic closure and interlock system. However, the extension of the Completion Time for a Required Action does not result in any hardware changes, and the function of the equipment does not change. Also, the extension of the Completion Time is short. Therefore, the Completion Time extension does not significantly increase the probability of occurrence of any previously analyzed accident. Further, the Completion Time for performance of Required Actions does not significantly increase the consequences of an accident because there is no change in the assumed response of the equipment in performing its specified mitigation functions and no change in the response of the core parameters to assumed scenarios from that considered during the original Completion Time.

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 necessitate a physical alteration of the plant (no new or different type of equipment will be installed) or changes in parameters governing normal plant operation. The proposed change will still ensure prompt restoration of compliance with the limiting condition for operation, or appropriate compensatory actions are taken. 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?**

This change does not involve a significant reduction in a margin of safety since the inoperable equipment compensatory actions are not revised. The increase in time allowed for evaluation and restoration is minimal and provides additional potential for the preferred action of restoration of the leakage to equipment to OPERABLE status, rather than requiring a shutdown transient.

3.4B L9

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

This change revises the Frequency for testing the electromatic relief valve (ERV) on the pressurizer. This change allows less frequent performances of this Surveillance Requirement. A less frequent performance of a Surveillance Requirement does not result in any hardware changes. The Frequency of performance also does not significantly increase the probability of occurrence for initiation of any analyzed event since the function of the equipment does not change (and therefore any initiation scenarios are not changed) and since the proposed Frequency has been determined to be adequate to demonstrate reliable operation of the equipment. Further, the Frequency of performance of a surveillance does not significantly increase the consequences of an accident because a change in Frequency does not change the assumed response of the equipment in performing its specified mitigation functions from that considered with the original Frequency. Therefore, this change does not involve an increase in 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 necessitate a physical alteration of the plant (no new or different type of equipment will be installed) or changes in parameters governing normal plant operation. The proposed change will still ensure proper surveillances are required for the ERV as considered in the LTOP safety analysis. 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?

Changes in the surveilled parameter have been determined to be relatively slow during the proposed intervals, and the proposed Frequency has been determined to be sufficient to identify significant impact on compliance with the assumed conditions of the safety analysis. In addition, other indications continue to be available to indicate potential noncompliance. Therefore, an extended surveillance interval does not involve a significant reduction in the margin of safety.

3.4B L10

Not Used

NO SIGNIFICANT HAZARDS CONSIDERATIONS STATEMENTS

3.4B L11

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

This change revises the Frequency for determining RCS activity. This change allows less frequent performances of this Surveillance Requirement. A less frequent performance of a Surveillance Requirement does not result in any hardware changes. The Frequency of performance also does not significantly increase the probability of occurrence for initiation of any analyzed event since the function of the equipment does not change (and therefore any initiation scenarios are not changed) and since the proposed Frequency has been determined to be adequate to properly trend the parameter. This proposed Frequency is also acceptable since other indications continue to be available to indicate potential noncompliance during the surveillance interval. Therefore, this change does not involve an increase in 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 necessitate a physical alteration of the plant (no new or different type of equipment will be installed) or changes in parameters governing normal plant operation. The proposed change will still ensure proper surveillances are required for RCS activity as considered in the safety analysis. 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?

Changes in the surveilled parameter have been determined to be relatively slow during the proposed intervals, and the proposed Frequency has been determined to be sufficient to identify significant impact on compliance with the assumed conditions of the safety analysis. In addition, other indications continue to be available to indicate potential noncompliance. Therefore, an extended surveillance interval does not involve a significant reduction in the margin of safety.

3.4B L12

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

This change revises the Frequency for determining RCS leakage by an inventory balance. This change allows less frequent performances of this Surveillance Requirement. A less frequent performance of a Surveillance Requirement does not result in any hardware changes. The Frequency of performance also does not significantly increase the probability of occurrence for initiation of any analyzed event since the function of the equipment does not change (and therefore any initiation scenarios are not changed) and since the proposed Frequency has been determined to be appropriate to determine accurate leakage results. Further, this proposed change in Frequency of performance does not significantly increase the consequences of an accident because other indications remain available to indicate potential non compliance during the surveillance interval. Therefore, this change does not involve a significant increase in 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 necessitate a physical alteration of the plant (no new or different type of equipment will be installed) or changes in parameters governing normal plant operation. The proposed change will still ensure proper surveillances are required for the leakage parameter as considered in the safety analysis. 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?

Changes in the surveilled parameter have been determined to be relatively slow during the proposed intervals, and the proposed Frequency has been determined to be sufficient to identify significant impact on compliance with the assumed conditions of the safety analysis. In addition, other indications continue to be available to indicate potential noncompliance. Therefore, an extended surveillance interval does not involve a significant reduction in the margin of safety.

3.4B L13

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

The RCS leakage limits are provided to assess the structural integrity of the reactor coolant system. However, the limits are not considered the initiator of any previously analyzed accident. Further, an extension of the Completion Time for a Required Action does not result in any hardware changes. The Completion Time for performance also does not significantly increase the probability of occurrence for initiation of any analyzed event since the function of the equipment, or limit for the parameter, does not change. Further, the Completion Time for performance of Required Actions does not significantly increase the consequences of an accident because there is no change in the assumed response of the equipment in performing its specified mitigation functions, and no change in the response of the core parameters to assumed scenarios, from that considered during the original Completion Time.

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 necessitate a physical alteration of the plant (no new or different type of equipment will be installed) or changes in parameters governing normal plant operation. The proposed change will still provide for detection of leakage before the leakage source would propagate to a "break", and ensure prompt restoration of compliance with the limiting condition for operation, or appropriate compensatory actions are taken. 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?

This change does not involve a significant reduction in a margin of safety since the structural integrity of the RCS continues to be evaluated in the same manner. The increase in time allowed for such an evaluation and restoration is minimal and provides additional potential for the preferred action of restoration of the leakage to equipment to OPERABLE status, rather than requiring a shutdown transient.

3.4B L14

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

The RCS leakage is an indication of RCS structural integrity. However, a change in the leakage limit does not require any hardware changes. Additionally, leakage is not considered as the initiator of any previously analyzed accident. Therefore, this change does not involve a significant increase in the probability of any accident previously evaluated. Further, the change in the limit is small, and therefore, this change does not involve a significant increase in 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 necessitate a physical alteration of the plant (no new or different type of equipment will be installed) or changes in parameters governing normal plant operation. The proposed change will still ensure that leakage is within the assumptions of the accident analysis. 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 margin of safety for dose consequences is provided by assuring that the results of the analyses are within the limits of 10 CFR 100. This change will not significantly increase the dose consequences and therefore, the increased limit is not considered to involve a significant reduction in a margin of safety.

3.4B L15

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

The Applicability for the requirement to maintain a single OPERABLE pressurizer safety valve while the reactor is subcritical is revised to include only MODE 3 above the LTOP enable temperature. The pressurizer safety valves provide protection to mitigate the consequences of an overpressurization event. However, the proposed change does not involve a physical alteration of the unit or changes in parameters governing normal plant operation. Therefore, the change does not involve a significant increase in the probability or consequences of any accident previously evaluated. Further, below the LTOP enable temperature the pressurizer safety valve is not used to mitigate overpressurization events. Therefore, the change does not involve a significant increase in 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 necessitate a physical alteration of the plant (no new or different type of equipment will be installed) or changes in parameters governing normal plant operation. The proposed change will still ensure that adequate overpressure protection is provided under conditions where it is appropriate to do so. 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 pressurizer safety valves are not included in the margin of safety for operation below the LTOP enable temperature. Therefore, the omission of requirements for a pressurizer safety valve under these conditions does not involve a significant reduction in a margin of safety.

3.4B L16

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

The Applicability for the requirement to perform RCS pressure isolation valves (PIVs) leakage testing is limited to MODES 1 & 2. This means the testing must be current for entry into and operation in MODES 1 & 2. The PIV leakage is considered as the initiator of an intersystem LOCA. However, since this change in Frequency for performing the test does not change the capability of the PIVs, and the eliminated testing can not be performed such that it provides accurate results, the change does not involve a significant increase in 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 necessitate a physical alteration of the plant (no new or different type of equipment will be installed) or changes in parameters governing normal plant operation. The proposed change will still ensure that the PIV leakage is within its limit under conditions where it is appropriate to do so. 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 margin of safety for PIV leakage is provided through dual barriers, indication of leakage, and small leakage limits. None of these are affected by this change. Further, the eliminated testing can not be performed such that it provides accurate results. Therefore, the limitation on the requirement to perform this surveillance is not considered to involve a significant reduction in a margin of safety.

3.4B L17

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

This change revises the OPERABILITY requirements to permit testing and examination of the pressurizer safety valves at high pressure and temperature near their normal operating range, but only after the valves have had a preliminary cold setting. The cold setting gives assurance that the valves are OPERABLE near their design condition. This change does not result in any hardware changes, but only affects the method for testing to verify the lift settings. The change in method does not significantly increase the probability of occurrence for initiation of any analyzed event since the function of the equipment does not change (and therefore any initiation scenarios are not changed). Further, no significant increase in the consequences of an accident is identified since the performance of the valves continues to be assured by the cold setting such that the assumed response of the equipment in performing its specified mitigation functions is not changed. Therefore, this change does not involve a significant increase in 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 necessitate a physical alteration of the plant (no new or different type of equipment will be installed) or changes in parameters governing normal plant operation. The proposed change will still ensure proper surveillances are required for the lift settings as considered in the safety analysis. 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 method of testing the pressurizer safety valve lift settings has been determined to be sufficient during HOT SHUTDOWN (MODE 4) and for a limited time during HOT STANDBY (MODE 3) to provide the necessary overpressure protection. Therefore, this change in the method of testing the pressurizer safety valve lift settings does not involve a significant reduction in the margin of safety.

3.4B L18

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

The RCS specific activity limits are provided to assure the consequences of a steam generator tube rupture are acceptable. This change omits a requirement to immediately initiate corrective action when specific activity is outside the limits. However, the restoration is not required to be completed until 24 hours later. The immediate corrective actions are not considered the initiator of any previously analyzed accident. Further, an extension of the initiating time for a Required Action does not result in any hardware changes. The Completion Time is unchanged for restoration and does not significantly increase the probability of occurrence for initiation of any analyzed event since the Required Action does not change. Further, this change of the performance of Required Actions does not significantly increase the consequences of an accident because there is no change in the assumed response of the equipment in performing its specified mitigation functions, and no change in the response of the core parameters to assumed scenarios, from that considered using the immediate initiation requirements.

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 necessitate a physical alteration of the plant (no new or different type of equipment will be installed) or changes in parameters governing normal plant operation. The proposed change will still provide for prompt restoration of compliance with the limiting condition for operation, or appropriate compensatory actions are taken. 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?**

This change does not involve a significant reduction in a margin of safety since the specific activity of the RCS continues to be restored within the acceptable Completion Time. The increase in time allowed for initiating action permits appropriate evaluation of the situation, its causes, and the impact of the corrective action being considered while maintaining a limited time period during which the restoration of compliance must be achieved.

NO SIGNIFICANT HAZARDS CONSIDERATIONS STATEMENTS

3.4B L19

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

The upper limit on returnable RCS leakage via the reactor coolant pump seals is provided to assure the consequences of a loss of all AC power are acceptable. This change omits this upper limit from the ITS and eliminates a requirement to shutdown the reactor should this limit be exceeded. This change does not alter requirements or actions associated with Identified Leakage. The upper limit on returnable leakage and the shutdown actions are not considered the initiator of any previously analyzed accident. Further, these changes do not constitute hardware changes or modification of system operating parameters. Therefore, the deletion of these requirements does not significantly increase the probability of occurrence for initiation of any analyzed event. Further, this change does not alter the functional characteristics of any component nor the assumed initial conditions of any evaluated event. Continued remedial measures will be available and taken in accordance with the corrective action program. Any changes to the limit will be controlled under 10 CFR 50.59. Therefore, this change does not significantly increase the consequences of an accident because there is no change in the assumed response of the equipment in performing its specified function.

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 necessitate a physical alteration of the plant (no new or different type of equipment will be installed) or changes in parameters governing normal plant operation. The proposed change will still provide for compliance with the assumed initial conditions during normal operation, and appropriate compensatory actions will be available via the corrective action process. 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?**

This change does not involve a significant reduction in a margin of safety since the limits on returnable RCS leakage will continue to exist in a license basis document controlled under 10 CFR 50.59. The deletion of the shutdown action statement will allow the initiation of actions that permit appropriate evaluation of the situation, its cause(s), and the impact of the corrective action being considered.