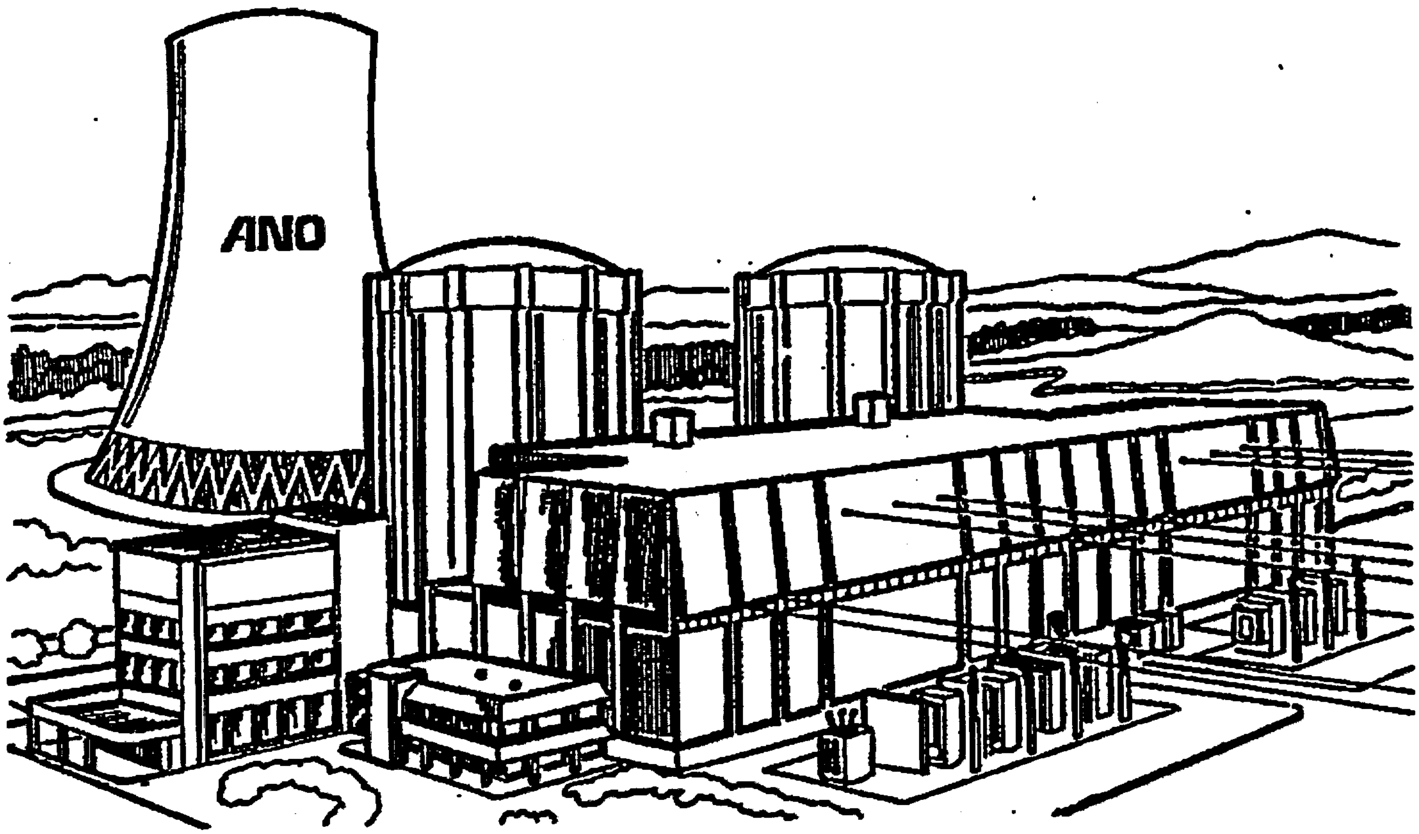
ARKANSAS NUCLEAR ONE - UNIT 1

IMPROVED TECHNICAL SPECIFICATIONS SUBMITTAL



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January 28, 2000

This Section Addresses the Following Specifications:

NUREG-1430	ANO-1 ITS	NUREG-1430 Title
3.2.1	3.2.1	Regulating Rod Insertion Limits
3.2.2	3.2.2	AXIAL POWER SHAPING ROD (APSR) Insertion Limits
3.2.3	3.2.3	AXIAL POWER IMBALANCE Operating Limits
3.2.4	3.2.4	QUADRANT POWER TILT (QPT)
3.2.5	3.2.5	Power Peaking Factors

3.2 POWER DISTRIBUTION LIMITS

3.2.1 Regulating Rod Insertion Limits

LCO 3.2.1 Regulating rod groups shall be within the physical insertion, sequence, and overlap limits specified in the COLR.

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

Not required for any regulating rod repositioned to perform SR 3.1.4.2.

APPLICABILITY: MODES 1 and 2.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Regulating rod groups inserted in restricted operation region.	A.1 -------- NOTE -------- Only required when THERMAL POWER is > 20% RTP. ------------------------ Perform SR 3.2.5.1.	Once per 2 hours
	<u>AND</u>	
	A.2 Restore regulating rod groups to within acceptable region.	24 hours from discovery of failure to meet the LCO
B. Required Action and associated Completion Time of Condition A not met.	B.1 Reduce THERMAL POWER to less than or equal to THERMAL POWER allowed by regulating rod group insertion limits.	2 hours
C. Regulating rod groups sequence or overlap requirements not met.	C.1 Restore regulating rod groups to within limits.	4 hours

CONDITION	REQUIRED ACTION	COMPLETION TIME
D. Regulating rod groups inserted in unacceptable operation region.	D.1 Initiate boration to restore SDM to within the limit provided in the COLR.	15 minutes
	AND	
	D.2.1 Restore regulating rod groups to within restricted operation region.	2 hours
	OR	
	D.2.2 Reduce THERMAL POWER to less than or equal to the THERMAL POWER allowed by the regulating rod group insertion limits.	2 hours
E. Required Actions and associated Completion Times of Conditions C or D not met.	E.1 Be in MODE 3.	6 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.2.1.1	Verify regulating rod groups are within the sequence and overlap limits as specified in the COLR.	12 hours
SR 3.2.1.2	Verify regulating rod groups meet the insertion limits as specified in the COLR.	12 hours

3.2 POWER DISTRIBUTION LIMITS

3.2.2 AXIAL POWER SHAPING ROD (APSR) Insertion Limits

LCO 3.2.2 APSRs shall be positioned within the limits specified in the COLR.

APPLICABILITY: MODES 1 and 2.

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. APSRs not within limits.	A.1	-----------NOTE------------ Only required when THERMAL POWER is > 20% RTP. -------------------------------	
		Perform SR 3.2.5.1.	Once per 2 hours
	AND		
	A.2	Restore APSRs to within limits.	24 hours
B. Required Action and associated Completion Time not met.	B.1	Be in MODE 3.	6 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.2.2.1	Verify APSRs are within acceptable limits specified in the COLR.	12 hours

3.2 POWER DISTRIBUTION LIMITS

3.2.3 AXIAL POWER IMBALANCE Operating Limits

LCO 3.2.3 AXIAL POWER IMBALANCE shall be maintained within the limits specified in the COLR.

APPLICABILITY: MODE 1 with THERMAL POWER > 40% RTP.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. AXIAL POWER IMBALANCE not within limits.	A.1 Perform SR 3.2.5.1. AND A.2 Reduce AXIAL POWER IMBALANCE to within limits.	Once per 2 hours 24 hours
B. Required Action and associated Completion Time not met.	B.1 Reduce THERMAL POWER to ≤ 40% RTP.	4 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.2.3.1	Verify AXIAL POWER IMBALANCE is within limits as specified in the COLR.	12 hours

3.2 POWER DISTRIBUTION LIMITS

3.2.4 QUADRANT POWER TILT (QPT)

LCO 3.2.4 QPT shall be maintained less than or equal to the steady state limits specified in the COLR.

APPLICABILITY: MODE 1 with THERMAL POWER > 20% RTP.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. QPT greater than the steady state limits specified in the COLR.	A.1.1 Perform SR 3.2.5.1. OR A.1.2.1 Reduce THERMAL POWER ≥ 2% RTP from the ALLOWABLE THERMAL POWER for each 1% of QPT greater than the steady state limit. AND A.1.2.2 Reduce nuclear overpower based on Reactor Coolant System flow and AXIAL POWER IMBALANCE trip setpoint ≥ 2% RTP from the ALLOWABLE THERMAL POWER for each 1% of QPT greater than the steady state limit. AND	Once per 2 hours 2 hours OR 2 hours after last performance of SR 3.2.5.1 10 hours OR 10 hours after last performance of SR 3.2.5.1 (continued)

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. (continued)	A.1.2.3	Reduce the regulating group insertion limits given in the COLR ≥ 2% RTP from the ALLOWABLE THERMAL POWER for each 1% of QPT greater than the steady state limit.	10 hours <u>OR</u> 10 hours after last performance of SR 3.2.5.1
		<u>AND</u>	
	A.1.2.4	Reduce the Operational Power Imbalance Setpoints given in the COLR ≥ 2% RTP from the ALLOWABLE THERMAL POWER for each 1% of QPT greater than the steady state limit.	10 hours <u>OR</u> 10 hours after last performance of SR 3.2.5.1
	<u>AND</u>		
	A.2	Restore QPT to less than or equal to the steady state limit.	24 hours from discovery of failure to meet the LCO
B. Required Action and associated Completion Time of Condition A not met.	B.1	Reduce THERMAL POWER to < 60% of the ALLOWABLE THERMAL POWER.	2 hours
	<u>AND</u>		
	B.2	Reduce nuclear overpower trip setpoint to ≤ 65.5% of the ALLOWABLE THERMAL POWER.	10 hours
C. Required Action and associated Completion Time for Condition B not met.	C.1	Reduce THERMAL POWER to ≤ 20% RTP.	4 hours
D. QPT greater than the maximum limit specified in the COLR.	D.1	Reduce THERMAL POWER to ≤ 20% RTP.	4 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.2.4.1	Verify QPT is within limits as specified in the COLR.	7 days AND When QPT has been restored to less than or equal to the steady state limit, 1 hour for 12 consecutive hours, or until verified acceptable at ≥ 95% RTP

3.2 POWER DISTRIBUTION LIMITS

3.2.5 Power Peaking

LCO 3.2.5 Linear Heat Rate (LHR) shall be within the limits specified in the COLR.

APPLICABILITY: MODE 1 with THERMAL POWER > 20% RTP.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. LHR not within limits.	A.1 Reduce THERMAL POWER to restore LHR to within limits.	2 hours
B. Required Action and associated Completion Time not met.	B.1 Reduce THERMAL POWER to ≤ 20% RTP.	4 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.2.5.1	-----NOTE----- Only required to be performed when specified in LCO 3.1.8, "PHYSICS TESTS Exceptions – MODE 1," or when complying with Required Actions of LCO 3.1.4, "CONTROL ROD Group Alignment Limits"; LCO 3.2.1, "Regulating Rod Insertion Limits"; LCO 3.2.2, "AXIAL POWER SHAPING ROD (APSR) Insertion Limits"; LCO 3.2.3, "AXIAL POWER IMBALANCE Operating Limits"; LCO 3.2.4, "QUADRANT POWER TILT (QPT)." ---------- Verify LHR is within limits by using the Incore Detector System to obtain a power distribution map.	As specified by the applicable LCO(s)

B 3.2 POWER DISTRIBUTION LIMITS

B 3.2.1 Regulating Rod Insertion Limits

BASES

BACKGROUND

The insertion limits of the regulating rods are initial condition assumptions used in all safety analyses that assume rod insertion upon reactor trip. The insertion limits directly affect the core power distributions, the worth of a potential ejected rod, the assumptions of SDM, and the initial reactivity insertion rate.

The applicable criteria for these reactivity and power distribution design requirements are described in SAR, Section 1.4, GDC 10, "Reactor Design," GDC 26, "Reactivity Control System Redundancy and Capability," GDC 28, "Reactivity Limits" (Ref. 1), and in 10 CFR 50.46, "Acceptance Criteria for Emergency Core Cooling Systems for Light Water Nuclear Power Reactors" (Ref. 2).

Limits on regulating rod insertion have been established, and all rod positions are monitored and controlled during power operation to ensure that the power distribution and reactivity limits defined by the design power peaking and SDM limits are not violated.

The regulating rod groups operate with a predetermined amount of position overlap, in order to approximate a linear relation between rod worth and rod position (integral rod worth). To achieve this approximately linear relationship, the regulating rod groups are withdrawn and operated in a predetermined sequence. The automatic control system controls reactivity by moving the regulating rod groups in sequence within analyzed ranges. The group sequence and overlap limits are specified in the COLR.

The regulating rods are used for precise reactivity control of the reactor. The positions of the regulating rods are normally controlled automatically by the automatic control system but can also be controlled manually. They are capable of rapid reactivity changes compared with borating or diluting the Reactor Coolant System (RCS).

The power density at any point in the core must be limited to maintain specified acceptable fuel design limits, including limits that ensure that the criteria specified in 10 CFR 50.46 (Ref. 2) are not violated. Together, LCO 3.2.1, "Regulating Rod Insertion Limits," LCO 3.2.2, "AXIAL POWER SHAPING ROD (APSR) Insertion Limits," LCO 3.2.3, "AXIAL POWER IMBALANCE Operating Limits," and LCO 3.2.4, "QUADRANT POWER TILT (QPT)," provide limits on control component operation and on monitored process variables to ensure that the core operates within the linear heat rate limits in the COLR. Operation within the linear heat rate limits given in the COLR prevents power peaks that would exceed the loss of coolant accident (LOCA) limits derived from the analysis of the Emergency Core Cooling Systems

(ECCS) and prevents departure from nucleate boiling (DNB) during a loss of forced reactor coolant flow accident. In addition to the linear heat rate limits, certain reactivity limits are met by regulating rod insertion limits. The regulating rod insertion limits also restrict the ejected CONTROL ROD worth to the values assumed in the safety analysis and support the minimum required SDM in MODES 1 and 2.

This LCO is required to minimize fuel cladding failures that breach the primary fission product barrier and release fission products into the reactor coolant in the event of a LOCA, loss of flow accident, ejected rod accident, or other postulated accidents requiring termination by a Reactor Protection System trip function.

APPLICABLE SAFETY ANALYSES

The fuel cladding must not sustain damage as a result of normal operation or abnormalities. The LCOs governing regulating rod insertion, APSR position, AXIAL POWER IMBALANCE, and QPT preclude core power distributions that violate the following fuel design criteria:

a. During a large break LOCA, the peak cladding temperature must not exceed 2200°F (Ref. 2).

b. During a loss of forced reactor coolant flow accident, there must be at least 95% probability at the 95% confidence level (the 95/95 DNB criterion) that the hot fuel rod in the core does not experience a DNB condition.

c. During an ejected rod accident, the fission energy input to the fuel must not exceed 280 cal/gm (Ref. 4).

d. The CONTROL RODS must be capable of shutting down the reactor with a minimum required SDM which assumes the highest worth CONTROL ROD stuck fully withdrawn.

Fuel cladding damage does not occur when the core is operated outside the conditions of these LCOs during normal operation. However, fuel cladding damage could result if an accident occurs with the simultaneous violation of one or more of the LCOs limiting the regulating rod position, the APSR position, the AXIAL POWER IMBALANCE, and the QPT. This potential for fuel cladding damage exists because changes in the power distribution can cause increased power peaking and correspondingly increased local linear heat rates (LHRs).

The SDM requirement is met by limiting the regulating and safety rod insertion limits such that sufficient inserted reactivity is available in the rods to shut down the reactor to hot zero power with a reactivity margin that assumes that the maximum worth rod remains fully withdrawn upon trip (Ref. 4). Operation at the SDM based regulating rod insertion limit may also indicate that the maximum ejected rod worth could be equal to the limiting value.

Operation at the regulating rod insertion limits may cause the local core power to approach the maximum linear heat generation rate or peaking factor with the allowed QPT present.

The regulating rod and safety rod insertion limits ensure that the safety analysis assumptions for SDM, ejected rod worth, and power distribution peaking factors remain valid (Refs. 3 and 4).

The regulating rod insertion limits LCO satisfies Criterion 2 of 10 CFR 50.36 (Ref. 5).

LCO

The limits on regulating rod group physical insertion, sequence, and overlap, as defined in the COLR, must be maintained because they ensure that the resulting power distribution is within the range of analyzed power distributions and that the SDM and ejected rod worth are maintained.

The overlap between regulating groups provides more uniform rates of reactivity insertion and withdrawal and is imposed to maintain acceptable power peaking during regulating rod motion.

Error adjusted maximum allowable setpoints for regulating rod insertion are provided in the COLR. The setpoints are derived by an adjustment of the measurement system independent limits to allow for THERMAL POWER level uncertainty and rod position errors.

LCO 3.2.1 has been modified by a Note that suspends the LCO requirement for those regulating rods not within the limits of the COLR solely due to testing in accordance with SR 3.1.4.2, which verifies the freedom of the rods to move. This SR may require the regulating rods to move below the LCO limit, out of group sequence, or beyond group overlap requirements, which would otherwise violate the LCO.

APPLICABILITY

The regulating rod physical insertion, sequence, and overlap limits shall be maintained with the reactor in MODES 1 and 2. These limits maintain the validity of the assumed power distribution, ejected rod worth, SDM, and reactivity rate insertion assumptions used in the safety analyses. Applicability in MODES 3, 4, and 5 is not required, because neither the power distribution nor ejected rod worth assumptions are exceeded in these MODES. SDM in MODES 3, 4, and 5 is governed by LCO 3.1.1, "SHUTDOWN MARGIN (SDM)."

ACTIONS

The regulating rod insertion setpoints provided in the COLR are based on the initial conditions assumed in the accident analyses and on the SDM. Specifically, separate insertion setpoints are specified to determine whether the unit is operating in violation of the initial conditions (e.g., the range of power distributions) assumed in the accident analyses or whether the unit is in violation of the SDM or ejected rod worth limits. Separate insertion setpoints are provided because different Required Actions and Completion Times apply, depending on which insertion setpoint has been violated. The area between the boundaries of the acceptable operation and unacceptable operation regions, illustrated on the regulating rod insertion setpoint figures in the COLR, is the restricted operation region. The actions required when operation occurs in the restricted operation region are described under Condition A. The actions required when operation occurs in the unacceptable operation region are described under Condition D. The actions required when operation occurs with the regulating rod group sequence or overlap requirements not met are described under Condition C.

A.1

Operation with the regulating rods in the restricted operation region shown on the regulating rod insertion setpoint figures specified in the COLR potentially violates the LOCA LHR limits, or the loss of flow accident DNB peaking limits.

For verification that LHRs are within their limits, SR 3.2.5.1 is performed using the Incore Detector System to obtain a three dimensional power distribution map. Verification that LHRs are within their limits ensures that operation with the regulating rods inserted into the restricted operation region does not violate the ECCS or DNB criteria. The required Completion Time of 2 hours is acceptable in that it allows the operator sufficient time for obtaining a power distribution map and for verifying the LHRs. Repeating SR 3.2.5.1 every 2 hours is acceptable because it ensures that continued verification of the LHRs is performed as core conditions (primarily regulating rod insertion and induced xenon redistribution) change.

Monitoring the LHRs does not provide verification that the reactivity insertion rate on the rod trip or the ejected rod worth limit is maintained, because worth is a reactivity parameter rather than a power peaking parameter. However, if the COLR figures do not show that a rod insertion setpoint is ejected rod worth limited, then the ejected rod worth is no more limiting than the SDM based rod insertion limit in the core design. Ejected rod worth limits are independently maintained by the Required Actions of Conditions A and D.

Required Action A.1 is modified by a Note that requires the performance of SR 3.2.5.1 only when THERMAL POWER is greater than 20% RTP. This establishes a Required Action that is consistent with the Applicability of LCO 3.2.5, "Power Peaking."

A.2

Indefinite operation with the regulating rods inserted in the restricted operation region is not prudent. Even if power peaking monitoring per Required Action A.1 is continued, reactivity limits may not be met and the abnormal regulating rod insertion may cause an adverse xenon redistribution, may cause the limits on AXIAL POWER IMBALANCE to be exceeded, or may adversely affect the long term fuel depletion pattern. Therefore, restoration of regulating rod groups to within their limits is required within 24 hours after discovery of failure to meet the requirements of this LCO. This required Completion Time is reasonable based on the low probability of an event occurring simultaneously with the limit out of specification in this relatively short time period. In addition, it precludes long term depletion with abnormal group insertions, thereby limiting the potential for an adverse xenon redistribution.

B.1

If the regulating rods cannot be positioned within the acceptable operation region shown on the figures in the COLR within the required Completion Time (i.e., Required Action A.2 not met), then the setpoints can be restored by reducing the THERMAL POWER to a value allowed by the regulating rod insertion setpoints in the COLR. The required Completion Time of 2 hours is sufficient to allow the operator to complete the power reduction in an orderly manner and without challenging the unit systems. Operation for up to 2 hours more in the restricted operation region shown in the COLR is acceptable, based on the low probability of an event occurring simultaneously with the regulating rod position out of specification in this relatively short time period.

C.1

Operation with the regulating rod groups out of sequence or with the group overlap limits exceeded may represent a condition beyond the assumptions used in the safety analyses. The design calculations assume no deviation in nominal overlap between regulating rod groups. However, small deviations in group overlap, as allowed by the COLR, may occur and would not cause significant differences in core reactivity, in power distribution, or rod worth, relative to the design calculations. Group sequence must be maintained because design calculations assume the regulating rods withdraw and insert in a predetermined order. The Completion Time of 4 hours is intended to restrict operation in this condition because of the potential severity associated with gross violations of group sequence or overlap requirements. The 4 hour Completion Time is based on operating experience which supports the restoration time without unnecessarily challenging unit operation and the low probability of an event occurring simultaneously with the limit out of specification.

D.1

Operation in the unacceptable operation region shown on the figures in the COLR corresponds to power operation with an SDM less than the minimum required value or with the ejected rod worth greater than the allowable value. The regulating rods may be inserted too far to provide sufficient negative reactivity insertion following a reactor trip and the ejected rod worth may exceed its initial condition limit. Therefore, the RCS boron concentration must be increased to restore the regulating rod insertion to a value that preserves the SDM and ejected rod worth limits. The required Completion Time of 15 minutes to initiate boration is reasonable, based on limiting the potential xenon redistribution, the low probability of an accident occurring in this relatively short time period, and the number of steps required to complete this Action. This period allows the operator sufficient time for aligning the required valves and for starting the boric acid pumps. Boration continues until the regulating rod group positions are restored to at least within the restricted operation region, which restores the minimum SDM and reduces the potential ejected rod worth to within its limit.

D.2.1

The required Completion Time of 2 hours from initial discovery of a regulating rod group in the unacceptable operation region until its restoration to within the restricted operation region shown on the figures in the COLR allows sufficient time for borated water to enter the RCS from the chemical addition and makeup and purification systems, thereby allowing the regulating rods to be withdrawn to the restricted operation region. Operation in the restricted operation region for up to 2 hours is reasonable, based on limiting the potential for an adverse xenon redistribution, the low probability of an accident occurring in this relatively short time period, and the number of steps required to complete this Action.

D.2.2

The SDM and ejected rod worth limit can also be restored by reducing the THERMAL POWER to a value allowed by the regulating rod insertion setpoints in the COLR. The required Completion Time of 2 hours is sufficient to allow the operator to complete the power reduction in an orderly manner and without challenging the unit systems. Operation for up to 2 hours in the restricted operation region shown in the COLR is acceptable, based on the low probability of an event occurring simultaneously with the limit out of specification in this relatively short time period. In addition, it precludes long term depletion with abnormal group insertions or configurations and limits the potential for an adverse xenon redistribution.

E.1

If the Required Actions and associated Completion Times of Conditions C or D are not met, then the reactor is placed in MODE 3, in which this LCO does not apply. This Action ensures that the reactor does not continue operating in violation of the peaking limits, the ejected rod worth, the reactivity insertion rate assumed as initial conditions in the accident analyses, or the required minimum SDM assumed in the accident analyses. The required Completion Time of 6 hours is reasonable, based on operating experience regarding the amount of time required to reach MODE 3 from RTP without challenging unit systems.

SURVEILLANCE REQUIREMENTS

SR 3.2.1.1

This Surveillance ensures that the sequence and overlap limits are not violated. A Surveillance Frequency of 12 hours is acceptable because little rod motion occurs during this period due to fuel burnup. Also, the Frequency takes into account other information available in the control room for monitoring the status of the regulating rods.

SR 3.2.1.2

Verification of the regulating rod insertion setpoints as specified in the COLR at a Frequency of 12 hours is sufficient to detect regulating rod banks that may be approaching the group insertion setpoints, because little rod motion due to fuel burnup occurs in 12 hours. Also, the Frequency takes into account other information available in the control room for monitoring the status of the regulating rods.

REFERENCES

1. SAR, Section 1.4, GDC 10, GDC 26 and GDC 28.
2. 10 CFR 50.46.
3. SAR, Chapter 3.
4. SAR, Chapter 14.
5. 10 CFR 50.36.

B 3.2 POWER DISTRIBUTION LIMITS

B 3.2.2 AXIAL POWER SHAPING ROD (APSR) Insertion Limits

BASES

BACKGROUND

The insertion limits of the APSRs are initial condition assumptions in all safety analyses that are affected by core power distributions. The applicable criterion for these power distribution design requirements are SAR Section 1.4, GDC 10, "Reactor Design" (Ref. 1), and 10 CFR 50.46, "Acceptance Criteria for Emergency Core Cooling Systems for Light Water Nuclear Power Reactors" (Ref. 2).

Limits on APSR insertion have been established, and all APSR positions are monitored and controlled during power operation to ensure that the power distribution defined by the design power peaking limits is maintained.

The power density at any point in the core must be limited to maintain specified acceptable fuel design limits, including limits that meet the criteria specified in Reference 2. Together, LCO 3.2.1, "Regulating Rod Insertion Limits," LCO 3.2.2, "AXIAL POWER SHAPING ROD (APSR) Insertion Limits," LCO 3.2.3, "AXIAL POWER IMBALANCE Operating Limits," and LCO 3.2.4, "QUADRANT POWER TILT (QPT)," provide limits on control component operation and on monitored process variables to ensure that the core operates within the linear heat rate (LHR) limits in the COLR. Operation within the LHR limits given in the COLR prevents power peaks that exceed the loss of coolant accident (LOCA) limits derived from the analysis of the Emergency Core Cooling Systems (ECCS) and prevents departure from nucleate boiling (DNB) during a loss of forced reactor coolant flow accident. The APSRs do not insert upon a reactor trip.

This LCO is required to minimize fuel cladding failures that would breach the primary fission product barrier and release fission products to the reactor coolant in the event of a LOCA, loss of flow accident, ejected rod accident, or other postulated accident requiring termination by a Reactor Protection System trip function.

APPLICABLE SAFETY ANALYSES

The fuel cladding must not sustain damage as a result of normal operation or abnormalities. Acceptance criteria for the safety and regulating rod insertion, APSR position, AXIAL POWER IMBALANCE, and QPT LCOs preclude core power distributions that violate the following fuel design criteria:

a. During a large break LOCA, the peak cladding temperature must not exceed 2200°F (Ref. 2);

b. During a loss of forced reactor coolant flow accident, there must be at least 95% probability at the 95% confidence level (the 95/95 DNB criterion) that the hot fuel rod in the core does not experience a DNB condition;

c. During an ejected rod accident, the fission energy input to the fuel must not exceed 280 cal/gm (Ref. 3); and

d. CONTROL RODS must be capable of shutting down the reactor with a minimum required SDM which assumes the highest worth CONTROL ROD stuck fully withdrawn (GDC 26, Ref. 1).

Fuel cladding damage does not occur when the core is operated outside these LCOs during normal operation. However, fuel cladding damage could result should an accident occur simultaneously with violation of one or more of these LCOs. This potential for fuel cladding damage exists because changes in the power distribution can cause increased power peaking and corresponding increased local linear heat rates.

Operation at the APSR insertion limits may approach the maximum allowable linear heat generation rate with the allowed QPT present.

In MODES 1 and 2 while critical, the APSR insertion limits satisfy Criterion 2 of 10 CFR 50.36 (Ref. 4). In MODE 2 while subcritical, the APSR insertion limits satisfy Criterion 4 of 10 CFR 50.36.

LCO

The setpoints on APSR physical insertion as defined in the COLR must be maintained because they serve the function of controlling the power distribution within an acceptable range.

The fuel cycle design assumes APSR withdrawal at the EFPD burnup window specified in the COLR. Prior to this window, the APSRs are maintained in accordance with operating guidelines provided by reactor engineering during steady state operation. After this window, the APSRs are not allowed to be reinserted for the remainder of the fuel cycle.

APPLICABILITY

The APSR physical insertion limits shall be maintained with the reactor in MODES 1 and 2. These limits maintain the power distribution within the range assumed in the accident analyses. In MODES 1 and 2, the limits on APSR insertion specified by this LCO maintain the axial fuel burnup design conditions assumed in the reload safety evaluation analysis. Applicability in MODES 3, 4, and 5 is not required, because the reactor is subcritical.

ACTIONS

For steady state power operation, a normal position for APSR insertion is specified in the station operating procedures. The APSRs may be positioned as necessary for transient AXIAL POWER IMBALANCE control until the fuel cycle design requires them to be fully withdrawn. (Not all fuel cycles may incorporate APSR withdrawal.) APSR position limits are not imposed for gray APSRs, with two exceptions. If the fuel cycle design incorporates an APSR withdrawal (usually near end of cycle (EOC)), the APSRs may not be maintained in the fully withdrawn position prior to the fuel cycle burnup for the APSR withdrawal. If this occurs, the APSRs must be restored to their normal inserted position. Conversely, after the fuel cycle burnup for the APSR withdrawal occurs, the APSRs may not be reinserted for the remainder of the fuel cycle. These restrictions apply to ensure the axial burnup distribution that accumulates in the fuel will be consistent with the expected (as designed) distribution.

A.1

For verification that the core linear heat rates are within their limits, SR 3.2.5.1 is performed using the Incore Detector System to obtain a three dimensional power distribution map. Successful verification that the LHRs are within their limits ensures that operation with the APSRs inserted or withdrawn in violation of the setpoints specified in the COLR do not violate either the ECCS or DNB criteria. The required Completion Time of 2 hours is reasonable to allow the operator to obtain a power distribution map and to verify the LHRs. Repeating SR 3.2.5.1 every 2 hours is reasonable to ensure that continued verification of the LHRs is obtained as core conditions (primarily the regulating rod insertion and induced xenon redistribution) change.

Required Action A.1 is modified by a Note that requires the performance of SR 3.2.5.1 only when THERMAL POWER is greater than 20% RTP. This establishes a Required Action that is consistent with the Applicability of LCO 3.2.5, "Power Peaking."

A.2

Indefinite operation with the APSRs positioned in violation of the setpoints specified in the COLR is not prudent. Even if LHR monitoring per Required Action A.1 is continued, the abnormal APSR positioning may cause an adverse xenon redistribution, may cause the limits on AXIAL POWER IMBALANCE to be exceeded, or may affect the long term fuel depletion pattern. Therefore, operation is allowed for up to 24 hours. This required Completion Time is reasonable based on the low probability of an event occurring simultaneously with the APSR position out of specification. In addition, it precludes long term depletion with the APSRs in positions that have not been analyzed, thereby limiting the potential for an adverse xenon redistribution. This time limit also ensures that the intended burnup distribution is maintained, and allows the operator sufficient time to reposition the APSRs to correct their positions.

Because the APSRs are not operated by the automatic control system, manual action by the operator is required to restore the APSRs to the positions specified in the COLR.

B.1

If the Required Action and associated Completion Time are not met, the reactor must be placed in MODE 3, in which this LCO does not apply. This action ensures that the fuel does not continue to be depleted in an unintended burnup distribution. The required Completion Time of 6 hours is reasonable, based on operating experience regarding the time required to reach MODE 3 from RTP in an orderly manner and without challenging unit systems.

SURVEILLANCE REQUIREMENTS

SR 3.2.2.1

Fuel cycle designs that allow APSR withdrawal near end of cycle (EOC) do not permit reinsertion of APSRs after the time of withdrawal. Verification that the APSRs are within their insertion setpoints at a 12 hour Frequency is sufficient to ensure that the APSR insertion setpoints are preserved. The 12 hour Frequency required for performing this verification is sufficient because APSRs are positioned by manual control and are normally moved infrequently. The Frequency takes into account other information available in the control room for monitoring the axial power distribution in the reactor core.

REFERENCES

1. SAR Section 1.4, GDC 10 and GDC 26.
2. 10 CFR 50.46.
3. SAR, Chapter 14.
4. 10 CFR 50.36.

B 3.2 POWER DISTRIBUTION LIMITS

B 3.2.3 AXIAL POWER IMBALANCE Operating Limits

BASES

BACKGROUND

This LCO is required to limit the core power distribution based on accident initial condition criteria.

The power density at any point in the core must be limited to maintain specified acceptable fuel design limits, including limits that satisfy the criteria specified in 10 CFR 50.46 (Ref. 1). This LCO provides limits on AXIAL POWER IMBALANCE to ensure that the core operates within the linear heat rate (LHR) limits given in the COLR. Operation within the LHR limits given in the COLR prevents power peaks that exceed the loss of coolant accident (LOCA) limits derived from the analysis of the Emergency Core Cooling Systems (ECCS) and prevents departure from nucleate boiling (DNB) during a loss of forced reactor coolant flow accident.

This LCO is required to limit fuel cladding failures that breach the primary fission product barrier and release fission products into the reactor coolant in the event of a LOCA, loss of forced reactor coolant flow accident, or other postulated accident requiring termination by a Reactor Protection System trip function. This LCO limits the amount of damage to the fuel cladding during an accident by maintaining the validity of the assumptions in the safety analyses related to the initial power distribution and reactivity.

Fuel cladding failure during a postulated LOCA is limited by restricting the maximum LHR so that the peak cladding temperature does not exceed 2200°F (Ref. 1). Peak cladding temperatures > 2200°F cause severe cladding failure by oxidation due to a Zircaloy water reaction. Other criteria must also be met (e.g., maximum cladding oxidation, maximum hydrogen generation, coolable geometry, and long term cooling). However, peak cladding temperature is usually most limiting.

Proximity to the DNB condition is expressed by the departure from nucleate boiling ratio (DNBR), defined as the ratio of the cladding surface heat flux required to cause DNB to the actual cladding surface heat flux. The minimum DNBR value during both normal operation and anticipated transients is limited to the DNBR correlation limit for the particular fuel design in use and is accepted as an appropriate margin to DNB. The DNB correlation limit ensures that there is at least 95% probability at the 95% confidence level (the 95/95 DNB criterion) that the hot fuel rod in the core does not experience DNB.

The measurement system independent limits on AXIAL POWER IMBALANCE are determined analytically by the reload safety evaluation analysis without adjustment for measurement system error and uncertainty. Operation beyond these limits could invalidate the assumptions used in the accident analyses regarding the core power distribution. The AXIAL POWER IMBALANCE setpoints provided in the COLR account for measurement system error and uncertainty.

APPLICABLE SAFETY ANALYSES

The fuel cladding must not sustain damage as a result of normal operation and abnormalities. The LCOs based on power distribution, LCO 3.2.1, "Regulating Rod Insertion Limits," LCO 3.2.2, "AXIAL POWER SHAPING ROD (APSR) Insertion Limits," LCO 3.2.3, "AXIAL POWER IMBALANCE Operating Limits," and LCO 3.2.4, "QUADRANT POWER TILT (QPT)," preclude core power distributions that would violate the following fuel design criteria:

a. During a large break LOCA, peak cladding temperature must not exceed 2200°F (Ref. 1);

b. During a loss of forced reactor coolant flow accident, there must be at least a 95% probability at the 95% confidence level (the 95/95 DNB criterion) that the hot fuel rod in the core does not experience a DNB condition.

The regulating rod positions, the APSR positions, the AXIAL POWER IMBALANCE, and the QPT are process variables that characterize and control the three dimensional power distribution of the reactor core.

Fuel cladding damage does not occur when the core is operated outside this LCO during normal operation. However, fuel cladding damage could result should an accident occur with simultaneous violation of one or more of the LCOs governing the four process variables cited above. This potential for fuel cladding damage exists because changes in the power distribution can cause increased power peaking and corresponding increased local LHRs.

The regulating rod insertion, the APSR positions, the AXIAL POWER IMBALANCE, and the QPT are monitored and controlled during power operation to ensure that the power distribution is within the bounds set by the safety analyses. The axial power distribution is maintained primarily by the AXIAL POWER IMBALANCE and the APSR position limits; and the radial power distribution is maintained primarily by the QPT limits. The regulating rod insertion limits affect both the radial and axial power distributions.

The dependence of the core power distribution on burnup, regulating rod insertion, APSR position, and spatial xenon distribution is taken into account when the reload safety evaluation analysis is performed.

Operation at the AXIAL POWER IMBALANCE limit must be interpreted as operating the core at the maximum allowable LHR assumed as initial conditions for the accident analyses with the allowed QPT present.

AXIAL POWER IMBALANCE satisfies Criterion 2 of 10 CFR 50.36 (Ref. 2).

LCO

The power distribution LCO limits have been established based on correlations between power peaking and easily measured process variables: regulating rod position, APSR position, AXIAL POWER IMBALANCE, and QPT. The AXIAL POWER IMBALANCE envelope contained in the COLR represents the setpoints beyond which the core power distribution could either exceed the LOCA LHR limits or cause a reduction in the DNBR below the Safety Limit during the loss of flow accident with the allowable QPT present and with the APSR positions consistent with the limitations on APSR withdrawal determined by the fuel cycle design and specified by LCO 3.2.2.

The AXIAL POWER IMBALANCE maximum allowable setpoints (measurement system dependent limits) applicable for the full Incore Detector System, the Minimum Incore Detector System, and the Excore Detector System are provided in the COLR.

APPLICABILITY

In MODE 1, the limits on AXIAL POWER IMBALANCE must be maintained when THERMAL POWER is > 40% RTP to prevent the core power distribution from exceeding the LOCA and loss of flow assumptions used in the accident analyses. Applicability of these limits at ≤ 40% RTP in MODE 1 is not required. This operation is acceptable based on engineering judgment because the combination of AXIAL POWER IMBALANCE with the maximum allowable THERMAL POWER level will not result in LHRs sufficiently large to violate the fuel design limits. In MODES 2, 3, 4, 5, and 6, this LCO is not applicable because the reactor is not generating sufficient THERMAL POWER to produce fuel damage.

ACTIONS

A.1

The AXIAL POWER IMBALANCE operating setpoints that maintain the validity of the assumptions regarding the power distributions in the accident analyses of the LOCA and the loss of flow accident are provided in the COLR. Operation within the AXIAL POWER IMBALANCE setpoints given in the COLR is the acceptable region of operation. Operation in violation of the AXIAL POWER IMBALANCE setpoints given in the COLR is the restricted region of operation.

Operation with AXIAL POWER IMBALANCE in the restricted region shown on the AXIAL POWER IMBALANCE figures in the COLR potentially violates the LOCA LHR limits or the loss of flow accident DNB peaking limits or both. For verification that core local LHRs are within their specified limits, SR 3.2.5.1 is performed using the Incore Detector System to obtain a three dimensional power distribution map. Verification that core local LHRs are within their specified limits ensures that operation with the AXIAL POWER IMBALANCE in the restricted region does not violate the ECCS or 95/95 DNB criteria. The required Completion Time of 2 hours provides reasonable time for the operator to obtain a power distribution map and to determine and verify that the core local LHRs are within their specified limits. The 2 hour Frequency provides reasonable time to ensure that continued verification of the core local LHRs is obtained as core conditions (primarily regulating rod insertion and induced xenon redistribution) change, because little rod motion occurs in 2 hours due to fuel burnup, the potential for xenon redistribution is limited, and the probability of an event occurring in this short time frame is low.

A.2

Indefinite operation with the AXIAL POWER IMBALANCE in the restricted region is not prudent. Even if LHR monitoring per Required Action A.1 is continued, excessive AXIAL POWER IMBALANCE over an extended period of time may cause a potentially adverse xenon redistribution to occur. Therefore, LHR monitoring is only allowed for a maximum of 24 hours. This required Completion Time is reasonable based on the low probability of a limiting event occurring simultaneously with the AXIAL POWER IMBALANCE outside the setpoints of this LCO. In addition, this limited Completion Time precludes long term depletion of the reactor fuel with excessive AXIAL POWER IMBALANCE and gives the operator sufficient time to reposition the APSRs or regulating rods to reduce the AXIAL POWER IMBALANCE because adverse effects of xenon redistribution and fuel depletion are limited.

B.1

If the Required Actions and the associated Completion Times of Condition A are not met, the AXIAL POWER IMBALANCE may exceed its specified limits and the reactor may be operating with a global axial power distribution mismatch. Continued operation in this configuration may induce an axial xenon oscillation and may result in an increased linear heat generation rate when the xenon redistributes. Reducing THERMAL POWER to ≤ 40% RTP reduces the maximum LHR to a value that does not exceed the LHR initial condition limits assumed in the accident analyses. The required Completion Time of 4 hours is reasonable based on limiting a potentially adverse xenon redistribution, the low probability of an accident occurring in this relatively short time period, and the number of steps required to complete this Action.

SURVEILLANCE REQUIREMENTS

SR 3.2.3.1

The AXIAL POWER IMBALANCE can be monitored by both the Incore and Excore Detector Systems. The AXIAL POWER IMBALANCE maximum allowable setpoints are derived from their corresponding measurement system independent limits by adjusting for both the system observability errors and instrumentation errors. Although they may be based on the same measurement system independent limits, the setpoints for the different systems are not identical because of differences in the errors applicable for each of these systems. The uncertainty analysis that defines the required error adjustment to convert the measurement system independent limits to full incore detector system limits assumes that 75% of the detectors in each quadrant are OPERABLE. Detectors located on the core major axes are assumed to contribute one half of their output to each quadrant; detectors in the center assembly are assumed to contribute one quarter of their output to each quadrant. For AXIAL POWER IMBALANCE measurements using the Incore Detector System, the Minimum Incore Detector System consists of OPERABLE detectors configured as follows:

a. Nine detectors shall be arranged such that there are three detectors in each of three strings and there are three detectors lying in the same axial plane, with one plane at the core midplane and one plane in each axial core half;

b. The axial planes in each core half shall be symmetrical about the core midplane; and

c. The detector strings shall not have radial symmetry.

Figure B 3.2.3-1 (Minimum Incore Detector System for AXIAL POWER IMBALANCE Measurement) depicts an example of this configuration. This arrangement is chosen to reduce the uncertainty in the measurement of the AXIAL POWER IMBALANCE by the Minimum Incore Detector System. For example, the requirement for placing one detector of each of the three strings at the core midplane puts three detectors in the central region of the core where the neutron flux tends to be higher. It also helps prevent measuring an AXIAL POWER IMBALANCE that is excessively large when the reactor is operating at low THERMAL POWER levels. The third requirement for placement of detectors (i.e., radial asymmetry) reduces uncertainty by measuring the neutron flux at core locations that are not radially symmetric.

Verification of the AXIAL POWER IMBALANCE indication every 12 hours ensures that the AXIAL POWER IMBALANCE setpoints are not violated and takes into account other information and alarms available in the control room. This Surveillance Frequency is acceptable because the mechanisms that can cause AXIAL POWER IMBALANCE, such as xenon redistribution or control rod drive mechanism malfunctions that cause slow AXIAL POWER IMBALANCE increases, can be discovered by the operator before the specified limits are violated.

REFERENCES

1. 10 CFR 50.46.
2. 10 CFR 50.36.

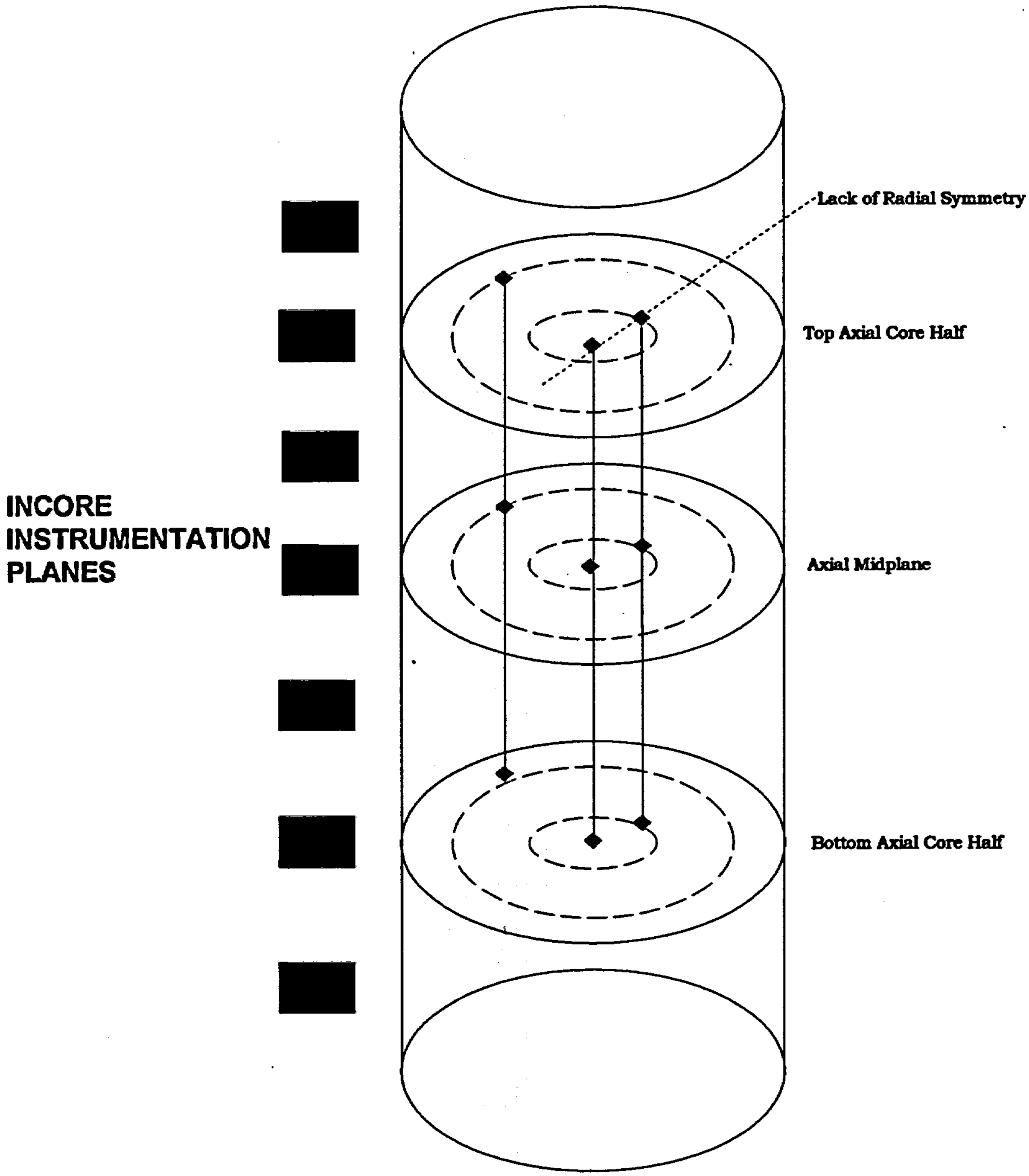


Figure B 3.2.3-1 (page 1 of 1)
Minimum Incore System for AXIAL POWER IMBALANCE Measurement

B 3.2 POWER DISTRIBUTION LIMITS

B 3.2.4 QUADRANT POWER TILT (QPT)

BASES

BACKGROUND

This LCO is required to limit the core power distribution based on accident initial condition criteria.

The power density at any point in the core must be limited to maintain specified acceptable fuel design limits, including limits that preserve the criteria specified in 10 CFR 50.46 (Ref. 1). Together, LCO 3.2.1, "Regulating Rod Insertion Limits," LCO 3.2.2, "AXIAL POWER SHAPING ROD (APSR) Insertion Limits," LCO 3.2.3, "AXIAL POWER IMBALANCE Operating Limits," and LCO 3.2.4, "QUADRANT POWER TILT (QPT)," provide limits on control component operation and on monitored process variables to ensure that the core operates within the linear heat rate (LHR) limits given in the COLR. Operation within the LHR limits given in the COLR prevents power peaks that exceed the loss of coolant accident (LOCA) limits derived by Emergency Core Cooling Systems (ECCS) analysis and prevents departure from nucleate boiling (DNB) during a loss of forced reactor coolant flow accident.

This LCO is required to limit fuel cladding failures that breach the primary fission product barrier and release fission products to the reactor coolant in the event of a LOCA, loss of forced reactor coolant flow, or other accident requiring termination by a Reactor Protection System trip function. This LCO limits the amount of damage to the fuel cladding during an accident by maintaining the validity of the assumptions used in the safety analysis related to the initial power distribution and reactivity.

Fuel cladding failure during a postulated LOCA is limited by restricting the maximum LHR so that the peak cladding temperature does not exceed 2200°F (Ref. 1). Peak cladding temperatures > 2200°F cause severe cladding failure by oxidation due to a Zircaloy water reaction. Other criteria must also be met (e.g., maximum cladding oxidation, maximum hydrogen generation, coolable geometry, and long term cooling). However, peak cladding temperature is usually most limiting.

Proximity to the DNB condition is expressed by the departure from nucleate boiling ratio (DNBR), defined as the ratio of the cladding surface heat flux required to cause DNB to the actual cladding surface heat flux. The minimum DNBR value during both normal operation and anticipated transients is limited to the DNBR correlation limit for the particular fuel design in use, and is accepted as an appropriate margin to DNB. The DNBR correlation limit ensures that there is at least 95% probability at the 95% confidence level (the 95/95 DNB criterion) that the hot fuel rod in the core does not experience DNB.

The measurement system independent limits on QPT are determined analytically by the reload safety evaluation analysis without adjustment for measurement system error and uncertainty. Operation beyond these limits could invalidate core power distribution assumptions used in the accident analysis. The error adjusted maximum allowable setpoints (measurement system dependent limits) for QPT are specified in the COLR.

APPLICABLE SAFETY ANALYSES

The fuel cladding must not sustain damage as a result of normal operation and abnormalities. The LCOs based on power distribution (LCO 3.2.1, LCO 3.2.2, LCO 3.2.3, and LCO 3.2.4) preclude core power distributions that violate the following fuel design criteria:

a. During a large break LOCA, the peak cladding temperature must not exceed 2200°F (Ref. 1).

b. During a loss of forced reactor coolant flow accident, there must be at least 95% probability at the 95% confidence level (the 95/95 DNB criterion) that the hot fuel rod in the core does not experience a DNB condition.

QPT is one of the process variables that characterize and control the three dimensional power distribution of the reactor core.

Fuel cladding damage does not occur when the core is operated outside this LCO during normal operation. However, fuel cladding damage could result if an accident occurs with simultaneous violation of one or more of the LCOs governing the core power distribution. Changes in the power distribution can cause increased power peaking and correspondingly increased local LHRs.

The dependence of the core power distribution on burnup, regulating rod insertion, APSR position, and spatial xenon distribution is taken into account during the reload safety evaluation analysis. An allowance for QPT is accommodated in the analysis and resultant LCO limits. The increase in peaking taken for QPT is developed from a database of full core power distribution calculations (Ref. 2). The calculations consist of simulations of many power distributions with tilt causing mechanisms (e.g., dropped or misaligned CONTROL RODS, broken APSR fingers fully inserted, misloaded assemblies, and burnup gradients). An increase of < 2% peak power per 1% QPT is supported by the analysis, therefore a value of 2% peak power increase per 1% QPT is used to bound peak power increases due to QPT.

Operation at the AXIAL POWER IMBALANCE or rod insertion limits must be interpreted as operating the core at the maximum allowable LHR for accident initial conditions with the allowed QPT present.

QPT satisfies Criterion 2 of 10 CFR 50.36 (Ref. 3).

LCO

The power distribution LCO limits have been established based on correlations between power peaking and easily measured process variables: regulating rod position, APSR position, AXIAL POWER IMBALANCE, and QPT. The regulating rod insertion setpoints and the AXIAL POWER IMBALANCE boundaries contained in the COLR represent the measurement system dependent limits at which the core power distribution could either exceed the LOCA LHR limits or cause a reduction in DNBR below the safety limit during a loss of flow accident with the allowable QPT present and with an APSR position consistent with the limitations on APSR position determined by the fuel cycle design and specified by LCO 3.2.2.

The allowable setpoints for steady state and maximum setpoints for QPT applicable for the full symmetrical Incore Detector System, Minimum Incore Detector System, and Excore Detector System are provided in the COLR. The setpoints for the three systems are derived by adjustment of the measurement system independent QPT limits also given in the COLR to allow for system observability and instrumentation errors.

APPLICABILITY

In MODE 1, the limits on QPT must be maintained when THERMAL POWER is > 20% RTP to prevent the core power distribution from exceeding the design limits. The minimum power level of 20% RTP is large enough to obtain meaningful QPT indications without compromising safety.

In MODE 2, the combination of QPT with maximum ALLOWABLE THERMAL POWER level does not result in LHRs sufficiently large to violate the fuel design limits, and therefore, applicability in this MODE is not required. Although not specifically addressed in the LCO, QPTs greater than the maximum setpoint specified in the COLR in MODE 1 with THERMAL POWER < 20% RTP are allowed based on engineering judgement.

In MODES 3, 4, 5, and 6, this LCO is not applicable, because the reactor is not generating significant THERMAL POWER and QPT is indeterminate.

ACTIONS

A.1.1

The steady state setpoint specified in the COLR provides an allowance for QPT that may occur during normal operation. A peaking increase to accommodate QPTs up to the steady state setpoint is allowed by the regulating rod insertion limits of LCO 3.2.1 and the AXIAL POWER IMBALANCE limits of LCO 3.2.3.

Operation with QPT greater than the steady state setpoint specified in the COLR potentially violates the LOCA LHR limits, or loss of flow accident DNB peaking limits, or both. For verification that core local LHRs are within their specified limits, SR 3.2.5.1 is performed using the Incore Detector System to obtain a three dimensional power distribution map. Verification that core local LHRs are within their limits ensures that operation with QPT greater than the steady state setpoint does not violate the ECCS or 95/95 DNB criteria. The required Completion Time of once per 2 hours is a reasonable amount of time to allow the operator to obtain a power distribution map and to verify the core local LHRs. Repeating SR 3.2.5.1 every 2 hours is a reasonable Frequency at which to ensure that continued verification of the core local LHRs is obtained as core conditions that influence QPT change.

A.1.2.1

The safety analysis has shown that a conservative corrective action is to reduce THERMAL POWER by 2% RTP or more from the ALLOWABLE THERMAL POWER for each 1% of QPT in excess of the steady state setpoint. This action limits the local LHR to a value corresponding to the assumed accident initial condition limits. The required Completion Time of 2 hours is reasonable, based on limiting the potential for xenon redistribution, the low probability of an accident occurring, and the steps required to complete the Required Action.

If QPT can be reduced to less than or equal to the steady state setpoint in < 2 hours, the reactor may return to normal operation without undergoing a power reduction. Significant radial xenon redistribution does not occur within this amount of time.

The required Completion Time of 2 hours after the last performance of SR 3.2.5.1 allows reduction of THERMAL POWER in the event the operators cannot or choose not to continue to perform SR 3.2.5.1 as required by Required Action A.1.1.

A.1.2.2

Power operation is allowed to continue if THERMAL POWER is reduced in accordance with Required Action A.1.2.1. The same reduction (i.e., 2% RTP or more) is also applicable to the nuclear overpower based on Reactor Coolant System (RCS) flow and AXIAL POWER IMBALANCE trip setpoint, for each 1% of QPT in excess of the steady state limit. This reduction maintains both core protection and thermal margins at the reduced THERMAL POWER level similar to that at RTP. The required Completion Time of 10 hours or 10 hours after the last performance of SR 3.2.5.1 is reasonable based on the need to limit the potentially adverse xenon redistribution, the low probability of an accident occurring while operating with the QPT limits not met, and the number of steps required to complete the Required Action.

A.1.2.3

Power operation is allowed to continue if restrictions are imposed on the allowed degree of regulating group insertion. This Required Action requires a reduction in the regulating group insertion setpoints given in the COLR by ≥ 2% RTP from the ALLOWABLE THERMAL POWER for each 1% of QPT greater than the steady state setpoint. Based on engineering judgment, this action is intended to reduce the potential power peaking associated with regulating rod group insertion into the core.

The Completion Time of 10 hours is reasonable based on the need to limit the potentially adverse xenon redistribution, the low probability of an accident occurring while operating with QPT limits not met, and the number of steps required to complete the Required Action. The second Completion Time of 10 hours after the last performance of SR 3.2.5.1 is based on the same reasoning and is provided in the event the operators cannot or choose not to continue to perform SR 3.2.5.1 as required by Required Action A.1.1.

A.1.2.4

Power operation is allowed to continue if restrictions are imposed on the allowed Operational Power Imbalance Setpoints given in the COLR. This Required Action results in a reduction in the allowed THERMAL POWER level as a function of AXIAL POWER IMBALANCE by ≥ 2% RTP from the ALLOWABLE THERMAL POWER for each 1% of QPT greater than the steady state limit. Based on engineering judgment, this action is intended to reduce the potential power peaking associated with the combined affects of operating with an AXIAL POWER IMBALANCE and a QPT.

The Completion Time of 10 hours is reasonable based on the need to limit the potentially adverse xenon redistribution, the low probability of an accident occurring while operating with QPT limits not met, and the number of steps required to complete the Required Action. The second Completion Time of 10 hours after the last performance of SR 3.2.5.1 is based on the same reasoning and is provided in the event the operators cannot or choose not to continue to perform SR 3.2.5.1 as required by Required Action A.1.1.

A.2

Although the actions directed by Required Action A.1.2.1 restore thermal margins, if the source of the QPT is not established and corrected, it is prudent to establish increased margins. A required Completion Time of 24 hours to reduce QPT to less than the steady state limit is a reasonable time for investigation and corrective measures.

B.1

If the Required Actions and associated Completion Times of Condition A are not met, a further power reduction is required. Power reduction to < 60% of ALLOWABLE THERMAL POWER provides conservative protection from increased peaking due to xenon redistribution. The required Completion Time of 2 hours is reasonable to allow the operator to reduce THERMAL POWER to < 60% of ALLOWABLE THERMAL POWER without challenging unit systems.

B.2

Reduction of the nuclear overpower trip setpoint to ≤ 65.5% of ALLOWABLE THERMAL POWER after THERMAL POWER has been reduced to < 60% of ALLOWABLE THERMAL POWER maintains both core protection and OPERABILITY margin at reduced power similar to that at full power. The required Completion Time of 10 hours allows the operator sufficient time to reset the trip setpoint and is reasonable based on operating experience.

C.1

If the Required Actions and associated Completion Times of Condition B are not met, then the reactor will continue in power operation with significant QPT. Either the power level has not been reduced to comply with the Required Action or the nuclear overpower trip setpoint has not been reduced within the required Completion Time. To preclude risk of fuel damage in any of these conditions, THERMAL POWER is reduced further. Operation below 20% RTP allows the operator to investigate the cause of the QPT and to correct it. Local LHRs with a large QPT do not violate the fuel design limits at or below 20% RTP. The required Completion Time of 4 hours is acceptable based on limiting the potential increase in local LHRs that could occur due to xenon redistribution with the QPT out of specification.

D.1

QPT in excess of the maximum setpoint specified in the COLR can be an indication of a severe power distribution anomaly, and a power reduction to at most 20% RTP ensures local LHRs do not exceed allowable limits while the cause is being determined and corrected.

The required Completion Time of 4 hours is reasonable to allow the operator to reduce THERMAL POWER to ≤ 20% RTP without challenging unit systems.

SURVEILLANCE REQUIREMENTS

QPT can be monitored by both the Incore and Excore Detector systems. The QPT setpoints are derived from their corresponding measurement system independent limits by adjustment for system observability errors and instrumentation errors. Although they may be based on the same measurement system independent limit, the limits for the different systems are not identical because of differences in the errors applicable for these systems. For QPT measurements using the Incore Detector System, the Minimum Incore Detector System consists of OPERABLE detectors configured as follows:

a. Two sets of four detectors shall lie in each core half. Each set of detectors shall lie in the same axial plane. The two sets in the same core half may lie in the same axial plane.

b. Detectors in the same plane shall have quarter core radial symmetry.

Figure B 3.2.4-1 (Minimum Incore Detector System for QPT Measurement) depicts an example of this configuration. The symmetric full Incore Detector System for QPT uses the Incore Detector System as described above and is configured such that at least 75% of the detectors in each core quadrant are OPERABLE.

SR 3.2.4.1

Checking the QPT indication every 7 days ensures that the operator can determine whether the plant computer software and Incore Detector System inputs for monitoring QPT are functioning properly, and takes into account other information and alarms available to the operator in the control room. This procedure allows the QPT mechanisms, such as xenon redistribution, burnup gradients, and CONTROL ROD drive mechanism malfunctions, which can cause slow development of a QPT, to be detected. Operating experience has confirmed the acceptability of a Surveillance Frequency of 7 days.

Following restoration of the QPT to within the setpoint, operation at ≥ 95% RTP may proceed provided the QPT is determined to remain within the setpoint at the increased THERMAL POWER level. In case QPT exceeds the setpoint for more than 24 hours (Condition A), the potential for xenon redistribution is greater. Therefore, the QPT is monitored for 12 consecutive hourly intervals to determine whether the period of any oscillation due to xenon redistribution causes the QPT to exceed the setpoint again.

REFERENCES

1. 10 CFR 50.46

2. BAW 10122A, "Normal Operating Controls," Rev. 1, May 1984.

3. 10 CFR 50.36

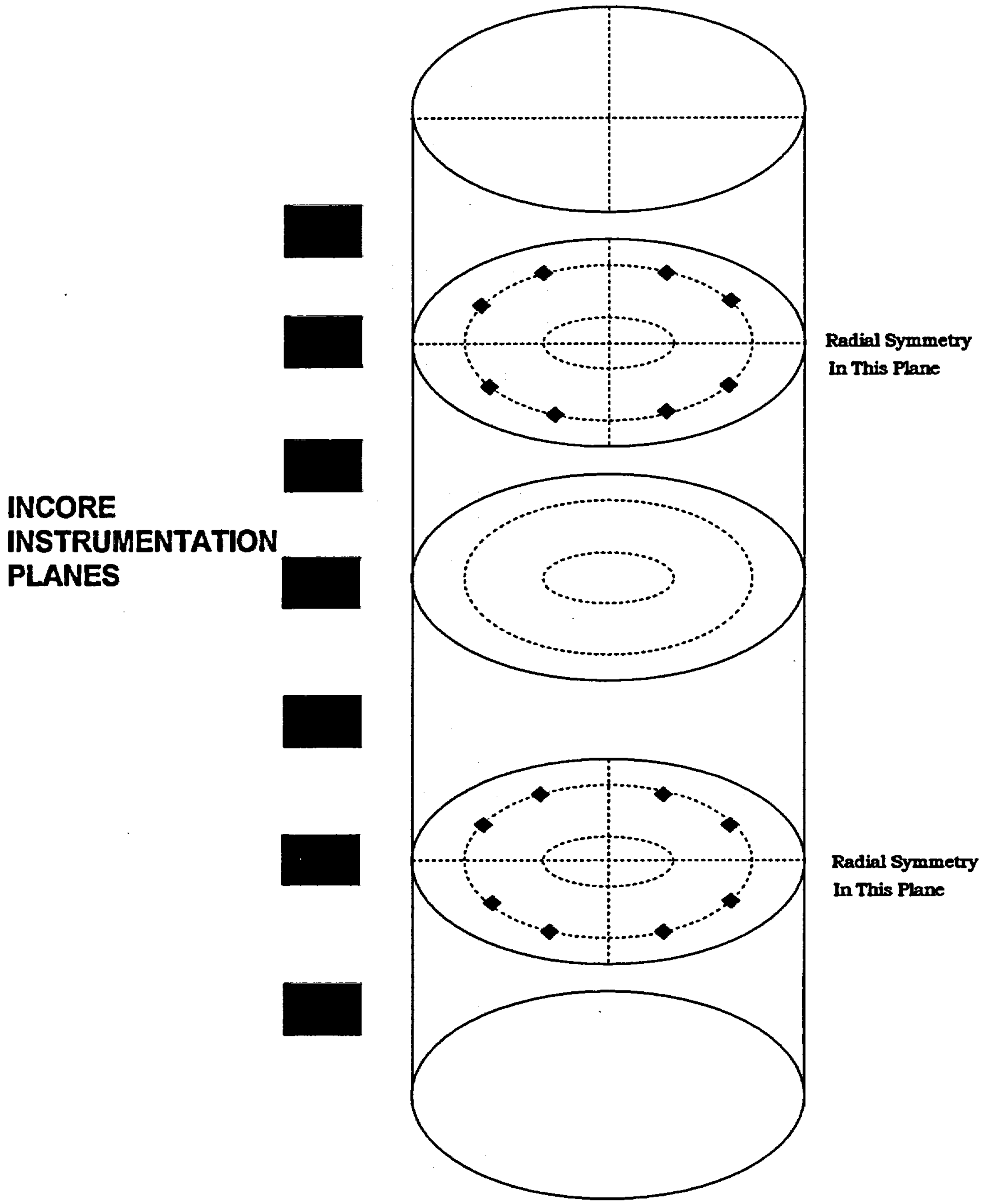


Figure B 3.2.4-1 (page 1 of 1)
Minimum Incore System for QUADRANT POWER Tilt Measurement

B 3.2 POWER DISTRIBUTION LIMITS

B 3.2.5 Power Peaking

BASES

BACKGROUND

The purpose of this LCO is to establish limits that constrain the core power distribution within design limits during normal operation, during abnormalities and such that accident initial condition protection criteria are preserved. The accident initial condition criteria are preserved by bounding operation within specified acceptable fuel design limits. This is accomplished by limiting the local linear heat rate (LHR) to three general constraints: 1) the LHR may not exceed a value that results in fuel centerline melt, 2) the LHR may not exceed a value that would result in peak cladding temperatures of greater than 2200°F during a loss of coolant accident (LOCA), and 3) the LHR may not exceed a value that would result in the minimum departure from nucleate boiling ratio (DNBR) dropping below the specified acceptable fuel design limits in the event of the limiting loss of flow transient.

The LOCA-limited LHR is a specified acceptable fuel design limit that preserves the initial conditions for the Emergency Core Cooling Systems (ECCS) analysis. The LOCA-limited LHR is dependent upon core axial location and fuel batch design. The LOCA-limited LHR may be designated as LHR in units kW/ft or as a power peaking factor. When expressed as a power peaking factor, the LOCA-limited LHR is designated as $F_Q(Z)$. $F_Q(Z)$ is defined as the maximum local fuel rod linear power density divided by the average fuel rod linear power density, assuming nominal fuel pellet and rod dimensions. Operation within the limits given by the LOCA LHR figure in the COLR prevents power generation rates that would exceed the LOCA-limited LHR limits derived from the analysis of the ECCS.

The LOCA-limited LHR bounds the fuel centerline melt LHR limit. Thus, compliance with the LOCA-limited LHR ensures compliance with the fuel centerline melt LHR.

The DNBR-limited LHR is a specified acceptable fuel design limit that preserves the initial conditions for the limiting loss of flow transient. DNBR is defined as the ratio of the heat flux that would cause departure from nucleate boiling (DNB) at a particular core location to the actual heat flux at that core location. The DNBR-limited LHR represents the linear power generation rate along the fuel rod on which the minimum DNBR occurs. Compliance with this LHR value may be accomplished: 1) by correlating the LHR at the limiting location to the critical heat flux (expressed as a LHR) for the limiting location, 2) by correlating the LHR to DNBR or DNB margin for the limiting location, or 3) by correlating the LHR to a power peaking factor (designated as $F^N_{\Delta H}$) for the limiting location.

The relationship between the observable parameters of neutron power, reactor coolant flow, temperature and pressure and the critical heat flux, DNBR or DNB margin is provided through use of a critical heat flux correlation. The critical heat flux correlations used to determine the critical heat flux for uniform and non-uniform heat flux distributions are described in the Bases for SL 2.1.1. $F^N_{\Delta H}$ is defined as the ratio of the integral of linear power along the fuel rod on which the minimum DNBR occurs to the average integrated rod power. Operation within the DNBR-limited LHR limit prevents DNB during a postulated loss of forced reactor coolant flow accident.

Measurement of the core core peaking factors using the Incore Detector System to obtain a three dimensional power distribution map provides direct confirmation that LHRs are within their limits and may be used to verify that the core local LHRs remain bounded when one or more normal operating parameters exceed their limits.

APPLICABLE SAFETY ANALYSES

The LOCA-limited LHR limits are determined by the ECCS analysis in order to limit peak cladding temperatures to 2200°F during a LOCA. The maximum acceptable cladding temperature is specified by 10 CFR 50.46 (Ref. 1). Higher cladding temperatures could cause severe cladding failure by oxidation due to a Zircaloy water reaction. Other criteria must also be met (e.g., maximum cladding oxidation, maximum hydrogen generation, coolable geometry, and long term cooling). However, peak cladding temperature is usually most limiting.

The DNBR-limited LHR limits provide protection from DNB during a limiting loss of flow transient. Proximity to the DNB condition is expressed by the DNBR, defined as the ratio of the heat flux that would cause DNB at a particular core location to the actual heat flux at that core location. The minimum DNBR value during both normal operation and anticipated transients is limited to the DNBR correlation limit for the particular fuel design in use, and is accepted as an appropriate margin to DNB. The DNBR correlation limit ensures that there is at least 95% probability at the 95% confidence level (the 95/95 DNB criterion) that the hot fuel rod in the core does not experience DNB. The critical heat flux correlations used to determine the critical heat flux for uniform and non-uniform heat flux distributions are described in the Bases for SL 2.1.1.

This LCO precludes core power distributions that violate the following fuel design criteria:

a. During a large break LOCA, peak cladding temperature must not exceed 2200°F (Ref. 1).

b. During a loss of forced reactor coolant flow accident, there must be at least 95% probability at the 95% confidence level (the 95/95 DNB criterion) that the hot fuel rod in the core does not experience a DNB condition.

The reload safety evaluation analysis determines limits on global core parameters that characterize the core power distribution. The primary parameters used to monitor and control the core power distribution are the regulating rod position, the

APSR position, the AXIAL POWER IMBALANCE, and the QPT. These parameters are normally used to monitor and control the core power distribution because their measurements are continuously observable. Limits are placed on these parameters to ensure that the core power peaking factors remain bounded during operation in MODE 1 with THERMAL POWER greater than 20% RTP. Nuclear design model calculational uncertainty, manufacturing tolerances (e.g., the engineering hot channel factor), effects of fuel densification and rod bow, and modeling simplifications (such as treatment of the spacer grid effects) are accommodated as necessary through use of peaking augmentation factors in the reload safety evaluation analysis (Ref. 2).

LHR limitations satisfy Criterion 2 of 10 CFR 50.36 (Ref. 3).

LCO

This LCO for power peaking ensures that the core operates within the LHR bounds assumed for the ECCS and thermal hydraulic analyses. Verification that LHR is within the limits of this LCO as specified in the COLR allows continued operation when the Required Actions of LCO 3.1.4, "CONTROL ROD Group Alignment Limits," LCO 3.2.1, "Regulating Rod Group Insertion Limits," LCO 3.2.2, "APSR Insertion Limits," LCO 3.2.3, "AXIAL POWER IMBALANCE Operating Limits," and LCO 3.2.4, "QUADRANT POWER TILT," are entered. Conservative THERMAL POWER reductions are required if the limits on LHR are exceeded. Verification that LHR is within the limits is also required during MODE 1 PHYSICS TESTS per LCO 3.1.8, "PHYSICS TESTS Exceptions - MODE 1."

Measurement uncertainties are applied when LHR is determined using the Incore Detector System. The measurement uncertainties applied to the measured values account for uncertainties in observability and instrument string signal processing.

APPLICABILITY

In MODE 1 with THERMAL POWER > 20% RTP, the limits on LHR must be maintained in order to prevent the core power distribution from exceeding the limits assumed in the analyses of the LOCA and loss of forced reactor coolant flow accidents. In MODE 1 with THERMAL POWER ≤ 20% RTP and in MODES 2, 3, 4, 5, and 6, this LCO is not applicable because the reactor has insufficient stored energy in the fuel or energy being transferred to the coolant to require a limit on the distribution of core power.

The minimum THERMAL POWER level of 20% RTP was chosen based on the ability of the incore detection system to satisfactorily obtain meaningful power distribution data.

ACTIONS

The operator must take care in interpreting the relationship of the LHRs, DNBRs, and power peaking factors to their limits. Limiting values may be expressed as an LHR, DNBR, margin to DNB or as power peaking factors. When expressed as power peaking factors, the value must be adjusted in inverse proportion to the THERMAL POWER level of the core as the power is reduced from RTP. Thus, the allowable peaking factors will increase as THERMAL POWER decreases.

A.1

When the LHR is determined not to be within its specified limit as determined by a three dimensional power distribution map, a THERMAL POWER reduction is taken to reduce the limiting LHR in the core. The Completion Time of 2 hours provides an acceptable time to reduce power in an orderly manner and without allowing the unit to remain in an unacceptable condition for an extended period of time.

B.1

If the Required Action and associated Completion Time for Condition A are not met, then THERMAL POWER operation should be reduced. The reactor is placed in MODE 1 with THERMAL POWER less than or equal to 20% RTP where this LCO does not apply. The required Completion Time of 4 hours is a reasonable amount of time for the operator to reduce THERMAL POWER in an orderly manner and without challenging unit systems.

SURVEILLANCE REQUIREMENTS

SR 3.2.5.1

Core power distribution monitoring is performed using the Incore Detector System to obtain a three dimensional power distribution map. Maximum LHR values obtained from this map may then be compared with the limits in the COLR to verify that the limits have not been exceeded. Minimum DNBR values or DNB margins determined from the core power distribution mapping may also be compared to their limits or correlated to LHR values to verify that the limits have not been exceeded. Measurement of the core power distribution in this manner may be used to verify that the measured LHR values remain within their specified limits when one or more of the limits specified by LCO 3.1.4, LCO 3.2.1, LCO 3.2.2, LCO 3.2.3, or LCO 3.2.4 is exceeded, or when LCO 3.1.8 is applicable. If the local LHRs remain within their limits when one or more of these parameters exceed their limits, operation at THERMAL POWER may continue because the true initial conditions (the core power distribution) remain within their specified limits.

Because the limits on LHR are preserved when the parameters specified by LCO 3.1.4, LCO 3.2.1, LCO 3.2.2, LCO 3.2.3, or LCO 3.2.4 are within their limits, a Note is provided in the SR to indicate that monitoring core local LHRs is required only when complying with the Required Actions of these LCOs and when LCO 3.1.8 is applicable.

Frequencies for monitoring of the core local LHRs are specified in the Action statements of the individual LCOs. These Frequencies are reasonable based on the low probability of a limiting event occurring simultaneously with LHR exceeding its limit, and they provide sufficient time for the operator to obtain a power distribution map from the Incore Detector System. Indefinite THERMAL POWER operation in a Required Action of LCO 3.1.4, LCO 3.2.1, LCO 3.2.2, LCO 3.2.3, or LCO 3.2.4 is permitted, because the core local LHRs assumed in the accident analyses are within analyzed core power distributions and spatial xenon distributions.

REFERENCES

1. 10 CFR 50.46.
2. BAW-10179P-A, "Safety Criteria and Methodology for Acceptable Cycle Reload Analyses," Rev. 2, October 1997.
3. 10 CFR 50.36.

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 B&W Standard Technical Specification, NUREG-1430, Revision 1. This change does not alter the requirements of the CTS or the NUREG. 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 CTS 3.5.2.6.4 establishes the Required Actions consistent with ITS 3.2.3 Condition B with the exception that a final specific power level is not explicitly established in the CTS. The final power level is implicitly established by the Applicability criteria specified in CTS 3.5.2.6.1 in that the Specification applies during power operation above 40% RTP. Based on the Applicability established in CTS 3.5.2.6.1 and the requirements of LCO 3.0.1, the maximum required power reduction would consist of placing the unit in a MODE in which the Specification no longer applied. In adopting this specified power level in CTS 3.5.2.6.4, the Required Actions have been made explicit. This change constitutes an administrative change intended to provide clarification and explicit guidance. No technical or intent change is associated with this editorial specification of an explicit power level. This change is consistent with NUREG-1430.

A4 CTS 3.5.2.4.2.b was modified to remove reference to the APSR withdrawal limits because they are not power dependent and the CTS 3.5.2.4.2.b action has no effect on the positioning of the APSRs. CTS 3.5.2.4.2.b was also modified to reflect that it applies to the regulating rods and not the safety rods. The CTS referenced the control rods indiscriminately. This is editorial because the safety rod positioning requirements of CTS 3.1.3.5 are unaffected by the QPT actions. The CTS action was also modified to specify that the setpoints shall be reduced rather than the limits. This is necessary because the COLR presents the error adjusted setpoints.

CTS 3.5.2.4.2.c was modified to refer to the operational power imbalance setpoints rather than the reactor power imbalance setpoints. This editorial change establishes consistency with the title of the figure given in the COLR.

CTS 3.5.2.4.2.b and 3.5.2.4.2.c were both modified to refer to the COLR as the location of the figures containing the setpoints modified by these CTS actions. The CTS originally referred to specific figures within these actions. These figures were relocated to the COLR in Amendment 31. However, Amendment 31 failed to incorporate a reference to the COLR. This change is editorial.

A5 Not used.

A6 In CTS 3.5.2.5.4, the exception to APSR alignment limits when performing CONTROL ROD exercise testing was shown as administratively deleted in the CTS markup. This is acceptable because this exception is not retained in the ITS. The exception need not be retained because the ITS will not require freedom of movement demonstrations (exercising) for APSRs. The freedom of movement demonstration is unnecessary since the APSRs do not insert on a reactor trip and are not contributors to the required SDM. This change is consistent with NUREG-1430.

A7 An Applicability of MODE 1 with THERMAL POWER ≥ 20% RTP is shown as adopted for ITS 3.2.5. CTS 4.1.d did not have a specific assigned Applicability. Current practice has been to require the performance of the CTS required Surveillance consistent with CTS 3.5.2.4 requirements for QUADRANT POWER TILT verification. The basis for this Applicability is the lower range of operability for the Incore Detector System. This adopted Applicability is consistent with NUREG-1430 3.2.4.

A8 The Applicability for CTS 3.1.3.5 is provided by the statement "prior to any other reduction in shutdown margin by deboration or regulating rod withdrawal during the approach to criticality." This statement precludes startup (ITS MODE 2) until the requirements of CTS 3.5.2.5 (ITS 3.2.1) are met. Because of the adoption of the Applicability of ITS 3.2.1 (MODES 1 and 2), and ITS LOC 3.0.4 (which precludes entering MODE 2 without meeting the LCO), the CTS and ITS maintain consistent requirements. Therefore, this change is administrative in nature. This change is consistent with NUREG-1430.

A9 CTS 3.5.2.4.4 established the Applicability for the CTS Quadrant Power Tilt requirements which correlate to NUREG-1430 3.2.4. The CTS established the Applicability as "during power operation above 15% of rated power." ITS 3.2.4 will establish the Applicability as MODE 1 with THERMAL POWER > 20% RTP. Both of these Applicabilities are based on the lower mode of OPERABILITY of the Incore Detector System; therefore, the adoption of the 20% RTP Applicability in the ITS is considered an Administrative change. Further, no practical operational benefit exists in raising the Applicability from 15% RTP to 20% RTP; thus, this change is not considered to result in the ITS being less restrictive with regards to the Applicability. The 20% RTP Applicability will help ensure meaningful data acquisition when using the Incore Detector System. This change is made solely to establish consistency between ITS Specifications which rely on the Incore Detector System as suggested by NUREG-1430.

CTS DISCUSSION OF CHANGES

TECHNICAL CHANGE – MORE RESTRICTIVE

M1 The CTS was marked to show adoption of ITS 3.2.4 Required Action A.2 and Conditions B and C. Required Action A.2 limits the time that the unit can operate with QPT greater than or equal to its steady state limit. This RA is necessary because of the limitations associated with the analyses that support Required Action A.1.2.1.

ITS Condition B provides the compensatory measures if the Required Actions and associated Completion Times of Condition A are not met. Continued unit operation is allowed provided THERMAL POWER is reduced to less than 60% ALLOWABLE THERMAL POWER (ATP) and the nuclear overpower trip setpoint is reduced to ≤ 65.5% ATP. These actions provide assurance of adequate core operating thermal margins and of a reasonable RPS protective action when operating with QPT above its steady state limit. The adoption of the Required Action is more restrictive in that no comparable CTS action is provided.

The adoption Condition C is more restrictive in that it will direct a reduction in THERMAL POWER to less than or equal to 20% RTP with a Completion Time of 4 hours. This action is necessary because it removes the unit from the LCO Applicability if the Condition B Required Actions can not be completed within the specified Completion Times. The 4 hour Completion Time is based on the need to take prompt corrective actions to reduce the core THERMAL POWER level when operating with a QUADRANT POWER TILT greater than its limits while adhering to unit operating procedures governing normal, non-emergency, power maneuvering rates of ≤30% per hour. This Completion Time also recognizes the low probability of an accident occurring coincident with the QUADRANT POWER TILT not within its limits. The CTS provided no explicit requirements when QPT was in excess of the limits for a period of time in excess of the CTS 3.5.2.4.2 completion time. This situation would have required entry into CTS 3.0.3 which would have allowed an indeterminate period of time, not to exceed 7 hours, to be below the CTS 3.5.2.4.4 applicability of 15% rated power. Adoption of ITS 3.2.4 Condition C, provides Required Actions and associated Completion Times where none existed in CTS. These changes are consistent with NUREG-1430.

M2 ITS 3.2.2 Condition B is shown on the CTS markup to indicate its adoption in the ITS. Currently, failure to provide compliance with the required actions given in CTS 3.5.2.5.4 would result in entry into CTS 3.0.3. ITS 3.2.2 Required Action B.1 provides explicit guidance should the Required Action or Completion Time of Condition A not be satisfied. The adoption of the specific requirements of Condition B constitutes a more restrictive change in that CTS 3.0.3 would have provided an hour for restoration of the LCO and a total of 13 hours to reach ITS MODE 3 equivalent conditions; whereas, the ITS will simply direct shutdown of the unit (establish MODE 3) within 6 hours. This change is consistent with NUREG-1430.

M3 CTS 3.5.2 defines its Applicability as being "during power operation." While the regulating rod and APSR insertion limits found in CTS 3.5.2 are in-practice applied during both Power Operations and Hot Standby conditions, the applicability of these requirements during both of these operating conditions is not clearly expressed in the CTS. The regulating rod and APSR insertion limits found in CTS 3.5.2 are being replaced by ITS 3.2.1 and ITS 3.2.2. ITS 3.2.1 and ITS 3.2.2 will have Applicability specified as MODES 1 and 2. The adoption of this Applicability represents more restrictive operating requirements than those presently specified in the CTS. By specifying Applicability in MODE 2, in addition to MODE 1, additional requirements have been added where none were previously specified. This change is consistent with NUREG-1430.

M4 ITS 3.2.1 Condition D requirements will be more restrictive than CTS 3.5.2.5.3 requirements for situations in which the regulating rod groups are inserted into the unacceptable operation region of the regulating group rod position limits given in the COLR. The Completion Time for restoring the regulating group insertion to within limits will be 2 hours (ITS 3.2.1 Required Action D.2.1), or a reduction in THERMAL POWER to less than or equal to the THERMAL POWER allowed by the regulating rod group insertion limits will be required within 2 hours (ITS 3.2.1 Required Action D.2.2). These ITS Required Actions will be more restrictive than the present 4 hour restoration requirement established by CTS 3.5.2.5.3. This change is consistent with NUREG-1430.

M5 ITS SR 3.2.1.1, SR 3.2.1.2 and SR 3.2.2.1 have been adopted. These SRs provide requirements for verifying that regulating rod groups are within the required sequence and overlap limits (SR 3.2.1.1), insertion limits (SR 3.2.1.2), and that the APSRs are within acceptable position limits (SR 3.2.2.1). This verification ensures that the initial conditions of the accident analyses are satisfied during operation. The adoption of these SRs represent more restrictive requirements because no comparable CTS SRs exist. This change is consistent with NUREG-1430 for SR 3.2.2.1 and NUREG-1430 as modified by TSTF-110, Rev 1 for SR 3.2.1.1 and SR 3.2.1.2.

M6 CTS 3.5.2.4.3 allowed continued operation of the unit above hot shutdown with QPT in excess of the maximum limit, for the purposes of "physics tests" and "diagnostic testing." Under this allowance, the unit could have operated at THERMAL POWER levels up to approximately 60% RTP (with four RCPs operating). ITS 3.2.4 Condition D will require that THERMAL POWER be reduced to less than or equal to 20% RTP within 4 hours. Thus, adoption of the ITS requirement is more restrictive. This Required Action is appropriate because: 1) it serves to remove the unit from the LCO Applicability; 2) it limits the THERMAL POWER level to a magnitude that will not exceed the thermal design limits of the core; and 3) it permits continued operation which may be necessary to resolve the cause of the QPT. The 4 hour Completion Time is based on the need to take prompt corrective actions to reduce the core THERMAL POWER level when operating with QPT greater than its maximum limit while adhering to unit operating procedures governing normal, non-emergency, power maneuvering rates of ≤30% per hour. The 4 hour Completion Time provides a reasonable period of time for the reactor operator to reduce the THERMAL POWER of the unit during a situation in which QPT has been made to exceed its maximum limit. This Completion Time also recognizes the low probability of an accident occurring coincident with the QPT not within its maximum limit. The adoption of the 4 hour Completion Time in the ITS will be more restrictive because the CTS did not previously establish a Completion Time for this required power reduction. This change is consistent with NUREG-1430.

M7 CTS 3.5.2.5.3 established the regulating rod group position and sequence requirements that correlate to ITS LCO 3.2.1. The CTS established that "corrective measures will be taken immediately" and that acceptable "positions shall be attained within 4 hours." However, in the event that compliance is not attained within 4 hours, CTS 3.0.3 would require the unit be in hot shutdown within 7 hours. In the ITS, should the requirements not be met as directed by other Actions, ITS 3.2.1 Required Action E.1 will establish that the unit be placed in MODE 3 within 6 hours. The more restrictive Completion Time is considered appropriate because of the potential reactivity effects and uncertainty associated with regulating rod group reactivity worth when sequence or overlap requirements are not met. This change is consistent with NUREG-1430.

M8 CTS 3.5.2.6.4 was modified to reflect that the required power reduction must be accomplished within a Completion Time of 4 hours. The 4 hour Completion Time is based on the need to take prompt corrective actions to reduce the core THERMAL POWER level when operating with an AXIAL POWER IMBALANCE greater than its limits while adhering to unit operating procedures governing normal, non-emergency, power maneuvering rates of ≤30% per hour. The 4 hour Completion Time provides a reasonable period of time for the reactor operator to reduce the THERMAL POWER of the unit during a situation in which AXIAL POWER IMBALANCE has been made to exceed its limits. This Completion Time also recognizes the low probability of an accident occurring coincident with the AXIAL POWER IMBALANCE not within its limits. The adoption of the 4 hour Completion Time in the ITS will be more restrictive because the CTS did not previously establish a Completion Time for this required power reduction.

M9 CTS 4.1.d provides a required surveillance with no corresponding LCO or Actions. Therefore, ITS LCO 3.2.5 Conditions A and B are shown as adopted on the CTS mark-up. Condition A establishes the Required Action and Completion Time should the linear heat rate (LHR) not be within its limit. Condition B establishes the Required Action and Completion Time should Condition A not be satisfied. These actions are necessary to establish un-ambiguous guidance for the Actions necessary to mitigate those circumstances that may have resulted in excessive linear heat rates. The adoption of these Conditions is shown as more restrictive because these Required Actions were not contained in the CTS.

M10 The CTS markup shows the adoption of ITS 3.2.1 Required Action A.1 and its associated Note. This Required Action provides verification of acceptable core power distribution, specifically local core linear heat rates (local power peaking), during conditions where the regulating rod group is inserted into the restricted operation region given on a figure in the COLR. This verification preserves the initial conditions of the ECCS accident analysis and DNBR analysis for loss of forced reactor coolant flow. In the ITS, the performance of this RA on a 2 hour Completion Time will allow continued unit operation for up to 24 hours. The Note indicates that the RA is only required to be performed when the THERMAL POWER level is greater than 20% RTP. This establishes an applicability for the RA that is consistent with the ITS 3.2.5 Applicability. The adoption of this RA, and its associated Note, imposes more restrictive requirements in that no similar requirements existed in the CTS.

Refer to ITS 3.2.1 Required Action A.2 and DOC L6 regarding the less restrictive aspects of this change. This change is consistent with NUREG-1430 as modified by TSTF-160, Rev 1.

M11 The CTS markup shows the adoption of ITS 3.2.2 Required Action A.1 and its associated Note. This Required Action provides verification of acceptable core power distribution, specifically local core linear heat rates (local power peaking), during conditions where the axial power shaping rod (APSR) group is not positioned within the limits of the COLR. This verification preserves the initial conditions of the ECCS accident analysis and DNBR analysis for loss of forced reactor coolant flow. In the ITS, the performance of this RA with a 2 hour periodic Completion Time will allow continued unit operation for up to 24 hours (ITS 3.2.2 Required Action A.2). The Note indicates that the RA is only required to be performed when the THERMAL POWER level is greater than 20% RTP. This establishes an applicability for the RA that is consistent with the ITS 3.2.5 Applicability.

The adoption of this RA, and its associated Note, imposes more restrictive requirements in that no similar requirements existed in the CTS. Further, if the RA is not completed within its specified 2 hour periodic Completion Time or is otherwise incapable of being completed, then ITS 3.2.2 Required Action B.1 would require that the unit be placed in MODE 3 within 6 hours. Thus, the ITS imposes a conditional Action that was not present in the CTS. The CTS allows 4 hours to complete the required action regardless of the ability to perform a verification of acceptable core power distribution.

Refer to ITS 3.2.2 Required Action A.2 and DOC L4 regarding the less restrictive aspects of this change. This change is consistent with NUREG-1430 as modified by TSTF-160, Rev 1.

M12 The CTS markup shows the adoption of ITS 3.2.3 Required Action A.1. This Required Action provides verification of acceptable core power distribution, specifically local core linear heat rates (local power peaking), during conditions where AXIAL POWER IMBALANCE is not within the limits of the COLR. This verification preserves the initial conditions of the ECCS accident analysis and DNBR analysis for loss of forced reactor coolant flow. In the ITS, the performance of this RA with a 2 hour periodic Completion Time will allow continued unit operation for up to 24 hours (ITS 3.2.3 Required Action A.2).

The adoption of this RA imposes more restrictive requirements in that no similar requirements existed in the CTS. Further, if the RA is not completed within its specified 2 hour Completion Time or is otherwise incapable of being completed, then ITS 3.2.3 Required Action B.1 would require that THERMAL POWER be reduced to less than or equal to 40% RTP within 4 hours. Thus, the ITS imposes a conditional Action that was not present in the CTS. The CTS allows 4 hours to complete the required action regardless of the ability to perform a verification of acceptable core power distribution.

Refer to ITS 3.2.3 Required Action A.2 and DOC L5 regarding the less restrictive aspects of this change. This change is consistent with NUREG-1430.

M13 CTS 3.5.2.4.1 presents the required action to reduce the THERMAL POWER level of the unit should the QUADRANT POWER TILT exceed its limits. CTS 3.5.2.4.2 establishes a 4 hour completion time for the power reduction. ITS 3.2.4 Required Action A.1.2.1 will require this power reduction be accomplished within 2 hours of entry into the Condition or 2 hours after the last performance of SR 3.2.5.1 (ITS 3.2.4 Required Action A.1.1). The 2 hour Completion Time is necessary to ensure that local linear heat rates are maintained within acceptable limits while limiting the potential for xenon redistribution. This change is consistent with NUREG-1430.

M14 The CTS markup shows the adoption of ITS 3.2.4 Required Action A.1.1. This Required Action provides verification of acceptable core power distribution, specifically local core linear heat rates (local power peaking), during conditions where QUADRANT POWER TILT is not within the steady state limits presented in the COLR. This verification preserves the initial conditions of the ECCS accident analysis and DNBR analysis for loss of forced reactor coolant flow. In the ITS, the performance of this RA on a 2 hour Frequency will allow unrestricted unit operation for up to 24 hours as long as the linear heat rate (power peaking) criteria are met.

The adoption of this RA imposes more restrictive requirements in that no similar requirements existed in the CTS. Further, if the RA is not completed within its specified 2 hour periodic Completion Time or is otherwise incapable of being completed, then ITS 3.2.4 Required Action A.1.2.1 would require that within 2 hours THERMAL POWER be reduced ≥2% RTP from the ALLOWABLE THERMAL POWER for each 1% of QPT greater than the limit. Thus, the ITS imposes a conditional Action that was not present in the CTS. Further, the CTS allows 4 hours to complete the required action regardless of the ability to perform a verification of acceptable core power distribution.

Refer to DOC L10 regarding the less restrictive aspects of this change. This change is consistent with NUREG-1430.

M15 The CTS markup shows the adoption of a second Completion Time for ITS 3.2.4 Required Action A.1.2.1. This second Completion Time imposes the requirement to complete the required THERMAL POWER reduction within 2 hours following the last performance of SR 3.2.5.1. This Completion Time limits the time that the unit may operate with a QPT coincident with a potential excessive core linear heat rate or excessive power peaking. The adoption of this Completion Time is more restrictive because the CTS had no similar SR requirement and merely required a THERMAL POWER reduction with a 4 hour completion time. This change is consistent with NUREG-1430.

M16 The CTS markup shows the adoption of a second Frequency for ITS SR 3.2.4.1. This second Frequency imposes the requirement to complete the SR at one hour intervals for 12 consecutive hours, or until verified acceptable at ≥ 95% RTP, following the restoration of QPT within limits. This Frequency is used to determine whether the period of any oscillation due to xenon redistribution might cause the QPT to subsequently exceed the limit. This change is more restrictive because the CTS contained no similar SR Frequency requirements. This change is consistent with NUREG-1430.

M17 CTS 4.1.d established the requirements for core power distribution measurement. LCO 3.2.5 will establish similar requirements in the ITS. The principle difference in the ITS will be that the Surveillance (SR 3.2.5.1) is only performed when directed by LCO 3.1.8, "PHYSICS TESTS Exceptions - MODE 1," or by the Required Actions of LCO 3.1.4, "CONTROL ROD Group Alignment Limits"; LCO 3.2.1, "Regulating Rod Insertion Limits"; LCO 3.2.2, "AXIAL POWER SHAPING ROD (APSR) Insertion Limits"; LCO 3.2.3, "AXIAL POWER IMBALANCE Operating Limits"; LCO 3.2.4, "QUADRANT POWER TILT (QPT)." This represents a more restrictive requirement than the CTS which required the performance of the Surveillance on a 10 effective full power day (EFPD) frequency. This is more restrictive because it requires a repetitive performance of the SR while operating in accordance with the Required Actions of the above LCOs. Further, if SR 3.2.5.1 is not performed, or is incapable of being performed within the required Completion Time, then a THERMAL POWER reduction is required within a shorter Completion Time than that established within the CTS.

This change in Frequency is acceptable because the steady state design considerations of the core ensure margin to the thermal operating limits which are easily preserved while operating in accordance with the LCO requirements previously listed. Thus, the 10 EFPD Frequency only provides a confirmation of already known conditions. However, when required because of a failure to meet one or more of the ITS LCOs (listed above), SR 3.2.5.1 is performed to ensure the continued acceptability of the core's local linear heat rates. This verification ensures the continued compliance with the core power distribution assumptions of the accident analyses even though specific LCO requirements may not be met. Thus, the ITS SR Frequency will better ensure the continued compliance with the safety analysis initial condition assumptions regarding core power distribution.

This change is consistent with NUREG-1430.

M18 CTS 3.5.4 established requirements for the OPERABILITY of the incore instrumentation system when above 80% of operating power determined by the reactor coolant pump combination (equivalent to 80% ATP in the ITS). The last paragraph of CTS 3.5.4 provided an action that if the incore detector system is inoperable, the system was not to be used for the applicable function (i.e., axial imbalance determination or radial tilt determination). The ITS will require the incore detector system to be OPERABLE anytime it is providing the required monitoring function specified in ITS 3.2.3, 3.2.4 and 3.2.5. This extends the Applicability for this system's OPERABILITY down to 40% RTP when satisfying ITS 3.2.3 and down to 20% RTP when satisfying ITS 3.2.4 and 3.2.5. Therefore, the ITS will impose requirements on system OPERABILITY that are more restrictive than those in the CTS. This more restrictive requirement is appropriate because it establishes monitoring system OPERABILITY requirements consistent with the Applicability of the LCOs for the parameters being monitored. This change is consistent with NUREG-1430.

TECHNICAL CHANGE -- LESS RESTRICTIVE

L1 The CTS markup shows the adoption of the Completion Times for ITS 3.2.4 Required Actions A.1.2.2, A.1.2.3 and A.1.2.4. CTS 3.5.2.4.2 establishes the required actions if the QUADRANT POWER TILT is not restored to within its limits. The CTS required actions correlate to ITS 3.2.4 Required Actions A.1.2.1, A.1.2.2, A.1.2.3 and A.1.2.4. The CTS states that these actions are to be completed within 4 hours. ITS Required Actions A.1.2.2, A.1.2.3 and A.1.2.4 have a Completion Time of 10 hours from entry into the Condition or 10 hours following the last performance of SR 3.2.5.1. Adoption of the ITS Completion Time effectively lengthens by 6 hours the amount of time allowed for the completion of these corrective Actions. The 10 hour Completion Time is considered appropriate in light of the 2 hour Completion Time associated with ITS Required Action A.1.2.1 and its required reduction in THERMAL POWER. During the course of reducing the THERMAL POWER level of the unit, it is considered imprudent to be simultaneously adjusting the setpoints of the Reactor Protection System and attempting to modify the operational restraints governing regulating rod position and axial power imbalance setpoints. The adoption of the 6 additional hours provides sufficient time for an orderly power reduction followed by an orderly execution of the tasks associated with ITS 3.2.4 Required Actions A.1.2.2, A.1.2.3 and A.1.2.4. The 10 hour Completion Time is consistent with NUREG-1430.

L2 Not Used.

L3 CTS 3.5.2.4.3 establishes Required Actions that are inconsistent with CTS 3.0.1 requirements and ITS LCO 3.0.1 requirements. Specifically, the CTS directs that the Unit be placed in hot shutdown (reactor subcritical) if the QUADRANT POWER TILT is in excess of 25% *unless diagnostic testing is to be performed or is being performed,* in which case, the unit is allowed to continue to operate provided THERMAL POWER is maintained below the ALLOWABLE THERMAL POWER as adjusted by CTS 3.5.2.4.1. CTS 3.5.2.4, as applied at ANO-1, is applicable when operating at greater than 15% of rated power. This applicability is based on the surveillance requirement found in CTS 3.5.2.4.4. The requirement to go to CTS hot shutdown (equivalent to ITS MODE 3) rather than to exit the Applicability (< 15% of rated power) presents required actions inconsistent with the requirements of CTS 3.0.1.

ITS 3.2.4 is Applicable in MODE 1 with THERMAL POWER above 20% RTP. NUREG 3.2.4 Condition F establishes the Required Actions if QPT is greater than the maximum limit. NUREG 3.2.4 Condition F establishes that the THERMAL POWER level of the unit be reduced to less than or equal to 20% RTP. The ITS will adopt this required reduction to less than or equal to 20% RTP as the Required Action for Condition D. This change represents less restrictive requirements in that continued operation, below 20% RTP, with QPT greater than the limits specified in the COLR, will be allowed even while not performing PHYSICS TESTS or "diagnostic testing." This change is consistent with NUREG-1430 3.2.4 Action F, LCO 3.0.1 and LCO 3.0.2.

Note - a discussion regarding the difference in Applicability between CTS 3.5.2.4 and ITS 3.2.4 is given in Section 3.2 DOC A9.

L4 CTS 3.5.2.5.4 established the LCO requirements and associated required actions for the AXIAL POWER SHAPING RODS (APSRs). The CTS required that the APSRs be restored to within their limits within 4 hours. ITS 3.2.2 Required Action A.2 will allow up to 24 hours to restore the APSRs to within their limits provided that core power distribution is being monitored at 2 hour intervals (Required Action A.1). The ITS will impose less restrictive requirements in that the unit will be allowed to operate for a longer period of time with the APSRs not in accordance with their position limits. However, this extension is only possible if ITS 3.2.2 Required Action A.1 is being performed which ensures the acceptability of the core power distribution. ITS 3.2.2 Required Action A.1 is only required when THERMAL POWER is greater than 20% RTP. The extension in the allowed operating time is acceptable because the initial conditions of the safety analyses are preserved by verification, using the Incore Detector System, that core power distribution is within the initial conditions of the safety analyses while operating at greater than 20% RTP. When operating below 20% RTP with the APSRs not positioned in accordance with their limits, the extension in the allowed operating time is acceptable because of the large operating margins that exist in the core. The CTS did not provide a comparable required action to perform core power distribution verification. This change is consistent with NUREG-1430 as modified by TSTF-160.

L5 CTS 3.5.2.6.3 and CTS 3.5.2.6.4 established the required actions for AXIAL POWER IMBALANCE not within limits. The CTS required that the AXIAL POWER IMBALANCE be restored to within its limits within 4 hours. ITS 3.2.3 Required Action A.2 will allow up to 24 hours to restore the AXIAL POWER IMBALANCE to within its limit provided that core power distribution is being monitored at 2 hour intervals (Required Action A.1). The ITS will impose less restrictive requirements in that the unit will be allowed to operate for a longer period of time with AXIAL POWER IMBALANCE not in accordance with its limit. However, this extension is only possible if ITS 3.2.3 Required Action A.1 is being performed which ensures the acceptability of the core power distribution. This extension is acceptable because the initial conditions of the safety analyses are preserved by verification that core power distribution is within the initial conditions of the safety analyses. The CTS did not provide a comparable required action to perform core power distribution verification. This change is consistent with NUREG-1430.

L6 CTS 3.5.2.5.3 directed that if the position setpoints were exceeded then corrective measures shall be immediately taken to achieve an acceptable CONTROL ROD position and that the acceptable CONTROL ROD position be achieved within 4 hours. The ITS will adopt NUREG-1430 3.2.1 as modified by TSTF-345. ITS 3.2.1 will establish Required Actions based on the safety significance of not having the regulating rod group position, sequence or required overlap within the limits. The Required Actions will be based on: 1) regulating rod group insertion into the restricted operation region (ITS 3.2.1 Condition A and B); 2) regulating rod group insertion in an incorrect sequence or group overlap requirements not within the limits (ITS 3.2.1 Condition C); or 3) regulating group insertion into the unacceptable operation region (ITS 3.2.1 Condition D). The ITS provides differentiation between the types of regulating rod group deviations, given above, that were not differentiated between in the CTS.

The ITS and CTS requirements will be similar for situations in which the regulating rod groups are inserted into the restricted operation region and the core power distribution is not being periodically verified. However, ITS provides a less restrictive Completion Time for restoration of adherence to the limits (24 hours from discovery of failure to meet the LCO (ITS 3.2.1 Required Action A.2)), provided that periodic surveillance of an acceptable linear heat rate (ITS 3.2.1 Required Action A.1) is performed at 2 hour intervals. If this surveillance is not performed, then ITS 3.2.1 Required Action B.1 requires a reduction in THERMAL POWER with a Completion Time of 2 hours. Similarly, if the surveillance determines that the linear heat rates are not within limits, the Actions of ITS 3.2.5 also require a power reduction within 2 hours. For the scenario where the linear heat rate surveillance is not performed, the combination of the Completion Times for Required Actions A.1 and B.1 maintains the present 4 hour restoration requirement established by CTS 3.5.2.5.3. This change is less restrictive because when the linear heat rate surveillance is being periodically performed the Completion Time is 24 hours. This Completion Time is acceptable for the following reasons:

1) The SDM requirements and ejected rod worth limitations are maintained by the fact that the regulating rod group is not inserted out-of-sequence, proper overlap requirements are met, and the group is not inserted into the unacceptable operation region as given in the COLR. ITS Conditions C and D would apply to the other cases and provide appropriate Required Actions.

2) During non-transient conditions, the power redistribution effects would be generally slow and limited to those associated with changes in the local xenon concentrations. Unacceptable changes in power distribution would be apparent as a result of the verification of acceptable core power distributions through the performance of ITS 3.2.1 Required Action A.1 and through observation of changes in other monitored core parameters such as AXIAL POWER IMBALANCE and QUADRANT POWER TILT. During transient conditions, other indication in the control room is available to indicate the upset condition of the unit. This indication is more than adequate to make a determination of whether the event has the potential to induce significant power redistribution.

3) For situations in which the regulating rod group was inserted into the unacceptable operation region (beyond the insertion limit) of the COLR figure, the ITS Required Action will result in the initiation of boration within 15 minutes. And, the regulating rod group position must be returned to the acceptable operation region given on the COLR figure or THERMAL POWER must be reduced to less than or equal to the THERMAL POWER allowed by the regulating group insertion limits within 2 hours (ITS 3.2.1 Condition D). The 15 minute Completion Time for initiation of boration serves to ensure maintenance of an adequate SHUTDOWN MARGIN and preservation of the limitations on ejected rod worth.

ITS Condition C will address those situations where the regulating rod group sequence or overlap requirements are not met. Required Action C.1 requires that the regulating rods be restored to within limits with a Completion Time of 4 hours, consistent with CTS 3.5.2.5.3. Therefore, this aspect of the ITS may be more restrictive. This change is consistent with NUREG-1430 as modified by TSTF-345 (except for ITS Required Action C.1 as discussed above, which is consistent with CTS).

L7 Not Used.

L8 CTS 3.1.3.5 established the LCO requirements for safety rod and regulating rod group positions as limited by CTS 3.5.2.1. CTS 3.5.2.1 established the requirement, that during power operation, the available SHUTDOWN MARGIN (SDM) be greater than or equal to the limit specified in the COLR with the highest worth CONTROL ROD fully withdrawn. In addition, CTS 3.5.2.1 established the Required Action should this SDM requirement not be satisfied (i.e., immediately initiate and continue boration until the required SDM is met).

All CTS requirements for SDM will be maintained in the ITS. However, the ITS will be less restrictive than the CTS in that the ITS will specify a Completion Time for the initiation of boration as 15 minutes (Ref. ITS Required Action D.1). The CTS specifies that this be initiated immediately. The 15 minute Completion Time of the ITS is acceptable because it presents a realistic time frame for the required operator manipulations to establish emergency boration. The 15 minute Completion Time is also acceptable in light of the low probability of an accident occurring within this relatively short time frame. This change is consistent with NUREG-1430.

L9 CTS 3.5.2.6.1 established a surveillance frequency of 2 hours for monitoring AXIAL POWER IMBALANCE. ITS SR 3.2.3.1 will have with a Frequency of 12 hours. The 12 hour Frequency is appropriate because the mechanisms that can cause AXIAL POWER IMBALANCE, such as xenon redistribution or CONTROL ROD drive mechanism malfunctions that cause AXIAL POWER IMBALANCE increase, can be discovered by the operator before the specified limits are violated. This is supported by the availability of other indication in the control room that would alert the operator of the presence of malfunctions likely to induce an AXIAL POWER IMBALANCE. This change is consistent with NUREG-1430 as modified by TSTF-110, Rev 2.

L10 The CTS markup shows the adoption of NUREG-1430 3.2.4 Required Action A.1.1. This Required Action directs the performance of SR 3.2.5.1 at 2 hour intervals. The structure of the ACTIONS in the ITS will allow unrestricted unit operation for up to 24 hours as long as this RA indicates that core local linear heat rates (power peaking) are within acceptable limits. This verification ensures that the safety analysis initial condition assumptions regarding core power distribution are met. Adoption of this Required Action is less restrictive than CTS requirements because a mandatory power reduction will not be required unless indicated as being necessary through performance of the RA, or as a result of a failure to perform the RA. The adoption of this Required Action is acceptable because the RA directly confirms the acceptability of the local linear heat rates within the core. This change is consistent with NUREG-1430.

L11 CTS 4.1.d established the requirements for core power distribution measurement. LCO 3.2.5 will establish similar requirements in the ITS. The principle difference in the ITS will be that the Surveillance (SR 3.2.5.1) is only performed when directed by LCO 3.1.8, "PHYSICS TESTS Exceptions - MODE 1," or by the Required Actions of LCO 3.1.4, "CONTROL ROD Group Alignment Limits"; LCO 3.2.1, "Regulating Rod Insertion Limits"; LCO 3.2.2, "AXIAL POWER SHAPING ROD (APSR) Insertion Limits"; LCO 3.2.3, "AXIAL POWER IMBALANCE Operating Limits"; and LCO 3.2.4, "QUADRANT POWER TILT (QPT)." This represents a less restrictive requirement than the CTS which required the performance of the Surveillance on a 10 effective full power day (EFPD) frequency. Note that periodic incore power distribution maps will continue to be performed per the recommendations from the core designer for the purpose of verifying core behavior methodology assumptions and determining fuel depletion characteristics.

This change in Frequency is acceptable because the steady state design considerations of the core ensure margin to the thermal operating limits which are easily preserved while operating in accordance with the LCO requirements previously listed. Thus, the 10 EFPD Frequency provides a confirmation of already known conditions. However, when required because of a failure to meet one or more of the ITS LCOs (listed above), SR 3.2.5.1 is performed to ensure the continued acceptability of the core's local linear heat rates. This verification ensures the continued compliance with the core power distribution assumptions of the accident analyses even though specific LCO requirements may not be met. Thus, the ITS SR Frequency will better ensure the continued compliance with the safety analysis initial condition assumptions regarding core power distribution.

Also shown on the CTS markup was the annotation that the SR 3.2.5.1 Note was being adopted. The adoption of this Note is an administrative function associated with the structure and format of NUREG-1430. The Note is discussed here because of its relationship with the change in SR Frequency.

The adoption of the SR Note and Frequency is consistent with NUREG-1430.

L12 CTS 3.5.2.4.4 established a QUADRANT POWER TILT (QPT) Surveillance Frequency of 2 hours. NUREG-1430, as modified by TSTF-110, establishes a Frequency of 7 days. ITS SR 3.2.4.1 will adopt this Frequency. The ITS SR Frequency is based on the relatively slow changing nature of the QPT during steady state conditions. During transient conditions, other indication is available in the control room to alert the operator to plant conditions that may result in QPT exceeding its limit. While operating within the Actions of other ITS LCOs due to events likely to induce power redistribution effects, the Required Actions directing performance of SR 3.2.5.1 are more than adequate in verifying an acceptable power distribution within the core. Thus, the reduction in SR Frequency is acceptable.

This change is consistent with NUREG-1430 as modified by TSTF-110, Rev 2.

LESS RESTRICTIVE – ADMINISTRATIVE DELETION OF REQUIREMENTS

LA1 This information has been moved to the SAR, COLR, ITS Bases, or 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 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. Changes to the SAR, COLR, and TRM are controlled by 10 CFR 50.59. Changes to the ITS Bases will be controlled in accordance with the Bases Control Program. This change is consistent with NUREG-1430.

CTS Location	New Location
3.5.2.4.3 (25% tilt limit value)	COLR
3.5.2.5.2 (Overlap value only)	COLR
3.5.2.7	SAR (7.2.2.3.2)
3.5.4	Bases (3.2.3 & 3.2.4)
3.5.4.1	Bases (3.2.3)
3.5.4.2	Bases (3.2.4)
Table 4.1-1, Item 39	TRM

3.2.1

<LATER> (3.4A) 3.1.3 Minimum Conditions for Criticality LATER

Specification LATER

<LATER> (3.4A) <LATER> (3.1) 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) 3.1.3.3 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

3.2.1 RA D.1

<LATER> (3.1) 3.2.1 AA1 LATER (3.1) 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. MODES 1+2 LATER A8 + 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

<LATER> (3.1 3.4A-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 within the next 15 minutes. -LATER

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. A2

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.

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.

A2

REFERENCES

(1) FSAR, Section 3

(2) FSAR, Section 3.2.2.1.5

3.2.1
3.2.2

{(LATER) (3.1)

3.5.2 Control Rod Group and Power Distribution Limits

3.2.1 APPL
3.2.2 APPL

{LATER

M3

Applicability

This specification applies to power distribution and operation of control rods during power operation. MODES 1 & 2

(LATER) (3.1)

LATER

Objective

~~To assure an acceptable core power distribution during power operation, to set a limit on potential reactivity insertion from a hypothetical control rod ejection, and to assure core subcriticality after a reactor trip.~~

Specification

3.2.1 LCO
3.2.1 RA D.1
& (LATER) (3.1)

L8

& LATER

3.5.2.1 The available shutdown margin shall be greater than or equal to that specified in the COLR with the highest worth control rod fully withdrawn. With the shutdown margin less than that required, immediately initiate and continue boration injection until the required shutdown margin is restored.

(LATER) (3.1)

LATER

~~3.5.2.2 Operation with inoperable rods:~~

~~1. Operation with more than one inoperable rod, as defined in Specification 4.7.1 and 4.7.2.3, in the safety or regulating rod groups shall not be permitted.~~

~~2. If a control rod in the regulating or safety rod groups is declared inoperable in the withdrawn position as defined in Specification 4.7.1.1 and 4.7.1.3, an evaluation shall be initiated immediately to verify the existence of an available shutdown margin greater than or equal to that specified in the COLR. Boration may be initiated either to the worth of the inoperable rod or until the regulating and transient rod groups are withdrawn to the limits of Specification 3.5.2.5.3, whichever occurs first. Simultaneously a program of exercising the remaining regulating and safety rods shall be initiated to verify operability.~~

~~3. If within one (1) hour of determination of an inoperable rod as defined in Specification 4.7.1, it is not determined that an available shutdown margin greater than or equal to that specified in the COLR exists combining the worth of the inoperable rod with each of the other rods, the reactor shall be brought to the Hot Standby condition until this margin is established.~~

~~4. Following the determination of an inoperable rod as defined in Specification 4.7.1, all remaining rods shall be exercised within 24 hours and exercised weekly until the rod problem is solved.~~

~~5. If a control rod in the regulating or safety rod groups is declared inoperable per 4.7.1.2, power shall be reduced to 60% of the thermal power allowable for the reactor coolant pump combination.~~

<Add SR 3.2.4.1 Frequency> M16

3.2.4

<Add 3.2.4 RA A.1.2.1 Comp. Time> M15

<Add 3.2.4 RA A.1.1> M14 L10

<Add 3.2.4 RA A.1.2.2, A.1.2.3, & A.1.2.4 CTs> L1

<LATER> (3.1)

LATER

~~6. If a control rod in the regulating or axial power shaping groups is declared inoperable per Specification 4.7.1.2 operation above 60 percent of the thermal power allowable for the reactor coolant pump combination may continue provided the rods in the group are positioned such that the rod that was declared inoperable is contained within allowable group average position limits of Specification 4.7.1.2 and the withdrawal limits of Specification 3.5.2.5.3.~~

~~3.5.2.3 The worth of single inserted control rods during criticality are limited by the restrictions of Specification 3.1.3.5 and the Control Rod Position Limits defined in Specification 3.5.2.5.~~

A1

3.5.2.4 Quadrant Power Tilt (QPT) LATER

A1

<LATER> (3.1)

3.2.4 LCO

3.2.4 RA A.1.2.1

1. ~~Except for physics tests,~~ if quadrant power tilt exceeds the tilt limit set in the CORE OPERATING LIMITS REPORT, reduce power so as not to exceed the allowable THERMAL power level for the existing reactor coolant pump ~~combination less at least 2%~~ for each 1% tilt in excess of the tilt limit. A1

<LATER> (3.1)

2. Within a period of ~~4~~ 2 hours, the quadrant power tilt shall be reduced to less than the tilt limit ~~except for physics tests,~~ or the following adjustments in setpoints and limit shall be made: M13 LATER

3.2.4 RA A.1.2.2

a. The Protection System Maximum Allowable Setpoints for Axial Power Imbalance in the COLR shall be reduced 2% in power for each 1% tilt in excess of the tilt limit.

3.2.4 RA A.1.2.3

b. The ~~control~~ regulating rod group ~~and APSR~~ withdrawal ~~limits~~ Setpoints given in the COLR shall be reduced 2% in power for each 1% tilt in excess of the tilt limit.

A4

3.2.4 RA A.1.2.4

c. The ~~reactor~~ Operational power imbalance setpoints given in the COLR shall be reduced 2% in power for each 1% tilt in excess of the tilt limit.

LA1 COLR

3.2.4 COND. D

<LATER> (3.1)

3. If quadrant power tilt is in excess of ~~25%~~ the maximum limit specified in the COLR ~~except for physics tests or diagnostic testing, the reactor will be placed in the hot shutdown condition~~. Diagnostic testing during power operation with a quadrant power tilt is permitted provided the thermal power allowable ~~for the reactor coolant pump combination~~ is restricted as stated in 3.5.2.4.1 above.

L3 LATER

≤ 20% RTP within 4 hours.

A1 M6

SR 3.2.4.1

3.2.4 Appl

4. Quadrant power tilt shall be monitored on a minimum frequency of once every ~~two hours~~ 7 days during ~~power operation~~ MODE 1 above ~~15% of rated power~~ 20% RTP.

L12 A9

<Add 3.2.4 RA A.2> M1

<Add 3.2.4 Condition B> M1

<Add 3.2.4 Condition C> M1

3.5.2.5 Control rod positions:

<LATER> (3.1)

1. ~~Techqical Specification 3.1.3.5 (safety rod withdrawal) does not prohibit the exercising of individual safety rods as required by Table 4.1-2 or apply to inoperable safety rod limits in Technical Specification 3.5.2.2.~~ LATER

3.2.1 LCO 2. Operating rod group overlap shall be 20% ±5 between two sequential groups, ~~except for physics tests.~~

LA1 COLR

LATER

<LATER> (3.1)

within the limits specified in the COLR

<Add SR 3.2.1.1> M5

<Add SR 3.2.1.2> M5

<Add 3.2.1 RA A.1 with Note> M10

<Add 3.2.2 RA A.1 and Note> M11
<Add 3.2.2 Condition B> M2
<Add SR 3.2.2.1> M5
<Add 3.2.3 RA A.1> M12

3.2.1
3.2.2
3.2.3

3.2.1 LCO Note
<Add 3.2.1 RA E.1> M7

<LATER> (3.1)
3.2.1 LCO

3. ~~Except for physics tests or~~ exercising control rods, the control rod position setpoints are specified in the CORE OPERATING LIMITS REPORT for 4, 3, AND 2 pump operation. M4

3.2.1 RA D.2.1 & D.2.2
RA A.2
RA B.1
RA C.1

If the applicable control rod position setpoints are exceeded, corrective measures shall be taken immediately to achieve an acceptable control rod position. Acceptable control rod positions shall be attained within 4 hours. L6

LATER

<LATER> (3.1)
3.2.2 LCO

4. ~~Except for physics tests or exercising axial power shaping rods (APSRs)~~, the limits for APSR position are specified in the CORE OPERATING LIMITS REPORT. A6

3.2.2 RA A.2

With the APSRs outside the specified limit provided in the CORE OPERATING LIMITS REPORT, corrective measures shall be taken immediately to achieve the correct position. Acceptable APSR positions shall be attained within ~~8~~ 24 hours. L4

3.5.2.6 ~~Reactor~~ AXIAL Power Imbalance Operating Limits A1

SR 3.2.3.1
3.2.3 APPL.

1. ~~Reactor~~ AXIAL power imbalance shall be monitored on a frequency not to exceed ~~2~~ 12 hours during ~~power operation~~ MODE 1 above 40% ~~rated power~~ RTP. L9 A1

LATER

<LATER> (3.1)
3.2.3 LCO

2. ~~Except for physics tests,~~ ~~reactor~~ AXIAL power imbalance shall be maintained within the ~~envelope defined by~~ limits specified in the ~~CORE OPERATING LIMITS REPORT~~ COLR. A1

3.2.3 RA A.2

3. If the ~~reactor~~ AXIAL power imbalance is not within the envelope defined by the CORE OPERATING LIMITS REPORT, corrective measures shall be taken to achieve an acceptable reactor power imbalance. L5

3.2.3 RA B.1

4. If an acceptable ~~reactor~~ AXIAL power imbalance is not achieved within ~~8~~ 24 hours, reactor power shall be reduced until ~~reactor power imbalance setpoints are met~~ THERMAL POWER is ≤ 40% RTP within the following 4 hours. A3 M8

~~3.5.2.7 The control rod drive patch panels shall be locked at all times with limited access to be authorized by the Superintendent.~~ LA1 SAR

Bases

The reactor power-imbalance envelope defined in the CORE OPERATING LIMITS REPORT is based on either LOCA analyses (which have defined the maximum linear heat rate (see CORE OPERATING LIMITS REPORT), such that the maximum cladding temperature will not exceed the Final Acceptance Criteria) or loss of forced reactor coolant flow analysis (such that the hot fuel rod does not experience a departure from nucleate boiling condition). Corrective measures will be taken immediately should the indicated quadrant power tilt, control rod position, or reactor power imbalance be outside their specified boundaries. Operation in a situation that would cause the Final Acceptance Criteria to be approached should a LOCA or loss of forced reactor coolant flow occur is highly improbable because all of the power distribution parameters (quadrant power tilt, rod position, and reactor power imbalance) must be at their limits while A2

The quadrant power tilt limits set forth in the CORE OPERATING LIMITS REPORT have been established within the thermal analysis design base using the definition of quadrant power tilt given in Technical Specifications, Section 1.6. These limits in conjunction with the control rod position setpoints in the CORE OPERATING LIMITS REPORT, ensure that design peak heat rate criteria are not exceeded during normal operation when including the effects of potential fuel densification.

The quadrant power tilt limits and reactor power imbalance setpoints in the CORE OPERATING LIMITS REPORT, apply when using the plant computer to monitor the limits. The 2-hour frequency for monitoring these quantities will provide adequate surveillance when the computer is out of service. Additional uncertainty is applied to the limits when other monitoring methods are used.

During the physics testing program, the high flux trip setpoints are administratively set as follows to ensure that an additional safety margin is provided.

Test Power	Trip Setpoint %
0	<5
15	50
40	50
50	60
75	85
>75	105.5

A2

REFERENCES

(1) FSAR, Section 3.2.2.1.2

(2) FSAR, Section 14.2.2.2

3.2.3
3.2.4

3.5.4 Incore Instrumentation

Applicability

Applies to the operability of the incore instrumentation system.

Objective

To specify the functional and operational requirements of the incore instrumentation system.

(A1)

Specification

Above 80 percent of operating power determined by the reactor coolant pump combination (Table 2.3-1) at least 23 individual incore detectors shall be operable to check gross core power distribution and to assist in the periodic calibration of the out-of-core detectors in regard to the core imbalance trip limits. The detectors shall be arranged as follows and may be a part of both basic arrangements.

(M18)

3.5.4.1 Axial Imbalance

A. Three detectors, one in each of three strings shall lie in th same axial plane with one plane in each axial core half.

B. The axial planes in each core half shall be symmetrical about the core mid-plane.

C. The detector shall not have radial symmetry.

3.5.4.2 Radial Tilt

A. Two sets of four detectors shall lie in each core half. Each set of four shall lie in the same axial plane. The two sets in the same core half may lie in the same axial plane.

B. Detectors in the same plane shall have quarter core radial symmetry.

(LA1) BASES

With the incore detector system inoperable, do not use the system for the above applicable monitoring function. The provisions of Specifications 3.0.3 are not applicable.

(M18)

Bases

A system of 52 incore flux detector assemblies with 7 detectors per assembly has been provided primarily for fuel management purposes. The system includes data display and record functions and is also used for out-of-core nuclear instrumentation calibration and for core power distribution verification.

A. The out-of-core nuclear instrumentation calibration includes:

1. Calibration of the split detectors at initial reactor startup during the power escalation program, and periodically thereafter.

(A2)

2. A comparison check with the incore instrumentation in the event one of the four out-of-core power range detector assemblies gives abnormal readings during operation.

3. Confirmation that the out-of-core axial power splits are as expected.

B. Core power distribution verification includes:

1. Measurement at low power initial reactor startup to check that power distribution is consistent with calculations.

2. Subsequent checks during operation to insure that power distribution is consistent with calculations.

3. Indication of power distribution in the event that abnormal situations occur during reactor operation.

C. The safety of unit operation at or below 80 percent of operating power[1] for the reactor coolant pump combinations without the core imbalance trip system has been determined by extensive 3-D calculations. This will be verified during the physics startup testing program.

D. The minimum requirement for 23 individual incore detectors is based on the following: A2

1. An adequate axial imbalance indication can be obtained with 9 individual detectors. Figure 3.5.4-1 shows a typical set of three detector strings with 3 detectors per string that will indicate an axial imbalance. The three detector strings are the center one, one from the inner ring of symmetrical strings and one from the outer ring of symmetrical strings.

2. Figure 3.5.4-2 shows a typical detection scheme which will indicate the radial power distribution with 16 individual detectors. The readings from 2 detectors in a radial quadrant at either plane can be compared with readings from the other quadrants to measure a radial flux tilt.

3. Figure 3.5.4-3 combines Figures 3.5.4-1 and 3.5.4-2 to illustrate a typical set of 23 individual detectors that can be specified as a minimum for axial imbalance determination and radial tilt indication, as well as for the determination of gross core power distributions. Startup testing will verify the adequacy of this set of detectors for the above functions.

E. At least 23 specified incore detectors will be operable to check power distribution above 80 percent power determined by reactor coolant pump combination. These incore detectors will be read out either on the computer or on a recorder. If a set of 23

3.2.3
3.2.4

~~detectors in specified locations is not operable, power will be decreased to or below 80 percent for the operating reactor coolant pump combination.~~

~~REFERENCE~~

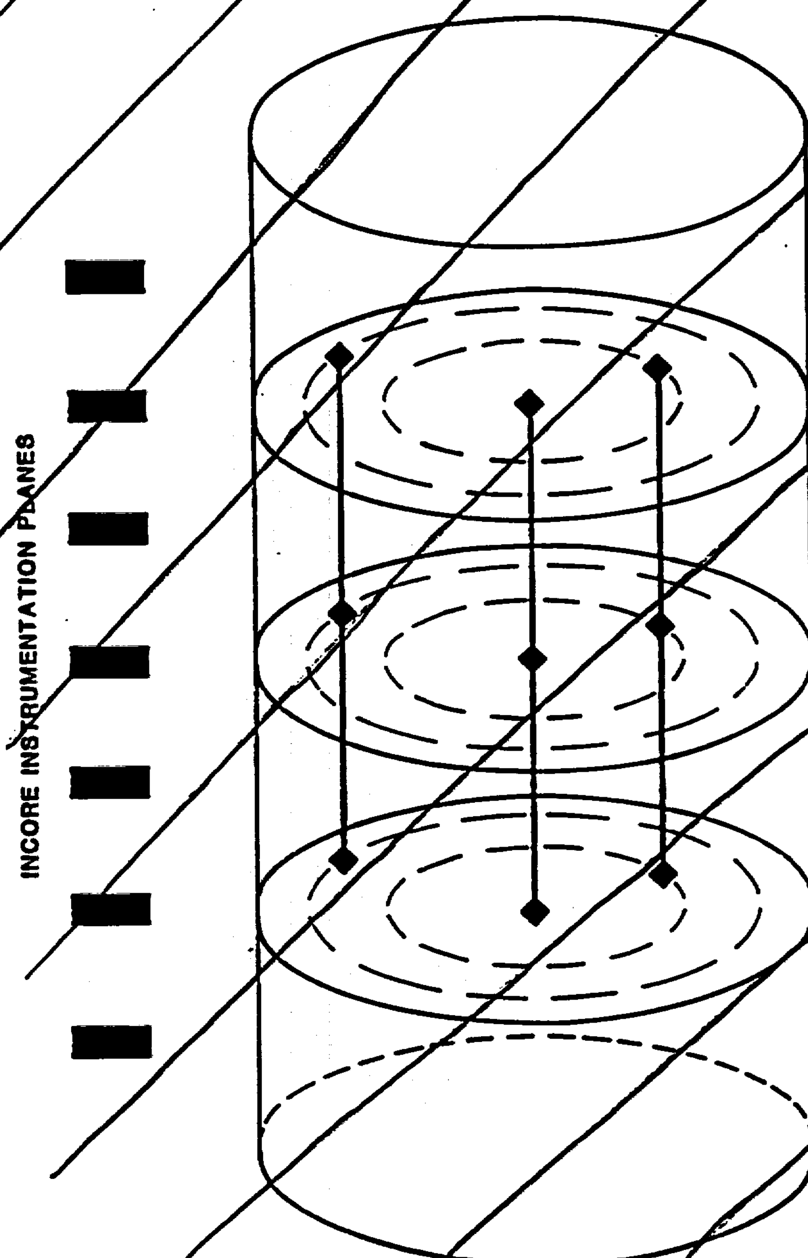
~~(1) FSAR, Section 4.1.1.3~~

(A2)

3.2.3
3.2.4

INCORE INSTRUMENTATION SPECIFICATION
AXIAL IMBALANCE INDICATION

Figure No. 3.5.4-1

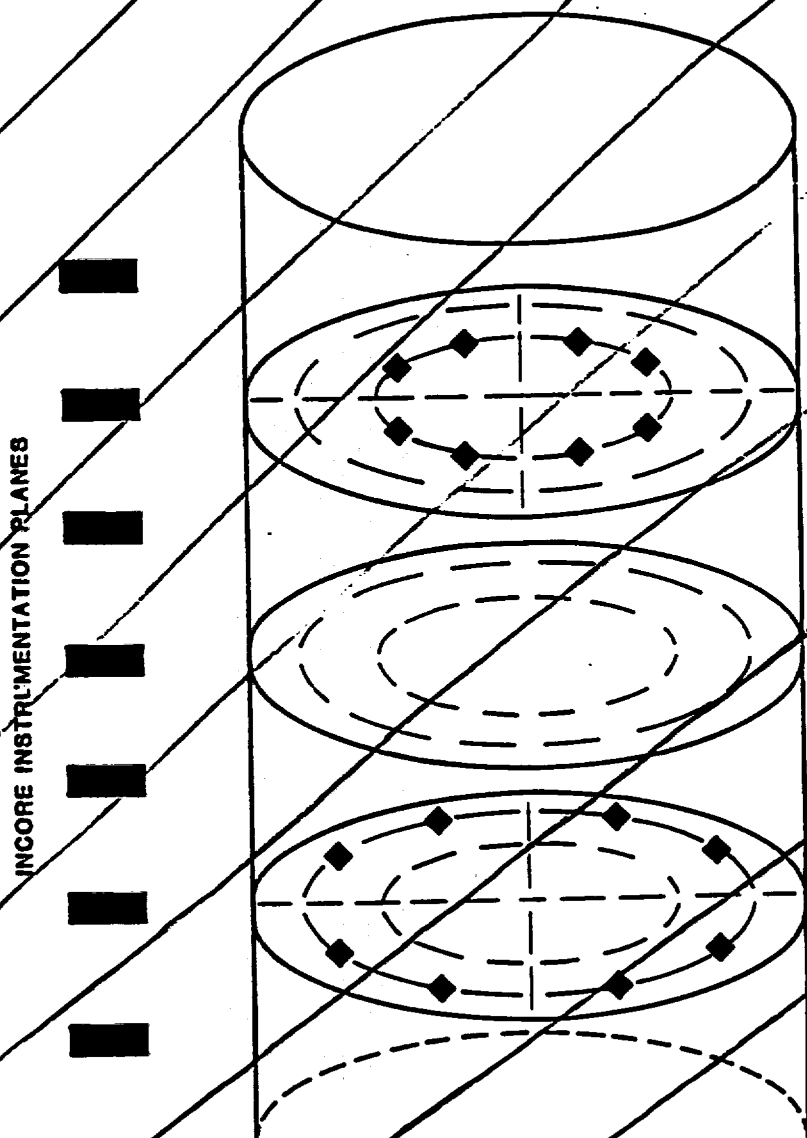


A2

3.2.3
3.2.4

INCORE INSTRUMENTATION SPECIFICAITON
RADIAL FLUX TILT INDICATION

Figure No. 3.5.4-2

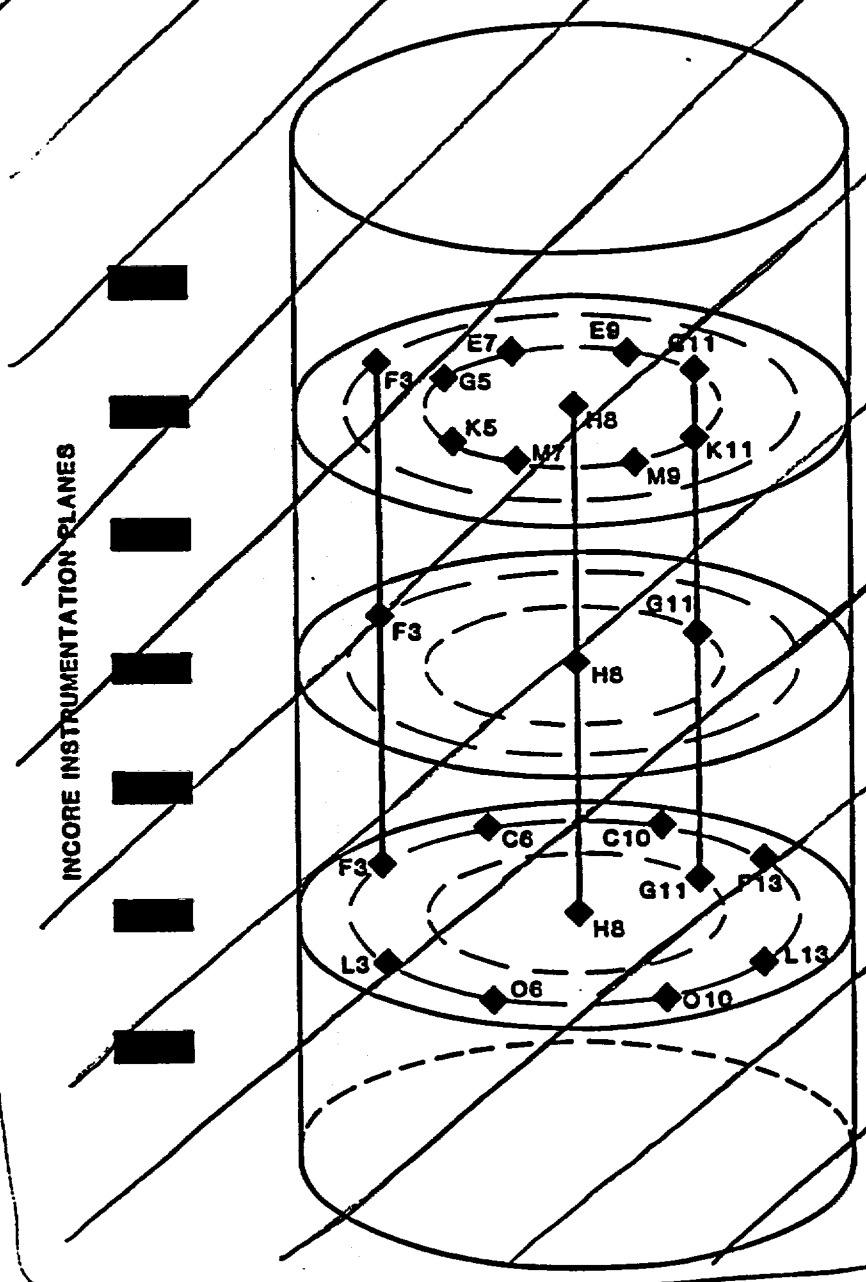


(A2)

3.2.3
3.2.4

INCORE INSTRUMENTATION SPECIFICATION

Figure No. 3.5.4-3



A2

OPERATIONAL SAFETY ITEMS (continued)

<LATER> (3.3A, 3.3B, 3.3C, 3.3D)

4.1 (Continued)

b. Equipment and sampling test shall be performed as detailed in Table 4.1-2 and 4.1-3.

c. Discrepancies noted during surveillance testing will be corrected and recorded.

LATER

M17

3.2.5 LCO

SR 3.2.5.1

d. A power distribution as specified by applicable LCO(s) ~~map shall be made to verify the expected~~ power distribution ~~at periodic intervals at least every 10 effective full power days~~ using the incore instrumentation detector system.

L11

BASES

4.0.1 through 4.0.5 Establish the general requirements applicable to Surveillance Requirements. These requirements are based on the Surveillance Requirements stated in the Code of Federal Regulations, 10CFR 50.36(c)(3):

"Surveillance Requirements are requirements relating to test, calibration, or inspection to ensure that the necessary quality of systems and components is maintained, that facility operation will be within safety limits, and that the limiting conditions of operation will be met."

<LATER> (3.0)

4.0.1 Establishes the requirement that surveillances must be performed during the operational modes or other conditions for which the requirements of the Limiting Conditions for Operation apply unless otherwise stated in an individual Surveillance Requirement. The purpose of this specification is to ensure that surveillances are performed to verify the operational status of systems and components and that parameters are within specified limits to ensure safe operation of the facility when the plant is in a mode or other specified condition for which the associated Limiting Conditions for Operation are applicable. Surveillance Requirements do not have to be performed when the facility is in an operational mode for which the requirements of the associated Limiting Condition for Operation do not apply unless otherwise specified.

LATER

<Add 3.2.5 Condition A> M9

<Add 3.2.5 Condition B> M9

<Add SR 3.2.5.1 Note> L11

<Add 3.2.5 Applicability> A7

Table 4.1-1 (Cont.)

	Channel Description	Check	Test	Calibrate	Remarks	
<LATER> (3.5)	36. Boric Acid Addition Tank					LATER
	a. Level Channel	NA	NA	R		
	b. Temperature Channel	M	NA	R		
<LATER> (3.3D)	37. Degraded Voltage Monitoring	M	R	R		LATER
	38. Sodium Hydroxide Tank Level Indicator	NA	NA	R		R TRM
	39. Incore Neutron Detectors	M(1)	NA	NA	(1) Check Functioning	LA1 TRM
<LATER> (3.3D)	40. Emergency Plant Radiation Instruments	M(1)	NA	R	(1) Battery Check	LATER
<LATER> (3.3A)	41. Reactor Trip Upon Turbine Trip Circuitry	M	PC	R		LATER
	42. Deleted					A1

3.2

NO SIGNIFICANT HAZARDS CONSIDERATIONS STATEMENTS

ANO-1 ITS SECTION 3.2: Power Distribution Limits

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:

NSHC 3.2 L1

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

An extension of the Completion Time for a Required Action does not result in any hardware changes. The Completion Time for performance 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 limits the impact on probability). Also, an extension of the Completion Time provides additional opportunity to restore compliance with the requirements and avoid the increased potential for a transient during the shutdown process. Further, the Completion Time for performance of Required Actions does not significantly increase the consequences of an accident because a change in the Completion Time does not change the assumed response of the equipment in performing its specified mitigatory functions, or change 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?

Prompt and appropriate Required Actions have been determined based on the safety analysis functions to be maintained. The proposed Completion Time has been determined appropriate based on a combination of the time required to perform the action, the relative importance of the function or parameter to be restored, and engineering judgment. Therefore, the short extension of the Completion Time interval does not involve a significant reduction in the margin of safety.

NSHC 3.2 L2 Not Used.

NO SIGNIFICANT HAZARDS CONSIDERATIONS STATEMENTS

NSHC 3.2 L3

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

QUADRANT POWER TILT (QPT) limits are used to control core power distribution to within the initial assumptions of the accident analysis. However, the QPT is not considered as an initiator of any previously analyzed accident. As such the proposed change in Applicability of the QPT limit requirements will not significantly increase the probability of any accident previously evaluated. The proposed change allows for continued operation with no QPT limits below 20% RTP since the resulting maximum linear heat rate (LHR) is not high enough to cause violation of the LOCA LHR limit or the initial condition DNB allowable peaking limit during accidents initiated at this low power level. 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 prompt restoration of compliance with the limiting condition for operation, or prompt and appropriate compensatory actions are taken, during the conditions which may result in violation of core power distribution limits. 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 Applicability and Required Actions have been determined based on the safety analysis functions and core parameters to be maintained. The proposed Applicability has been determined appropriate based on the lack of need to monitor and maintain the core power distribution at the low power levels. Therefore, the change of the Applicability and Required Actions does not involve a significant reduction in the margin of safety.

NO SIGNIFICANT HAZARDS CONSIDERATIONS STATEMENTS

NSHC 3.2 L4

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

An extension of the Completion Time for a Required Action does not result in any hardware changes. The Completion Time 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 limits the impact on probability). An extension of the Completion Time provides additional opportunity to restore compliance with the requirements and avoid the increased potential for a transient during the shutdown process. The Completion Time for performance of Required Actions does not significantly increase the consequences of an accident because a change in the Completion Time does not change the assumed response of the equipment in performing its specified mitigatory functions, or change the response of the core parameters to assumed scenarios from that considered during the original Completion Time. In addition, the extension in Completion Time is dependent upon the performance of a new Required Action that provides verification of local linear heat rates within the core. This verification preserves the initial conditions of the accident analysis regarding core power distribution.

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?

Appropriate Required Actions have been determined based on the safety analysis functions to be maintained. The proposed Completion Time has been determined appropriate based on a combination of the importance of the function or parameter to be restored and engineering judgment. In addition, a new Required Action has been adopted which provides verification of local core linear heat rates while operating within the extension of the Completion Time. Therefore, the extension of the Completion Time interval does not involve a significant reduction in the margin of safety.

NO SIGNIFICANT HAZARDS CONSIDERATIONS STATEMENTS

NSHC 3.2 L5

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

An extension of the Completion Time for a Required Action does not result in any hardware changes. The Completion Time 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 limits the impact on probability). An extension of the Completion Time provides additional opportunity to restore compliance with the requirements and avoid the increased potential for a transient during the shutdown process. The Completion Time for performance of Required Actions does not significantly increase the consequences of an accident because a change in the Completion Time does not change the assumed response of the equipment in performing its specified mitigatory functions, or change the response of the core parameters to assumed scenarios, from that considered during the original Completion Time. In addition, the extension in Completion Time is dependent upon the performance of a new Required Action that provides verification of local linear heat rates within the core. This verification preserves the initial conditions of the accident analysis regarding core power distribution.

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?**

Appropriate Required Actions have been determined based on the safety analysis functions to be maintained. The proposed Completion Time has been determined appropriate based on a combination of the importance of the function or parameter to be restored and engineering judgment. In addition, a new Required Action has been adopted which provides verification of local core linear heat rates while operating within the extension of the Completion Time. Therefore, the extension of the Completion Time interval does not involve a significant reduction in the margin of safety.

NSHC 3.2 L6

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

An extension of the Completion Times for the Required Actions do not result in any hardware changes. The extension of the Completion Times 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). An extension of the Completion Times provides additional opportunity to restore compliance with the requirements and avoid the increased potential for a transient during the shutdown process. The Completion Times for performance of the Required Actions do not significantly increase the consequences of an accident because a change in the Completion Times does not change the assumed response of the equipment in performing its specified mitigatory functions, or change the response of the core parameters to assumed scenarios, from that considered during the original Completion Times. For example, the extension of one of the Completion Times is dependent upon the performance of a new Required Action that provides verification of local linear heat rates within the core. This verification preserves the initial conditions of the accident analysis regarding core power distribution. An extension of another Completion Time is premised on the initiation of boration to re-establish the required SHUTDOWN MARGIN while simultaneously reducing THERMAL POWER to preserve the ejected rod worth reactivity worth assumptions. The third and fourth extensions in the Completion Time establish a realistic opportunity to perform the Required Action without unduly challenging the ability of the operator to control the unit. All of these function to implement appropriate Required Actions that provide mitigatory measures to the out-of-LCO-compliance condition. Therefore, the extension of the Completion Times does not significantly increase the consequences of an evaluated accident.

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

The proposed changes do not 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 changes 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?

Prompt and appropriate Required Actions have been determined based on the safety analysis functions to be maintained. The proposed Completion Times have been determined appropriate based on a combination of the time required to perform the action, the relative importance of the function or parameter to be restored, and engineering judgment. Therefore, the extension of the Completion Time intervals do not involve a significant reduction in the margin of safety.

NO SIGNIFICANT HAZARDS CONSIDERATIONS STATEMENTS

NSHC 3.2 L7 Not Used.

NSHC 3.2 L8

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

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 (and therefore any initiation scenarios are not changed) and the proposed Completion Time extension is short (and therefore limits the impact on probability). Also, an extension of the Completion Time provides additional opportunity to restore compliance with the requirements and avoid the increased potential for a transient during the shutdown process. Further, the Completion Time for performance of Required Actions does not significantly increase the consequences of an accident because a change in the Completion Time does not change the assumed response of the equipment in performing its specified mitigation functions, or change 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?**

Prompt and appropriate Required Actions have been determined based on the safety analysis functions to be maintained. The proposed Completion Time has been determined appropriate based on a combination of the time required to perform the action, the relative importance of the function or parameter to be restored, and engineering judgment. Therefore, the short extension of the Completion Time interval does not involve a significant reduction in the margin of safety.

NSHC 3.2 L9

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

An extension of the Surveillance Frequency does not result in any hardware changes. The Frequency 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). Further, the Frequency for performance of a Surveillance does not significantly increase the consequences of an accident because a change in the Frequency does not change the assumed response of the equipment in performing its specified mitigatory functions, or change the response of the core parameters to assumed scenarios, from that considered during the original Frequency.

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?**

Prompt and appropriate Required Actions have been determined based on the safety analysis functions to be maintained. The proposed Frequency has been determined appropriate based on a combination of the time required to perform the surveillance, the relative importance of the function or parameter to be verified, the causes or events that would induce a change in the monitored parameter, available instrumentation for recognition of events that might cause a change in the monitored parameter, and engineering judgment. Therefore, the extension of the Frequency interval does not involve a significant reduction in the margin of safety.

NO SIGNIFICANT HAZARDS CONSIDERATIONS STATEMENTS

NSHC 3.2 L10

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

The 24 hour delay of the CTS requirement to initiate a mandatory power reduction based on indication of a QUADRANT POWER TILT (QPT) above its steady state limit does not result in any hardware changes. The delay of the mandatory power reduction requirement 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). The delay of the mandatory power reduction requirement provides additional opportunity to restore compliance with the LCO requirements and avoid the increased potential for a transient during the power reduction process. The delay of the mandatory power reduction also minimizes power redistribution phenomena associated with the power reduction which may exacerbate the QPT. The delay of the mandatory power reduction does not significantly increase the consequences of an accident because the core power distribution continues to be verified as acceptable through the performance of ITS SR 3.2.5.1. This Surveillance verifies that core power distribution remains within the ECCS accident analysis assumptions and the DNBR loss of flow analyses. If this Surveillance indicates that an unacceptable power distribution exists, then LCO 3.2.5 Required Actions exist that require a prompt reduction in core THERMAL POWER.

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?**

Appropriate Required Actions have been determined based on the safety analysis functions to be maintained. The proposed Required Actions for QPT have been determined appropriate based on a combination of the importance of the function or parameter to be restored and engineering judgment. In addition, a new Required Action has been adopted which provides verification of local core linear heat rates while operating with a QPT in excess of its steady state limit. Therefore, the delay of the mandatory CTS power reduction does not involve a significant reduction in the margin of safety.

NSHC 3.2 L11

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

The establishment of a conditional Frequency for the performance of SR 3.2.5.1 vice the CTS 4.1.d requirement that the SR be performed every 10 EFPD does not constitute a hardware change or other physical alteration of the plant. The Frequency for performance of SR 3.2.5.1 in the ITS will be when required by LCO 3.1.8 and the Required Actions of LCO 3.1.4, LCO 3.2.1, LCO 3.2.2, LCO 3.2.3, and LCO 3.2.4. The deletion of the fixed CTS SR Frequency 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). The consequences of a previously evaluated accident will not be significantly increased because the actual core power distribution will be verified within its limits by the performance of SR 3.2.5.1 when required by the appropriate LCO or Required Action, given above. The ITS will key performance of the SR on operational conditions that might lead to a challenge of the core local linear heat rates such that the ECCS or DNBR analyses are not satisfied.

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?

Appropriate Required Actions have been determined based on the safety analysis functions to be maintained. The proposed SR Frequency has been determined appropriate based on a combination of the importance of the function or parameter to be restored and engineering judgment. In addition, the new SR Frequency provides verification of local core linear heat rates while operating within the Actions of the various specifications listed above. Therefore, the change in SR Frequency does not involve a significant reduction in the margin of safety.

NSHC 3.2 L12

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

The extension of the Frequency for the performance of a Surveillance does not constitute a hardware change or other physical alteration of the plant. The extension of the SR Frequency 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). The SR Frequency is based on the relatively slow changing nature of the QPT during steady state conditions. During transient conditions, other indication is available in the control room to alert the operator to plant conditions that may result in QPT exceeding its limit. While operating within the Actions of other ITS LCOs due to events likely to induce power redistribution effects, the Required Actions directing performance of SR 3.2.5.1 are more than adequate in verifying an acceptable power distribution within the core. Thus, the consequences of a previously evaluated accident will not be significantly increased because the actual core power distribution will be verified within its limits by the performance of SR 3.2.5.1 when required by the appropriate LCO or Required Action.

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?

Appropriate Required Actions have been determined based on the safety analysis functions to be maintained. The proposed SR Frequency has been determined appropriate based on a combination of the importance of the function or parameter to be restored and engineering judgment. In addition, the new SR Frequency acknowledges the slow nature of changes in QPT during steady state conditions. Appropriate Required Actions provide verification of local core linear heat rates while operating within the Actions of the various specifications referenced above. Therefore, the change in SR Frequency does not involve a significant reduction in the margin of safety.

1. Several changes were made to the ACTIONS established for ITS 3.2.1. These changes include: 1) a minor editorial change, and 2) the addition of a new Condition C which is necessary because the NUREG-1430 ACTIONS do not appropriately address the Required Action and associated Completion Time for regulating rod groups that are not positioned in accordance with the required sequence or overlap requirements. The following paragraphs describe these changes in detail.

   1) Editorial changes were made to reflect consistent titles for the regions on the regulating rod group insertion limits figures contained in the COLR. In ITS 3.2.1 Condition A and Condition D, the word "operational" was changed to "operation." In ITS 3.2.1 Required Action D.2.1, the word "operating" was changed to "operation." These changes establish titles consistent with the NUREG-1430 3.2.1 Bases.

   2) NUREG 3.2.1 Condition A is entered when the regulating rods are inserted into the restricted operation region, or sequence or overlap requirements are not met. However, NUREG Required Actions A.1 and A.2 do not address the group(s) out of sequence or the group overlap requirements not met condition. Therefore, ITS Condition C is added (as in NRC approved TSTF-345) so that a specific Required Action is provided to restore compliance with the LCO should the regulating rod group sequence or overlap requirements not be met. ITS Required Action C.1 requires that the regulating rod groups be restored to within the limits with a Completion Time of 4 hours. Four hours was chosen based on CTS 3.5.2.5.3, which also provides 4 hours (Note that TSTF-345 provided 2 hours for this Required Action). This change is consistent with generic change TSTF-345, as modified to match CTS.

   3) The aforementioned changes require that NUREG 3.2.1 Condition C be revised to represent ITS 3.2.1 Condition D and that NUREG 3.2.1 Condition D be revised to represent ITS 3.2.1 Condition E. Further, the inclusion of ITS 3.2.1 Condition C (and re-designation of NUREG 3.2.1 Condition C as ITS 3.2.1 Condition D) requires that ITS 3.2.1 Condition E read "Required Actions and associated Completion Times of Conditions C or D not met" to ensure that appropriate actions are provided should Conditions C or D not be satisfied. The appropriate action is to remove the unit from the LCO Applicability, which is accomplished by having the unit proceed to MODE 3 with a Completion Time of 6 hours.

   4) TSTF-160, Rev 1, was incorporated which reflects that ITS 3.2.1 Required Action A.1, performance of SR 3.2.5.1, is only required when THERMAL POWER is greater than 20% RTP. This Note provides an Applicability for the Required Action which is consistent with the ITS LCO 3.2.5 Applicability.

The Bases for LCO 3.2.1 were similarly marked to reflect these changes.

2. NUREG-1430 LCO 3.2.1 Required Action C.1 was modified to reflect generic change TSTF-009, Rev 1.

3. NUREG-1430 3.2.1 incorporates TSTF-110, Rev 2.

   The Bases for ITS 3.2.1 were similarly marked to reflect the modification in SR Frequency requirements. In addition to the material deleted by the TSTF, sentences containing reference to the alarm function were deleted for consistency and clarification.

4. NUREG-1430 SR 3.2.1.3 requires that the SDM be verified to be within limits within 4 hours prior to achieving criticality. This is redundant to the requirements of LCO 3.1.1 which require that the SDM be within the same limits while in MODES 3, 4 and 5. SR 3.1.1.1 requires that the SDM verification be performed every 24 hours while in these MODES. Additionally, the regulating rod group position limits and safety rod insertion limits have been determined, by verified methodology, to maintain the required SDM in MODES 1 and 2. These rod group position limits are required to be met prior to entry into the Applicability of LCO 3.2.1, i.e., prior to entry into MODE 2, by SR 3.0.4. Therefore, SDM is periodically verified by calculation while in MODES 3, 4, and 5, and verified by rod position limits again prior to entry into MODE 2. An additional verification within 4 hours prior to criticality is redundant to these required verifications and unnecessary. Also, the ANO-1 CTS does not require the performance of a surveillance equivalent to NUREG-1430 SR 3.2.1.3. Therefore, the NUREG-1430 Surveillance is not adopted in the ITS.

   The Bases for ITS 3.2.1 were similarly marked to reflect that NUREG SR 3.2.1.3 was not adopted in the ITS.

5. Bases of LCO 3.2.2 - Potentially misleading material was removed regarding the APSRs. The APSRs are designed not to insert into the reactor on a reactor trip (scram). Because they do not insert, they were never credited in the analyses as contributing to the rate of reactivity addition, net reactivity addition or the SDM.

6. NUREG-1430 3.2.2 Required Action A.1 was modified by a Note to reflect that this Required Action is only required when THERMAL POWER is greater than 20% RTP. This Note provides a Required Action which is consistent with the ITS LCO 3.2.5 Applicability. This change is consistent with TSTF-160, Rev 1.

7. ITS Completion Times for 3.2.3 RA B.1 (NUREG 3.2.3 RA B.1), 3.2.4 RA C.1 (NUREG 3.2.4 RA E.1), 3.2.4 RA D.1 (NUREG 3.2.4 RA F.1), and 3.2.5 RA B.1 (NUREG 3.2.5 RA C.1) were revised to specify 4 hours. The 4 hour Completion Time provides a more reasonable time frame for performing the required power reduction to less than or equal to 20% RTP (40% RTP for ITS 3.2.3) from full power conditions (RTP). The NUREG 2 hour Completion Time would have required the operators to violate the established normal, non-emergency, maneuvering rate of ≤30% per hour and unnecessarily challenged the operator's ability to control the unit with the potential introduction of a unit transient. Although the CTS established comparable Required Actions, it did not establish a Completion Time for those actions. Based on the foregoing discussion, the ITS 4 hour Completion Time is established which results in a prompt compensatory action while adhering to the unit's operating procedures.

   The Bases were similarly marked to reflect these changes.

8. NUREG-1430 3.2.3 incorporates TSTF-110, Rev 2.

   The Bases for ITS 3.2.3 were similarly marked to indicate this change. In addition to the material deleted by the TSTF, sentences containing reference to the alarm function were deleted for consistency and clarification.

9. NUREG-1430 LCO 3.2.4 is premised on the existence of a steady state limit, transient limit and a maximum limit for QUADRANT POWER TILT (QPT). The ANO-1 CTS and the ANO-1 COLR do not establish a transient limit for QPT. Further, the ANO-1 CTS does not provide any differentiation between the possible causes of an excessive QPT (i.e., QPT due to CONTROL ROD misalignment versus other potential causes) in specifying the required actions. Therefore, reference to a transient limit was removed from ITS LCO 3.2.4. Consequently, NUREG Condition B (which addresses the situation where QPT may exceed the transient limit but still be less then the maximum limit), and Condition D (which addresses the situation where QPT may exceed the transient limit but still be less then the maximum limit due to causes other than the misalignment of either CONTROL ROD(S) or APSR(S)) are not adopted in the ITS. NUREG Condition E has been modified in the ITS to provide the Required Action should the Required Action and associated Completion Time for Condition B not be met. These changes retain the intent that THERMAL POWER be reduced and that the ACTIONS lead to removal of the unit from the LCO Applicability if compliance is not restored. These changes maintain requirements consistent with current license basis.

   The Bases for LCO 3.2.4 were similarly marked to reflect these changes.

10. 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.

    For ITS LCO 3.2.2, the 10 CFR 50.36 Criterion statement was modified to preserve consistency with the ANO-1 license basis. Specifically, ANO-1 safety analyses, upon which ITS LCO 3.2.2 was based, were performed with the reactor critical. Thus, the Criterion statement was revised to specify that the LCO parameter satisfies Criterion 2 of 10 CFR 50.36 when in MODES 1 and 2 while critical. When in MODE 2 with the reactor subcritical, the LCO parameter satisfies Criterion 4 of 10 CFR 50.36. This change is consistent with current license basis and 10 CFR 50.36.

11. Bases - Throughout Section 3.2 Bases, numerous references to "limits" have been changed to "setpoints." In a few instances, references to "setpoints" have been changed to specify "limits." The COLR defines the regulating group insertion setpoints, group overlap limits, the APSR insertions setpoints, the AXIAL POWER IMBALANCE setpoints, and the QUADRANT POWER TILT limits and setpoints. These values are established in accordance with the NRC approved reload methodology established by BAW-10179P-A, "Safety Criteria and Methodology for Acceptable Cycle Reload Analyses," February 1991. This change is consistent with current license basis.

12. NUREG 3.2.3 and 3.2.4 Bases - In the Applicability section of the Bases for ITS 3.2.3 and 3.2.4, statements were added that the acceptability of continued operation with a significant AXIAL POWER IMBALANCE or QUADRANT POWER TILT is based on engineering judgment. ANO-1 has not performed analysis to substantiate statements made in the NUREG Bases because the accident initial conditions discussed are inconsistent with the unit's license basis accident initial conditions.

13. NUREG 3.2.2 Bases – Incorporates TSTF-125, Rev. 1.

14. Bases - In the Applicable Safety Analysis section of the Bases for LCO 3.2.4, reference was made to ANSI N18.2-1973 as establishing the requirement that the peak cladding temperature not exceed 2200°F. All similar statements in the NUREG-1430 reference 10 CFR 50.46 as the basis for this requirement. Because the statements used in all of the Bases of Section 3.2 cite 10 CFR 50.46 as the reference, the Bases for ITS LCO 3.2.4 will be similarly changed to reference 10 CFR 50.46.

15. CTS 3.5.2.4.2 establishes that the overpower protection, during periods when QPT is greater than its limit, is provided by an adjustment in the nuclear overpower based on Reactor Coolant System flow and AXIAL POWER IMBALANCE trip function. The CTS does not impose a requirement that the nuclear overpower trip setpoint be reduced. Therefore, ITS 3.2.4 Required Action A.1.2.2 will specify the current license requirement to implement a reduction in the nuclear overpower based on Reactor Coolant System flow and AXIAL POWER IMBALANCE trip setpoint. These changes maintain requirements consistent with current license basis.

    The Bases for 3.2.4 were similarly marked to reflect this change.

16. NUREG 3.2.1 - Incorporates TSTF-216.

17. ITS 3.2.4 Required Action A.1.2.2 Completion Time is modified to include a second conditional Completion Time of 10 hours after the last performance of SR 3.2.5.1. This second Completion Time is necessary to establish a Completion Time dependent on the failure to perform SR 3.2.5.1 similar to that established for NUREG RA A.1.2.1. As written in the NUREG, RA A.1.2.2 would have to be completed within 10 hours of entry in Condition A any time the A.1.2.X alternative Required Actions were chosen. However, if NUREG RA A.1.1 (SR 3.2.5.1) was being performed for an extended period of time, assume 10 hours, and then stopped, then RA A.1.2.2 could not be completed within its required Completion Time. The operators would immediately have to enter NUREG Condition B due to the failure to complete the Required Actions and associated Completion Times of Condition A within the required time frames. This change is consistent with NUREG-1430 Section 1.3 guidance on Completion Times as well as the NUREG Writer's Guide.

    The Bases for ITS 3.2.4 were similarly marked to reflect this change.

18. The ITS was marked to indicate the addition of Required Actions A.1.2.3 and A.1.2.4 consistent with the current license basis (CTS 3.5.2.4.2.b and 3.5.2.4.2.c). ITS 3.2.4 Required Action A.1.2.3 requires modification of the allowed regulating group insertion setpoints given in the COLR to help ensure that core thermal limits remain acceptable for continued operation. ITS 3.2.4 Required Action A.1.2.4 requires modification of the Operational Power Imbalance setpoints as given in the COLR that similarly helps ensure that core thermal limits remain acceptable for continued operation. This is consistent with current license basis.

    The Completion Times for these Required Actions are stated as 10 hours or 10 hours after last performance of SR 3.2.5.1 because they constitute alternative actions to RA A.1.1 (SR 3.2.5.1). This Completion Time is consistent with NUREG-1430 Section 1.3 guidance on Completion Times as well as the NUREG Writer's Guide.

    The Bases for ITS 3.2.4 were similarly marked to reflect this change.

19. Not used.

20. Not used.

21. Bases of various Actions were corrected to accurately describe the Condition. Wording similar to that of the Condition was inserted in each case to remove possibly misleading or inaccurate wording from the Bases for these Actions. These changes do not change the intent or usage of these Actions but serve only as clarification.

22. NUREG 3.2.4 - Incorporates TSTF-110, Rev 2.

    The Bases for ITS 3.2.4 were similarly marked to reflect these changes. In addition to the material deleted by the TSTF, sentences containing reference to the alarm function were deleted for consistency and clarification.

23. NUREG 3.2.5 incorporates TSTF-160, Rev 1. The Applicability was modified to specify MODE 1 with THERMAL POWER > 20% RTP. This establishes an Applicability that coincides with the lower operable range for the Incore Detector system. This change in Applicability is necessary because the Incore Detector system is used to satisfy SR 3.2.5.1. Further, below 20% RTP, the probability of experiencing an event that could result in excessive linear heat rates or result in DNB is small. This establishes the LCO 3.2.5 Applicability as one that is consistent with the Applicability of ITS LCO 3.2.4, QUADRANT POWER TILT.

    ITS Required Action B.1 (NUREG-1430 Required Action C.1) was modified to maintain consistency between this Required Action and the new Applicability of this LCO.

    The Bases for LCO 3.2.5 were similarly marked to reflect these changes.

24. Not used.

25. Not used.

26. Bases - At multiple locations in the Bases for Section 3.2, paragraphs stating that the actual alarm setpoints may be more conservative than the maximum allowable setpoints were deleted to remove any possible misinterpretation that this was not an acceptable practice in all other situations. Generally, alarm setpoints are conservative with respect to the allowable setpoint. The presence of this paragraph implies that this is not an acceptable practice in other circumstances. Further, this paragraph implies that this monitoring function is performed by the plant computer and is credited within the ITS; when in fact, the plant computer monitoring functions are not credited as performing or satisfying the requirements of these surveillances.

27. The Bases of Specifications 3.2.3 and 3.2.4 were revised to indicate the following changes:
    1) Bases LCO discussion which was more appropriate for the Bases Action section and which essentially duplicated information in the Bases Action section was removed.
    2) Bases discussion of PHYSICS TEST exceptions was removed from 3.2.3 Applicability Bases section. This change was made to maintain consistency between the Bases of this Specification and the Bases of other Specifications which are the subject of PHYSICS TEST exceptions.
    3) Bases Applicability for ITS 3.2.4 was revised to remove a statement that lacks an analytical justification.

28. Not used.

29. Present APSR position limitations given in the COLR specify that the APSRs are to be positioned as necessary for the control of AXIAL POWER IMBALANCE prior to 483 ±10 EFPD [Cycle 15 specific value]. After this burnup value, the APSRs shall be fully withdrawn and not reinserted. No specific limitation exists to prevent their complete withdrawal prior to this burnup value, although this would not be an expected occurrence. Therefore, the Bases for LCO 3.2.2 were modified to state that the APSRs are positioned in accordance with control rod operating guidelines provided by reactor engineering. Further, because there are no specific limits associated with APSR positioning, the discussion of error adjusted setpoints in the bases is not pertinent. Hence, its deletion.

30. ITS SR 3.2.2.1 - ANO-1 does not credit the computer generated alarm function as satisfying this surveillance requirement. The 12 hour Frequency for verification of APSR position is retained because of the infrequent usage of the APSRs and the fact that devices must be manually positioned by the operator. This change preserves the current license basis.

31. CTS 4.1.d establishes a requirement that "a power distribution map shall be made to verify the expected power distribution at periodic intervals at least every 10 effective full power days using the incore instrumentation detector system." The intent of this requirement is to ensure steady state power distributions are consistent with design and operation assumptions.

    ANO-1 presently verifies the acceptability of the core power distribution by determining that the linear heat rate (LHR) is within the limits established for various core elevations and fuel batch designs. Further, ANO-1 presently verifies that an extrapolated DNBR value at the protective system actuation point is within its limits. By performing these two verifications, core power distribution is demonstrated to satisfy LHR limitations based on the ECCS (LOCA) analyses as well as the limitations for the limiting DNBR transient (loss of forced reactor coolant flow). The current methodology does not specifically refer to or perform a verification of power peaking factors. However, the current methodology does result in a verification of acceptable power distribution equivalent to the requirements for verification of the power peaking factors referenced in NUREG LCO 3.2.5. Therefore, the ANO-1 current methodology will be retained. NUREG LCO 3.2.5 was renamed "Power Peaking" in the ITS to reflect the current methodology.

The wording in ITS 3.2.5 LCO was modified to reflect that LHR is the parameter that is required to be verified. In addition, ITS Condition A, Required Action A.1 and SR 3.2.5.1 were modified to indicate that LHR has been substituted for the NUREG-1430 peaking factor, $F_Q(Z)$.

NUREG-1430 3.2.5 Required Action A.1 was modified to direct a reduction in THERMAL POWER to restore the LHR to within the limit. A Completion Time of 2 hours was specified. No other Required Actions are specified because the reduction in THERMAL POWER will continue until the LHR is within its limit. The 2 hour Completion Time ensures that prompt corrective measures are initiated while providing the operator with the ability to implement a power reduction in an orderly and controlled manner in the presence of a condition that has resulted in the adverse power distribution. The CTS does not establish any specific Required Actions or Completion Times for this LCO.

NUREG-1430 3.2.5 Condition B was deleted in its entirety because ITS Condition A provides the necessary corrective action when the LHR is not within its limits. NUREG-1430 3.2.5 Condition C was editorially relabeled as ITS Condition B containing Required Action B.1. This change preserves the format of NUREG-1430 and the actions that result in the unit exiting the Applicability if the LHR cannot be restored to within its limits. The 4 hour Completion Time is based on the need to take prompt corrective actions to reduce the core THERMAL POWER level when operating with LHR greater than its limits while adhering to unit operating procedures governing normal, non-emergency, power maneuvering rates of ≤30% per hour. The 4 hour Completion Time provides a reasonable period of time for the reactor operator to reduce the THERMAL POWER of the unit during a situation in which LHR has been made to exceed its limits. This Completion Time also recognizes the low probability of an accident occurring coincident with the LHR not within its limits. The adoption of the 4 hour Completion Time in the ITS will be more restrictive because the CTS did not previously establish a Completion Time for this required power reduction.

SR 3.2.5.1 was modified to specify that LHR is the parameter being verified consistent with the above discussion.

The Bases were rewritten to reflect that the LHR is the limiting parameter and that operational constraints are based on this parameter. Through a variety of correlations, the LHR may be expressed in terms of DNBR, margin to DNB or as power peaking factors. By establishing the LHR as the operational parameter, all confusion regarding which power peaking factor is limiting and how to adjust the power peaking factor for operation at THERMAL POWER levels less than 100% RTP has been eliminated.

At numerous locations through the Section 3.2 Bases of the ITS, reference to the linear heat rate (LHR) has been substituted for the power peaking factors. This establishes consistency between the Bases of LCOs 3.2.1 through 3.2.4 and the Bases for LCO 3.2.5.

32. Text in the ITS 3.2.4 Bases providing reference to an allowance for movement through the specified Applicability conditions as an exception to ITS LCO 3.0.3 was removed from the Bases because it is unnecessary. The ITS LCO 3.2.4 Required Actions direct the necessary remedial measures. Other Condition statements provide the Required Actions should those remedial measures not be satisfied (i.e. Required Action or associated Completion Time not met). No circumstances should exist that require entry into ITS LCO 3.0.3 and no exceptions should be necessary should entry into ITS LCO 3.0.3 be required. Further, the most limiting Required Action would require that the THERMAL POWER of the unit be reduced to less than 20% RTP. This would place the unit in a condition outside of the Applicability of the Specification and simultaneously satisfy the requirements of ITS LCO 3.0.3.

CTS

3.2 POWER DISTRIBUTION LIMITS

3.2.1 Regulating Rod Insertion Limits

LCO 3.2.1 Regulating rod groups shall be within the physical insertion, sequence, and overlap limits specified in the COLR. 3.1.3.5 3.5.2.1 3.5.2.5.2 3.5.2.5.3

<< move

APPLICABILITY: MODES 1 and 2. 3.1.3.5 3.5.2

-----NOTE-----
This LCO is not applicable while performing SR 3.1.4.2. — 3.5.2.5.3
Not required for any regulating rod repositioned to perform — 16

ACTIONS

<Insert 3.2-1A>

CONDITION	REQUIRED ACTION	COMPLETION TIME	CTS
A. Regulating rod groups inserted in restricted operational region, ~~or sequence or overlap, or any combination, not met.~~	A.1 Perform SR 3.2.5.1.	Once per 2 hours	N/A
	AND		1
	A.2 Restore regulating rod groups to within ~~limits~~ acceptable region.	24 hours from discovery of failure to meet the LCO	3.5.2.5.3
B. Required Action and associated Completion Time of Condition A not met.	B.1 Reduce THERMAL POWER to less than or equal to THERMAL POWER allowed by regulating rod group insertion limits.	2 hours	3.5.2.5.3
C. Regulating rod groups sequence or overlap requirements not met.	C.1 Restore regulating rod groups to within limits	4 hours	3.5.2.5.3 1

(continued)

<INSERT 3.2-1A>

A.1	------NOTE------ Only required when THERMAL POWER is > 20% RTP. ---------------

CTS

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME	CTS
D. ~~C.~~ Regulating rod groups inserted in unacceptable operation~~al~~ region.	D.1 ~~C.1~~ Initiate boration to restore SDM to ~~≥ 1% Δk/k.~~ within the limit provided in the COLR.	15 minutes	3.5.2.1 (2)
	AND D.2.1 ~~C.2.1~~ Restore regulating rod groups to within restricted ~~operating~~ operation region.	2 hours	3.5.2.5.3 (1)
	OR D.2.2 ~~C.2.2~~ Reduce THERMAL POWER to less than or equal to the THERMAL POWER allowed by the regulating rod group insertion limits.	2 hours	3.5.2.5.3 (1)
E. ~~D.~~ Required Action(s) and associated Completion Time(s) of Condition C or D not met.	E.1 ~~D.1~~ Be in MODE 3.	6 hours	3.5.2.5.3 (1)

CTS

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY	
SR 3.2.1.1	Verify regulating rod groups are within the sequence and overlap limits as specified in the COLR.	~~4 hours when the CONTROL ROD drive sequence alarm is inoperable~~ ~~AND~~ 12 hours ~~when the CONTROL ROD drive sequence alarm is OPERABLE~~	N/A 3
SR 3.2.1.2	Verify regulating rod groups meet the insertion limits as specified in the COLR	~~4 hours when the regulating rod insertion limit alarm is inoperable~~ ~~AND~~ 12 hours ~~when the regulating rod insertion limit alarm is OPERABLE~~	N/A 3
~~SR 3.2.1.3~~	~~Verify SDM ≥ 1% Δk/k.~~	~~Within 4 hours prior to achieving criticality~~	4

CTS

3.2 POWER DISTRIBUTION LIMITS

3.2.2 AXIAL POWER SHAPING ROD (APSR) Insertion Limits

LCO 3.2.2 APSRs shall be positioned within the limits specified in the COLR. 3.5.2.5.4

APPLICABILITY: MODES 1 and 2. 3.5.2

ACTIONS

<INSERT 3.2-4A>

CONDITION	REQUIRED ACTION	COMPLETION TIME	CTS
A. APSRs not within limits.	A.1 Perform SR 3.2.5.1.	Once per 2 hours	6 N/A
	AND		
	A.2 Restore APSRs to within limits.	24 hours	3.5.2.5.4
B. Required Action and associated Completion Time not met.	B.1 Be in MODE 3.	6 hours	N/A

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY	
SR 3.2.2.1	Verify APSRs are within acceptable limits specified in the COLR.	12 hours	N/A

<INSERT 3.2-4A>

A.1	------NOTE------ Only required when THERMAL POWER is > 20% RTP. ----------------

3.2 POWER DISTRIBUTION LIMITS

CTS

3.2.3 AXIAL POWER IMBALANCE Operating Limits

LCO 3.2.3 AXIAL POWER IMBALANCE shall be maintained within the limits specified in the COLR. 3.5.2.6.2

APPLICABILITY: MODE 1 with THERMAL POWER > 40% RTP. 3.5.2.6.1

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME	
A. AXIAL POWER IMBALANCE not within limits.	A.1	Perform SR 3.2.5.1.	Once per 2 hours	N/A
	AND			
	A.2	Reduce AXIAL POWER IMBALANCE to within limits.	24 hours	3.5.2.6.3 3.5.2.6.4 edit
B. Required Action and associated Completion Time not met.	B.1	Reduce THERMAL POWER to ≤ 40% RTP.	~~2~~ 4 hours	7 3.5.2.6.4

CTS

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY	
SR 3.2.3.1	Verify AXIAL POWER IMBALANCE is within limits as specified in the COLR.	~~1 hour when AXIAL POWER IMBALANCE alarm is inoperable~~ ~~AND~~ 12 hours ~~when AXIAL POWER IMBALANCE alarm is OPERABLE~~	3.5.2.6.1

8

CTS

3.2 POWER DISTRIBUTION LIMITS

3.2.4 QUADRANT POWER TILT (QPT)

LCO 3.2.4 QPT shall be maintained less than or equal to the steady state limits specified in the COLR. 3.5.2.4.1

APPLICABILITY: MODE 1 with THERMAL POWER > ~~[~~20~~]~~% RTP. 3.5.2.4.4

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME	CTS
			9
A. QPT greater than the steady state ~~limit and less than or equal to the transient limit.~~ Limits specified in the COLR.	A.1.1 Perform SR 3.2.5.1.	Once per 2 hours	N/A
	OR		
	A.1.2.1 Reduce THERMAL POWER ≥ 2% RTP from the ALLOWABLE THERMAL POWER for each 1% of QPT greater than the steady state limit.	2 hours OR 2 hours after last performance of SR ~~3.2.2.1~~ 3.2.5.1	3.5.2.4.1 3.5.2.4.2 N/A EDIT
	AND		15
	A.1.2.2 Reduce ~~nuclear overpower trip setpoint and~~ nuclear overpower based on Reactor Coolant System flow and AXIAL POWER IMBALANCE trip setpoint ≥ 2% RTP from the ALLOWABLE THERMAL POWER for each 1% of QPT greater than the steady state limit.	10 hours OR 10 hours after last performance of SR 3.2.5.1	3.5.2.4.2.a N/A 17
<INSERT 3.2-7A>			18
	AND		
		(continued)	

<INSERT 3.2-7A>

			CTS
	AND		
A.1.2.3	Reduce the regulating group insertion limits given in the COLR ≥ 2% RTP from the ALLOWABLE THERMAL POWER for each 1% of QPT greater than the steady state limit.	10 hours OR 10 hours after last performance of SR 3.2.5.1	3.5.2.4.2.b N/A
	AND		
A.1.2.4	Reduce the Operational Power Imbalance Setpoints given in the COLR ≥ 2% RTP from the ALLOWABLE THERMAL POWER for each 1% of QPT greater than the steady state limit.	10 hours OR 10 hours after last performance of SR 3.2.5.1	3.5.2.4.2.c N/A

CTS

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME	CTS
A. (continued)	A.2	Restore QPT to less than or equal to the steady state limit.	24 hours from discovery of failure to meet the LCO	N/A
~~B. QPT greater than the transient limit and less than or equal to the maximum limit due to misalignment of a CONTROL ROD or an APSR.~~	~~B.1~~	~~Reduce THERMAL POWER ≥ 2% RTP from ALLOWABLE THERMAL POWER for each 1% of QPT greater than the steady state limit.~~	~~30 minutes~~	9
	~~AND~~			
	~~B.2~~	~~Restore QPT to less than or equal to the transient limit.~~	~~2 hours~~	
B ~~C~~. Required Action and associated Completion Time of Condition A ~~or B~~ not met.	B ~~C~~.1	Reduce THERMAL POWER to < 60% of the ALLOWABLE THERMAL POWER.	2 hours	H9 N/A
	AND			
	B ~~C~~.2	Reduce nuclear overpower trip setpoint to ≤ 65.5% of the ALLOWABLE THERMAL POWER.	10 hours	H9

(continued)

CTS

ACTIONS (continued)

CONDITION	REQUIRED ACTION		COMPLETION TIME	CTS
~~D. QPT greater than the transient limit and less than or equal to the maximum limit due to causes other than the misalignment of either CONTROL ROD or APSR.~~	~~D.1~~	~~Reduce THERMAL POWER to < 60% of the ALLOWABLE THERMAL POWER.~~	~~2 hours~~	9
	~~AND~~			
	~~D.2~~	~~Reduce nuclear overpower trip setpoint to ≤ 65.5% of the ALLOWABLE THERMAL POWER.~~	~~10 hours~~	
C. ~~E.~~ Required Action and associated Completion Time for Condition ~~C~~ B ~~or D~~ not met.	C ~~E~~.1	Reduce THERMAL POWER to ≤ ~~20~~% RTP.	4 ~~2~~ hours	7 N/A 9
D. ~~F.~~ QPT greater than the maximum limit ~~.~~ specified in the COLR.	D ~~F~~.1	Reduce THERMAL POWER to ≤ ~~20~~% RTP.	4 ~~2~~ hours	3.5.2.4.3

CTS

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY	
SR 3.2.4.1	Verify QPT is within limits as specified in the COLR.	~~12 hours when the QPT alarm is inoperable~~ ~~AND~~ 7 days ~~when the QPT alarm is OPERABLE~~ AND When QPT has been restored to less than or equal to the steady state limit, 1 hour for 12 consecutive hours, or until verified acceptable at ≥ 95% RTP	3.5.2.4.4 22 N/A

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CTS

3.2 POWER DISTRIBUTION LIMITS

3.2.5 Power Peaking ~~Factors~~

Linear heat rate (LHR)

LCO 3.2.5 ~~$F_Q(Z)$ and $F^N_{\Delta H}$~~ shall be within the limits specified in the COLR. 4.1.d

N/A

APPLICABILITY: MODE 1, with THERMAL POWER > 20% RTP. 23

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. LHR ~~$F_Q(Z)$~~ not within limit. s	~~A.1~~	~~Reduce THERMAL POWER ≥ 1% RTP for each 1% that $F_Q(Z)$ exceeds limit.~~	2 hours ~~15 minutes~~
	~~AND~~		
	~~A.2~~	~~Reduce nuclear overpower trip setpoint and nuclear overpower based on Reactor Coolant System (RCS) flow and AXIAL POWER IMBALANCE trip setpoint ≥ 1% RTP for each 1% that $F_Q(Z)$ exceeds limit.~~	~~8 hours~~
	~~AND~~		
	~~A.3~~	Restore LHR ~~$F_Q(Z)$~~ to within limit. s	~~24 hours~~

move

to

N/A

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(continued)

CTS

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
B. $F_{\Delta H}^{N}$ not within limit.	B.1 Reduce THERMAL POWER ≥ RH(%) RTP (specified in the COLR) for each 1% that $F_{\Delta H}^{N}$ exceeds limit.	15 minutes
	AND	
	B.2 Reduce nuclear overpower trip setpoint and nuclear overpower based on RCS flow and AXIAL POWER IMBALANCE trip setpoint ≥ RH(%) RTP (specified in the COLR) for each 1% that $F_{\Delta H}^{N}$ exceeds limit.	8 hours
	AND	
	B.3 Restore $F_{\Delta H}^{N}$ to within limit.	24 hours
B. ~~C.~~ Required Action and associated Completion Time not met.	B.1 ~~C.1~~ ~~Be in MODE 2.~~ Reduce THERMAL POWER to ≤20% RTP.	4 ~~2~~ hours

31

N/A

7

23

CTS

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY	CTS
SR 3.2.5.1	-------------------NOTE------------------- Only required to be performed when specified in LCO 3.1.8, "PHYSICS TESTS Exceptions—MODE 1," or when complying with Required Actions of LCO 3.1.4, "CONTROL ROD Group Alignment Limits"; LCO 3.2.1, "Regulating Rod Insertion Limits"; LCO 3.2.2, "AXIAL POWER SHAPING ROD (APSR) Insertion Limits"; LCO 3.2.3, "AXIAL POWER IMBALANCE Operating Limits"; LCO 3.2.4, "QUADRANT POWER TILT (QPT)." --		NA
	Verify LHR is ~~$F_Q(Z)$ and $F_{\Delta H}^N$ are~~ within limits by using the Incore Detector System to obtain a power distribution map.	As specified by the applicable LCO(s)	(31) 4.1.d

B 3.2 POWER DISTRIBUTION LIMITS

B 3.2.1 Regulating Rod Insertion Limits

BASES

BACKGROUND

The insertion limits of the regulating rods are initial condition assumptions used in all safety analyses that assume rod insertion upon reactor trip. The insertion limits directly affect the core power distributions, the worth of a potential ejected rod, the available SDM, and the initial reactivity insertion rate.

edit

SAR Section 1.4

The applicable criteria for these reactivity and power distribution design requirements are described in 10 CFR 50, Appendix A, GDC 10, "Reactor Design," GDC 26, "Reactivity Control System Redundancy and Capability," GDC 28, "Reactivity Limits" (Ref. 1), and in 10 CFR 50.46, "Acceptance Criteria for Emergency Core Cooling Systems for Light Water Nuclear Power Plants" (Ref. 2).

edit
edit
edit

Reactors

Limits on regulating rod insertion have been established, and all rod positions are monitored and controlled during power operation to ensure that the power distribution and reactivity limits defined by the design power peaking and SDM limits are not violated.

The regulating rod groups operate with a predetermined amount of position overlap, in order to approximate a linear relation between rod worth and rod position (integral rod worth). To achieve this approximately linear relationship, the regulating rod groups are withdrawn and operated in a predetermined sequence. The automatic control system controls reactivity by moving the regulating rod groups in sequence within analyzed ranges. The group sequence and overlap limits are specified in the COLR.

The regulating rods are used for precise reactivity control of the reactor. The positions of the regulating rods are normally controlled automatically by the automatic control system but can also be controlled manually. They are capable of adding reactivity quickly compared with borating or diluting the Reactor Coolant System (RCS).

rapid

changes

edit

The power density at any point in the core must be limited to maintain specified acceptable fuel design limits, including limits that ensure that the criteria specified in 10 CFR 50.46 (Ref. 2) are not violated. Together,

(continued)

BASES

BACKGROUND (continued)

LCO 3.2.1, "Regulating Rod Insertion Limits," LCO 3.2.2, "AXIAL POWER SHAPING ROD (APSR) Insertion Limits," LCO 3.2.3, "AXIAL POWER IMBALANCE Operating Limits," and LCO 3.2.4, "QUADRANT POWER TILT (QPT)," provide limits on control component operation and on monitored process variables to ensure that the core operates within the $F_Q(Z)$ and $F^N_{\Delta H}$ limits in the COLR. Operation within the $F_Q(Z)$ limits given in the COLR prevents power peaks that would exceed the loss of coolant accident (LOCA) limits derived from the analysis of the Emergency Core Cooling Systems (ECCS). ~~Operation within the $F^N_{\Delta H}$ limits given in the COLR~~ prevents departure from nucleate boiling (DNB) during a loss of forced reactor coolant flow accident. In addition to the $F_Q(Z)$ and $F^N_{\Delta H}$ limits, certain reactivity limits are met by regulating rod insertion limits. The regulating rod insertion limits also restrict the ejected CONTROL ROD worth to the values assumed in the safety analysis and ~~maintain~~ the minimum required SDM in MODES 1 and 2.

linear heat rate
and
31
Support
edit

This LCO is required to minimize fuel cladding failures that breach the primary fission product barrier and release fission products into the reactor coolant in the event of a LOCA, loss of flow accident, ejected rod accident, or other postulated accidents requiring termination by a Reactor Protection System trip function.

APPLICABLE SAFETY ANALYSES

The fuel cladding must not sustain damage as a result of normal operation ~~(Condition 1)~~ or ~~anticipated operational occurrences (Condition 2)~~. The LCOs governing regulating rod insertion, APSR position, AXIAL POWER IMBALANCE, and QPT preclude core power distributions that violate the following fuel design criteria:

abnormalities
edit

a. During a large break LOCA, the peak cladding temperature must not exceed 2200°F (Ref. 2).

b. During a loss of forced reactor coolant flow accident, there must be at least 95% probability at the 95% confidence level (the 95/95 DNB criterion) that the hot fuel rod in the core does not experience a DNB condition ~~(Ref. 1)~~. edit

c. During an ejected rod accident, the fission energy input to the fuel must not exceed 280 cal/gm (Ref. ~~3~~ 4). edir

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

d. The CONTROL RODS must be capable of shutting down the reactor with a minimum required SDM ~~with~~ which assumes the highest worth CONTROL ROD stuck fully withdrawn ~~(Ref. 1)~~. edit

Fuel cladding damage does not occur when the core is operated outside the conditions of these LCOs during normal operation. However, fuel cladding damage could result if an accident occurs with the simultaneous violation of one or more of the LCOs limiting the regulating rod position, the APSR position, the AXIAL POWER IMBALANCE, and the QPT. This potential for fuel cladding damage exists because changes in the power distribution can cause increased power peaking and correspondingly increased local linear heat rates (LHRs).

The SDM requirement is met by limiting the regulating and safety rod insertion limits such that sufficient inserted reactivity is available in the rods to shut down the reactor to hot zero power with a reactivity margin that assumes that the maximum worth rod remains fully withdrawn upon trip (Ref. 4). Operation at the SDM based regulating rod insertion limit may also indicate that the maximum ejected rod worth could be equal to the limiting value.

Operation at the regulating rod insertion limits may cause the local core power to approach the maximum linear heat generation rate or peaking factor with the allowed QPT present.

The regulating rod and safety rod insertion limits ensure that the safety analysis assumptions for SDM, ejected rod worth, and power distribution peaking factors remain valid (Refs. 3 ~~, 5, and 6)~~. and 4). edit

The regulating rod insertion limits LCO satisfies Criterion 2 of ~~the NRC Policy Statement.~~ 10CFR 50.36 (Ref. 5). 10

LCO

The limits on ~~CONTROL ROD~~ regulating rod group physical insertion, sequence, ~~including group~~ overlap, ~~and insertion positions~~ and as defined in the COLR, must be maintained because they ensure that the resulting power distribution is within the range of analyzed power distributions and that the SDM and ejected rod worth are maintained. edit

(continued)

BASES

LCO (continued)

The overlap between regulating groups provides more uniform rates of reactivity insertion and withdrawal and is imposed to maintain acceptable power peaking during regulating rod motion.

Error adjusted maximum allowable setpoints for regulating rod insertion are provided in the COLR. The setpoints are derived by an adjustment of the measurement system independent limits to allow for THERMAL POWER level uncertainty and rod position errors.

~~Actual alarm setpoints implemented in the unit may be more restrictive than the maximum allowable setpoint values to provide additional conservatism between the actual alarm setpoint and the measurement system independent limit.~~ 26

APPLICABILITY

The regulating rod sequence, overlap, and physical insertion limits shall be maintained with the reactor in MODES 1 and 2. These limits maintain the validity of the assumed power distribution, ejected rod worth, SDM, and reactivity rate insertion assumptions used in the safety analyses. Applicability in MODES 3, 4, and 5 is not required, because neither the power distribution nor ejected rod worth assumptions are exceeded in these MODES. SDM in MODES 3, 4, and 5 is governed by LCO 3.1.1, "SHUTDOWN MARGIN (SDM)." edit

move 16

LCO ~~3.1.2~~ 3.2.1 has been modified by a Note that suspends the LCO requirement ~~during the performance of~~ for those regulating rods not within the limits of the COLR solely due to testing in accordance with SR 3.1.4.2, which verifies the freedom of the rods to move. This SR ~~requires~~ may require the regulating rods to move below the LCO limit, out of group sequence, or beyond group overlap requirements, which ~~normally~~ would otherwise ~~violates~~ violate the LCO. EDIT. EDIT. EDIT.

ACTIONS

The regulating rod insertion ~~alarm~~ setpoints provided in the COLR are based on both the initial conditions assumed in the accident analyses and on the SDM. Specifically, separate insertion ~~limits~~ setpoints are specified to determine whether the unit is operating in violation of the initial conditions (e.g., the range of power distributions) assumed in the accident analyses or whether the unit is in violation of the SDM or ejected rod worth limits. Separate insertion ~~limits~~ setpoints are provided because different Required Actions and Completion Times apply, depending on which insertion ~~limit~~ setpoints has been edit 11 11 11

(continued)

BASES

ACTIONS (continued)

violated. The area between the boundaries of the acceptable operation and unacceptable operation, illustrated on the regulating rod insertion ~~limit~~ figures in the COLR, is the restricted region. The actions required when operation occurs in the restricted region are described under Condition A. The actions required when operation occurs in D ~~the unacceptable region are described under Condition C~~. The actions required when operation occurs with the regulating rod group sequence or overlap requirements not met are described under Condition C.

regions — edit — 11 — Setpoint — Operation — 1 — edit — edit

A.1

Operation with the regulating rods in the restricted region shown on the regulating rod insertion figures specified in the COLR or with any group sequence or overlap outside the ~~limits specified in the COLR~~ potentially violates the LOCA LHR limits ~~(F₀(Z) limits)~~, or the loss of flow accident DNB peaking limits ~~($F^N_{\Delta H}$ limits)~~. ~~The design calculations assume no deviation in nominal overlap between regulating rod groups. However, deviations of 5% of the core height above or below the nominal overlap may be typical and do not cause significant differences in core reactivity, in power distribution, or in rod worth, relative to the design calculations. The group sequence must be maintained because design calculations assume the regulating rods withdraw and insert in a predetermined order.~~

Operation — Setpoint — 11 — 1 — 31 — 1

For verification that $F_Q(Z)$ and $F^N_{\Delta H}$ are within their limits, SR 3.2.5.1 is performed using the Incore Detector System to obtain a three dimensional power distribution map. Verification that $F_Q(Z)$ and $F^N_{\Delta H}$ are within their limits ensures that operation with the regulating rods inserted into the restricted region does not violate the ECCS or DNB criteria ~~(Ref. 7)~~. The required Completion Time of 2 hours is acceptable in that it allows the operator sufficient time for obtaining a power distribution map and for verifying the power peaking factors. Repeating SR 3.2.5.1 every 2 hours is acceptable because it ensures that continued verification of the power peaking factors is performed as core conditions (primarily regulating rod insertion and induced xenon redistribution) change.

LHRs — operation — edit — 31

Monitoring the ~~power peaking factors $F_Q(Z)$ and $F^N_{\Delta H}$~~ does not provide verification that the reactivity insertion rate on the rod trip or the ejected rod worth limit is maintained, because worth is a reactivity parameter rather than a power peaking parameter. However, if the COLR figures do not show

(continued)

BASES

ACTIONS

A.1 (continued)

Setpoint

that a rod insertion limit is ejected rod worth limited, then the ejected rod worth is no more limiting than the SDM based rod insertion limit in the core design. ~~(Ref. 8).~~ Ejected rod worth limits are independently maintained by the Required Actions of Conditions A and ~~C~~ D.

11
1

<INSERT B3.2-6A> ——>>

A.2

operation

Indefinite operation with the regulating rods inserted in the restricted region ~~or in violation of the group sequence or overlap limits,~~ is not prudent. Even if power peaking monitoring per Required Action A.1 is continued, reactivity limits may not be met and the abnormal regulating rod insertion ~~or group configuration~~ may cause an adverse xenon redistribution, may cause the limits on AXIAL POWER IMBALANCE to be exceeded, or may adversely affect the long term fuel depletion pattern. Therefore, ~~power peaking monitoring is allowed for up to~~ 24 hours after discovery of failure to meet the requirements of this LCO. This required Completion Time is reasonable based on the low probability of an event occurring simultaneously with the limit out of specification in this relatively short time period. In addition, it precludes long term depletion with abnormal group insertions ~~or configurations,~~ thereby limiting the potential for an adverse xenon redistribution.

restoration of regulating rod groups to within their limits is required within

1

B.1

positioned

operation region

Setpoints

unit

operation

regulating rod position

If the regulating rods cannot be ~~restored~~ within the acceptable ~~operating limits~~ shown on the figures in the COLR within the required Completion Time (i.e., Required Action A.2 not met), then the ~~limits~~ can be restored by reducing the THERMAL POWER to a value allowed by the regulating rod insertion ~~limits~~ in the COLR. The required Completion Time of 2 hours is sufficient to allow the operator to complete the power reduction in an orderly manner and without challenging the ~~plant~~ systems. Operation for up to 2 hours more in the restricted region shown in the COLR is acceptable, based on the low probability of an event occurring simultaneously with the ~~limit~~ out of specification in this relatively short time period. ~~In addition, it precludes long term depletion with abnormal group insertions~~

11
11
edit
1

(continued)

<INSERT B3.2-6A>

Required Action A.1 is modified by a Note that requires the performance of SR 3.2.5.1 only when THERMAL POWER is greater than 20% RTP. This establishes a Required Action that is consistent with the Applicability of LCO 3.2.5, "Power Peaking."

BASES

ACTIONS (continued) 1

~~B.1 (continued)~~

~~or configurations and limits the potential for an adverse xenon redistribution.~~ 1

<INSERT B 3.2-7A> 1

D.1 ~~C.1~~ 1

Operation in the unacceptable Operation region shown on the figures in the COLR corresponds to power operation with an SDM less than the minimum required value or with the ejected rod worth greater than the allowable value. The regulating rods may be inserted too far to provide sufficient negative reactivity insertion following a reactor trip and the ejected rod worth may exceed its initial condition limit. Therefore, the RCS boron concentration must be increased to restore the regulating rod insertion to a value that preserves the SDM and ejected rod worth limits. ~~The RCS boration must occur as described in Section B 3.1.1.~~ The required Completion Time of 15 minutes to initiate boration is reasonable, based on limiting the potential xenon redistribution, the low probability of an accident occurring in this relatively short time period, and the number of steps required to complete this Action. This period allows the operator sufficient time for aligning the required valves and for starting the boric acid pumps. Boration continues until the regulating rod group positions are restored to at least within the restricted operation~~al~~ region, which restores the minimum SDM ~~capability~~ and reduces the potential ejected rod worth to within its limit. edit 1 edit

D.2.1 ~~C.2.1~~

The required Completion Time of 2 hours from initial discovery of a regulating rod group in the unacceptable operation region until its restoration to within the restricted ~~operating~~ operation region shown on the figures in the COLR allows sufficient time for borated water to enter the RCS from the chemical addition and makeup and purification systems, thereby allowing the regulating rods to be withdrawn to the restricted operation region. Operation in the restricted operation region for up to ~~an additional~~ 2 hours is reasonable, based on limiting the potential for an adverse xenon redistribution, the low probability of an accident occurring in this relatively short time period, and the number of steps required to complete this Action. 1 edit

(continued)

<INSERT B3.2-7A>

C.1

Operation with the regulating rod groups out of sequence or with the group overlap limits exceeded may represent a condition beyond the assumptions used in the safety analyses. The design calculations assume no deviation in nominal overlap between regulating rod groups. However, small deviations in group overlap, as allowed by the COLR, may occur and would not cause significant differences in core reactivity, in power distribution, or rod worth, relative to the design calculations. Group sequence must be maintained because design calculations assume the regulating rods withdraw and insert in a predetermined order. The Completion Time of 4 hours is intended to restrict operation in this condition because of the potential severity associated with gross violations of group sequence or overlap requirements. The 4 hour Completion Time is based on operating experience which supports the restoration time without unnecessarily challenging unit operation and the low probability of an event occurring simultaneously with the limit out of specification.

BASES

ACTIONS (continued)

D.2.2
~~C.2.2~~

The SDM and ejected rod worth limit can also be restored by reducing the THERMAL POWER to a value allowed by the regulating rod insertion ~~limits~~ Setpoints in the COLR. The required Completion Time of 2 hours is sufficient to allow the operator to complete the power reduction in an orderly manner and without challenging the ~~plant~~ unit systems. Operation for up to 2 hours ~~more~~ in the restricted operation region shown in the COLR is acceptable, based on the low probability of an event occurring simultaneously with the limit out of specification in this relatively short time period. In addition, it precludes long term depletion with abnormal group insertions or configurations and limits the potential for an adverse xenon redistribution.

1 11 edit 1

E.1
~~D.1~~

If the ~~regulating rods cannot be restored to within the acceptable operating limits for the original THERMAL POWER or if the power reduction cannot be completed within the required Completion Time~~ Required Actions and associated Completion Times of Conditions C or D are not met, then the reactor is placed in MODE 3, in which this LCO does not apply. This Action ensures that the reactor does not continue operating in violation of the peaking limits, the ejected rod worth, the reactivity insertion rate assumed as initial conditions in the accident analyses, or the required minimum SDM assumed in the accident analyses. The required Completion Time of 6 hours is reasonable, based on operating experience regarding the amount of time required to reach MODE 3 from RTP without challenging ~~plant~~ unit systems.

21 edit

SURVEILLANCE REQUIREMENTS

SR 3.2.1.1

This Surveillance ensures that the sequence and overlap limits are not violated. A Surveillance Frequency of 12 hours ~~or 4 hours, depending on whether the CONTROL ROD drive sequence alarm is OPERABLE or not,~~ is acceptable because little rod motion occurs ~~in 4 hours~~ during this period due to fuel burnup~~, and the probability of a deviation occurring simultaneously with an inoperable sequence monitor in this relatively short time frame is low~~. Also, the Frequency

3 3

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.2.1.1 (continued)

takes into account other information available in the control room for monitoring the status of the regulating rods.

SR 3.2.1.2

CAP Setpoints With an OPERABLE regulating rod insertion limit alarm, verification of the regulating rod insertion limits as specified in the COLR at a Frequency of 12 hours is sufficient to ensure the OPERABILITY of the regulating rod insertion limit alarm and to detect regulating rod banks that may be approaching the group insertion limits, because little rod motion due to fuel burnup occurs in 12 hours. ~~If the insertion limit alarm becomes inoperable, verification of the regulating rod group position at a Frequency of 4 hours is sufficient to detect whether the regulating rod groups may be approaching or exceeding their group insertion limits, although more frequent surveillance is prudent if the regulating rod insertion limit alarm is not OPERABLE.~~ Also, the Frequency takes into account other information available in the control room for monitoring the status of the regulating rods. 11 11 3

~~SR 3.2.1.3~~

~~Prior to achieving criticality, an estimated critical position for the CONTROL RODS is determined. Verification that SDM meets the minimum requirements ensures that sufficient SDM capability exists with the CONTROL RODS at the estimated critical position if it is necessary to shut down or trip the reactor after criticality. The Frequency of 4 hours prior to criticality provides sufficient time to verify SDM capability and establish the estimated critical position.~~ 4

REFERENCES

1. SAR, Section 1.4 ~~10 CFR 50, Appendix A~~, GDC 10, and GDC ~~26~~, GDC 28. edit
2. 10 CFR 50.46.

(continued)

BASES

REFERENCES (continued)

3. FSAR, Chapter ~~Section [ ]~~. 3 edit
4. FSAR, Chapter ~~Section [ ]~~. 14 edit
~~5. FSAR, Section [ ].~~ edit
~~6. FSAR, Section [ ].~~
~~7. FSAR, Section [ ].~~ 1
~~8. FSAR, Section [ ].~~

5. 10 CFR 50.36 10

B 3.2 POWER DISTRIBUTION LIMITS

B 3.2.2 AXIAL POWER SHAPING ROD (APSR) Insertion Limits

BASES

BACKGROUND

The insertion limits of the APSRs are initial condition assumptions in all safety analyses that are affected by core power distributions. The applicable criterion for these power distribution design requirements are ~~10 CFR 50, Appendix A~~ SAR Section 1.4, GDC 10, "Reactor Design" (Ref. 1), and 10 CFR 50.46, "Acceptance Criteria for Emergency Core Cooling Systems for Light Water Nuclear Power ~~Plants~~ Reactors" (Ref. 2). edit edit

Limits on APSR insertion have been established, and all APSR positions are monitored and controlled during power operation to ensure that the power distribution defined by the design power peaking limits is maintained.

The power density at any point in the core must be limited to maintain specified acceptable fuel design limits, including limits that meet the criteria specified in Reference 2. Together, LCO 3.2.1, "Regulating Rod Insertion Limits," LCO 3.2.2, "AXIAL POWER SHAPING ROD (APSR) Insertion Limits," LCO 3.2.3, "AXIAL POWER IMBALANCE Operating Limits," and LCO 3.2.4, "QUADRANT POWER TILT (QPT)," provide limits on control component operation and on monitored process variables to ensure that the core operates within the ~~$F_Q(Z)$~~ linear heat rate (LHR) ~~and~~ and $F^N_{\Delta H}$ limits in the COLR. Operation within the ~~$F_Q(Z)$~~ LHR limits given in the COLR prevents power peaks that exceed the loss of coolant accident (LOCA) limits derived from the analysis of the Emergency Core Cooling Systems (ECCS) ~~. Operation within the $F^N_{\Delta H}$ limits given in the COLR~~ and prevents departure from nucleate boiling (DNB) during a loss of forced reactor coolant flow accident. The APSRs ~~are not required for reactivity insertion rate on trip or SDM and, therefore, they~~ do not ~~trip~~ insert upon a reactor trip. 31 5 edit

This LCO is required to minimize fuel cladding failures that would breach the primary fission product barrier and release fission products to the reactor coolant in the event of a LOCA, loss of flow accident, ejected rod accident, or other postulated accident requiring termination by a Reactor Protection System trip function.

(continued)

BASES (continued)

APPLICABLE SAFETY ANALYSES

The fuel cladding must not sustain ~~damage as a result of normal operation (Condition 1)~~ or ~~anticipated operational occurrences (Condition 2)~~ abnormalities. Acceptance criteria for the safety and regulating rod insertion, APSR position, AXIAL POWER IMBALANCE, and QPT LCOs preclude core power distributions that violate the following fuel design criteria: edit

a. During a large break LOCA, the peak cladding temperature must not exceed 2200°F (Ref. 2);

b. During a loss of forced reactor coolant flow accident, there must be at least 95% probability at the 95% confidence level (the 95/95 DNB criterion) that the hot fuel rod in the core does not experience a DNB condition;

c. During an ejected rod accident, the fission energy input to the fuel must not exceed 280 cal/gm (Ref. 3); and

d. CONTROL RODS must be capable of shutting down the reactor with a minimum required SDM ~~with~~ which assumes the highest worth CONTROL ROD stuck fully withdrawn (GDC 26, Ref. 1). edit

Fuel cladding damage does not occur when the core is operated outside these LCOs during normal operation. However, fuel cladding damage could result should an accident occur simultaneously with violation of one or more of these LCOs. This potential for fuel cladding damage exists because changes in the power distribution can cause increased power peaking and corresponding increased local linear heat rates.

Operation at the APSR insertion limits may approach the maximum allowable linear heat generation rate ~~or peaking factor~~ with the allowed QPT present. 31

In MODES 1 and 2 while critical, The APSR insertion limits satisfy Criterion 2 of ~~the NRC Policy Statement.~~ 10CFR50.36 (Ref 4). 10

In MODE 2 while subcritical, the APSR insertion limits satisfy Criterion 4 of 10CFR50.36.

LCO

The ~~limits~~ Setpoints on APSR physical insertion as defined in the COLR must be maintained because they serve the function of 11

(continued)

BASES

LCO (continued)

controlling the power distribution within an acceptable range.

The fuel cycle design assumes APSR withdrawal at the ~~effective full power day~~ ~~(~~EFPD~~)~~ burnup window specified in the COLR. Prior to this window, the APSRs ~~cannot be~~ are maintained in accordance with operating guidelines provided by reactor engineering during ~~fully withdrawn in~~ steady state operation. After this window, the APSRs are not allowed to be reinserted for the remainder of the fuel cycle. (13) (29)

~~Error adjusted maximum allowable setpoints for APSR insertion are provided in the COLR. The setpoints are derived by adjustment of the measurement system independent limits to allow for THERMAL POWER level uncertainty and rod position errors.~~

~~Actual alarm setpoints implemented in the unit may be more restrictive than the maximum allowable setpoint values to allow for additional conservatism between the actual alarm setpoints and the measurement system independent limits.~~ (26)

APPLICABILITY

The APSR physical insertion limits shall be maintained with the reactor in MODES 1 and 2. These limits maintain the power distribution within the range assumed in the accident analyses. In MODES 1 and 2, the limits on APSR insertion specified by this LCO maintain the axial fuel burnup design conditions assumed in the reload safety evaluation analysis. ~~In MODE 2, applicability is required because $k_{eff} \geq 0.99$.~~ Applicability in MODES 3, 4, and 5 is not required, because ~~the power distribution assumptions in the accident analyses would not be exceeded in these MODES~~ reactor is subcritical. edit edit edit

ACTIONS

For steady state power operation, a normal position for APSR insertion is specified in the station operating procedures. The APSRs may be positioned as necessary for transient AXIAL POWER IMBALANCE control until the fuel cycle design requires them to be fully withdrawn. (Not all fuel cycles may incorporate APSR withdrawal.) APSR position limits are not imposed for gray APSRs, with two exceptions. If the fuel cycle design incorporates an APSR withdrawal (usually near end of cycle (EOC)), the APSRs may not be maintained in the fully withdrawn position prior to the fuel cycle burnup for

(continued)

BASES

ACTIONS (continued)

the APSR withdrawal. If this occurs, the APSRs must be restored to their normal inserted position. Conversely, after the fuel cycle burnup for the APSR withdrawal occurs, the APSRs may not be reinserted for the remainder of the fuel cycle. These restrictions apply to ensure the axial burnup distribution that accumulates in the fuel will be consistent with the expected (as designed) distribution.

A.1

Linear heat rates

For verification that the core ~~parameters $F_Q(Z)$ and $F^N_{\Delta H}$~~ are within their limits, SR 3.2.5.1 is performed using the Incore Detector System to obtain a three dimensional power distribution map. Successful verification that ~~$F_Q(Z)$ and $F^N_{\Delta H}$~~ the LHRs are within their limits ensures that operation with the APSRs inserted or withdrawn in violation of the ~~times~~ setpoints specified in the COLR do not violate either the ECCS or DNB criteria ~~(Ref. 4)~~. The required Completion Time of 2 hours is reasonable to allow the operator to obtain a power distribution map and to verify the ~~power peaking factors~~ LHRs. Repeating SR 3.2.5.1 every 2 hours is reasonable to ensure that continued verification of the ~~power peaking factors~~ is obtained as core conditions (primarily the regulating rod insertion and induced xenon redistribution) change. 31 11 edit 6

<INSERT B3.2-14A> —>>

A.2

positioned

Indefinite operation with the APSRs ~~inserted or withdrawn~~ in violation of the ~~times~~ setpoints specified in the COLR is not prudent. Even if ~~power peaking~~ LHR monitoring per Required Action A.1 is continued, the abnormal APSR ~~insertion or withdrawal~~ positioning may cause an adverse xenon redistribution, may cause the limits on AXIAL POWER IMBALANCE to be exceeded, or may affect the long term fuel depletion pattern. Therefore, ~~power peaking monitoring~~ operation is allowed for up to 24 hours. This required Completion Time is reasonable based on the low probability of an event occurring simultaneously with the APSR ~~limit~~ position out of specification. In addition, it precludes long term depletion with the APSRs in positions that have not been analyzed, thereby limiting the potential for an adverse xenon redistribution. This time limit also ensures that the intended burnup distribution is maintained, and allows the operator sufficient time to reposition the APSRs to correct their positions. edit 11 edit 31 edit

(continued)

<INSERT B3.2-14A>

Required Action A.1 is modified by a Note that requires the performance of SR 3.2.5.1 only when THERMAL POWER is greater than 20% RTP. This establishes a Required Action that is consistent with the Applicability of LCO 3.2.5, "Power Peaking."

BASES

ACTIONS

A.2 (continued)

Because the APSRs are not operated by the automatic control system, manual action by the operator is required to restore the APSRs to the positions specified in the COLR.

B.1

21

Required Action and associated

If the ~~APSRs cannot be restored to their intended positions within the required~~ Completion Time ~~of 24 hours~~ are not met, the reactor must be placed in MODE 3, in which this LCO does not apply. This action ensures that the fuel does not continue to be depleted in an unintended burnup distribution. The required Completion Time of 6 hours is reasonable, based on operating experience regarding the time required to reach MODE 3 from RTP in an orderly manner and without challenging ~~plant~~ unit systems.

edit

SURVEILLANCE REQUIREMENTS

SR 3.2.2.1

End of Cycle

edit

Fuel cycle designs that allow APSR withdrawal near EOC do not permit reinsertion of APSRs after the time of withdrawal. ~~When the plant computer is OPERABLE, the operator will receive a computer alarm if the APSRs insert after that time in core life when the APSR withdrawal occurs.~~ 30 Verification that the APSRs are within their insertion ~~limits~~ setpoints at a 12 hour Frequency is sufficient to ensure that the APSR insertion ~~limits~~ setpoints are preserved ~~and the computer alarm remains OPERABLE~~. 11 30 The 12 hour Frequency required for performing this verification is sufficient because APSRs are positioned by manual control and are normally moved infrequently. ~~The probability of a deviation occurring simultaneously with an inoperable computer alarm is low in this relatively short time frame. Also~~, The 30 CAP Frequency takes into account other information available in the control room for monitoring the axial power distribution in the reactor core.

(continued)

BASES (continued)

REFERENCES

1. SAR, Section 1.4 ~~10 CFR 50, Appendix A~~ GDC 10 and GDC 26. edit
2. 10 CFR 50.46.
3. ~~F~~SAR, Chapter ~~[ ]~~ 14. edit
4. ~~4. FSAR, Chapter [ ].~~ 6
4. 10CFR50.36. 10

B 3.2 POWER DISTRIBUTION LIMITS

B 3.2.3 AXIAL POWER IMBALANCE Operating Limits

BASES

BACKGROUND

This LCO is required to limit the core power distribution based on accident initial condition criteria.

The power density at any point in the core must be limited to maintain specified acceptable fuel design limits, including limits that satisfy the criteria specified in 10 CFR 50.46 (Ref. 1). This LCO provides limits on AXIAL POWER IMBALANCE to ensure that the core operates within the Linear heat rate (LHR) ~~$F_Q(Z)$ and $F_{\Delta H}$~~ limits given in the COLR. Operation within the LHR ~~$F_Q(Z)$~~ limits given in the COLR prevents power peaks that exceed the loss of coolant accident (LOCA) limits derived from the analysis of the Emergency Core Cooling Systems (ECCS) ~~. Operation within the $F_{\Delta H}$ limits given in the COLR~~ and prevents departure from nucleate boiling (DNB) during a loss of forced reactor coolant flow accident. 31

This LCO is required to limit fuel cladding failures that breach the primary fission product barrier and release fission products into the reactor coolant in the event of a LOCA, loss of forced reactor coolant flow accident, or other postulated accident requiring termination by a Reactor Protection System trip function. This LCO limits the amount of damage to the fuel cladding during an accident by maintaining the validity of the assumptions in the safety analyses related to the initial power distribution and reactivity.

Fuel cladding failure during a postulated LOCA is limited by restricting the maximum ~~linear heat rate~~ (LHR) so that the peak cladding temperature does not exceed 2200°F (Ref. ~~2~~ 1). edit edit
Peak cladding temperatures > 2200°F cause severe cladding failure by oxidation due to a Zircaloy water reaction. Other criteria must also be met (e.g., maximum cladding oxidation, maximum hydrogen generation, coolable geometry, and long term cooling). However, peak cladding temperature is usually most limiting.

Proximity to the DNB condition is expressed by the departure from nucleate boiling ratio (DNBR), defined as the ratio of the cladding surface heat flux required to cause DNB to the actual cladding surface heat flux. The minimum DNBR value

(continued)

BASES

BACKGROUND (continued)

during both normal operation and anticipated transients is limited to the DNBR correlation limit for the particular fuel design in use and is accepted as an appropriate margin to DNB. The DNB correlation limit ensures that there is at least 95% probability at the 95% confidence level (the 95/95 DNB criterion) that the hot fuel rod in the core does not experience DNB.

The measurement system independent limits on AXIAL POWER IMBALANCE are determined directly analytically by the reload safety evaluation analysis without adjustment for measurement system error and uncertainty. Operation beyond these limits could invalidate the assumptions used in the accident analyses regarding the core power distribution. The AXIAL POWER IMBALANCE setpoints provided in the COLR account for measurement system error and uncertainty.

edit
(11)

APPLICABLE SAFETY ANALYSES

The fuel cladding must not sustain damage as a result of normal operation (Condition 1) and ~~anticipated operational occurrences (Condition 2)~~ abnormalities. The LCOs based on power distribution, LCO 3.2.1, "Regulating Rod Insertion Limits," LCO 3.2.2, "AXIAL POWER SHAPING ROD (APSR) Insertion Limits," LCO 3.2.3, "AXIAL POWER IMBALANCE Operating Limits," and LCO 3.2.4, "QUADRANT POWER TILT (QPT)," preclude core power distributions that would violate the following fuel design criteria:

edit

a. During a large break LOCA, peak cladding temperature must not exceed 2200°F (Ref. 1);

b. During a loss of forced reactor coolant flow accident, there must be at least a 95% probability at the 95% confidence level (the 95/95 DNB criterion) that the hot fuel rod in the core does not experience a DNB condition.

The regulating rod positions, the APSR positions, the AXIAL POWER IMBALANCE, and the QPT are process variables that characterize and control the three dimensional power distribution of the reactor core.

Fuel cladding damage does not occur when the core is operated outside this LCO during normal operation. However, fuel cladding damage could result should an accident occur with simultaneous violation of one or more of the LCOs governing the four process variables cited above. This

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

potential for fuel cladding damage exists because changes in the power distribution can cause increased power peaking and corresponding increased local LHRs.

The regulating rod insertion, the APSR positions, the AXIAL POWER IMBALANCE, and the QPT are monitored and controlled during power operation to ensure that the power distribution is within the bounds set by the safety analyses. The axial power distribution is maintained primarily by the AXIAL POWER IMBALANCE and the APSR position limits; and the radial power distribution is maintained primarily by the QPT limits. The regulating rod insertion limits affect both the radial and axial power distributions.

The dependence of the core power distribution on burnup, regulating rod insertion, APSR position, and spatial xenon distribution is taken into account when the reload safety evaluation analysis is performed.

Operation at the AXIAL POWER IMBALANCE limit must be ~~interpreted as operating the~~ core at the maximum allowable ~~$F_Q(Z)$ or $F_{\Delta H}$ peaking factors~~ LHR assumed as initial conditions for the accident analyses with the allowed QPT present. 31

AXIAL POWER IMBALANCE satisfies Criterion 2 of ~~the NRC Policy Statement.~~ 10 CFR 50.36 (Ref. 2) 10

LCO

The power distribution LCO limits have been established based on correlations between power peaking and easily measured process variables: regulating rod position, APSR position, AXIAL POWER IMBALANCE, and QPT. The AXIAL POWER IMBALANCE envelope contained in the COLR represents the setpoints ~~for~~ beyond which the core power distribution ~~would~~ could either exceed the LOCA LHR limits or cause a reduction in the DNBR below the Safety Limit during the loss of flow accident with the allowable QPT present and with the APSR positions consistent with the limitations on APSR withdrawal determined by the fuel cycle design and specified by LCO 3.2.2.

~~Operation beyond the power distribution based LCO limits for the corresponding ALLOWABLE THERMAL POWER and simultaneous occurrence of either the LOCA or loss of forced reactor coolant flow accident has an acceptably low probability.~~ 27

(continued)

BASES

LCO (continued)

~~Therefore, if the LCO limits are violated, a short time is allowed for corrective action before a significant power reduction is required.~~ 27

The AXIAL POWER IMBALANCE maximum allowable setpoints (measurement system dependent limits) applicable for the full Incore Detector System, the Minimum Incore Detector System, and the Excore Detector System are provided in the COLR.

~~Actual alarm setpoints implemented in the unit may be more restrictive than the maximum allowable setpoint values to provide additional conservatism between the actual alarm setpoints and the measurement system independent limit.~~ 26

APPLICABILITY

In MODE 1, the limits on AXIAL POWER IMBALANCE must be maintained when THERMAL POWER is > 40% RTP to prevent the core power distribution from exceeding the LOCA and loss of flow assumptions used in the accident analyses. Applicability of these limits at ≤ 40% RTP in MODE 1 is not required. This operation is acceptable based on engineering judgment because the combination of AXIAL POWER IMBALANCE with the maximum allowable THERMAL POWER level will not result in LHRs sufficiently large to violate the fuel design limits. In MODES 2, 3, 4, 5, and 6, this LCO is not applicable because the reactor is not generating sufficient THERMAL POWER to produce fuel damage. edit 12

~~In MODE 1, it may be necessary to suspend the AXIAL POWER IMBALANCE limits during PHYSICS TESTS per LCO 3.1.8, "PHYSICS TESTS Exceptions—MODE 1." Suspension of these limits is permissible because the reactor protection criteria are maintained by the remaining LCOs governing the three dimensional power distribution and by the Surveillances required by LCO 3.1.8.~~ 27

ACTIONS

A.1

The AXIAL POWER IMBALANCE operating ~~limits~~ setpoints that maintain the validity of the assumptions regarding the power distributions in the accident analyses of the LOCA and the loss of flow accident are provided in the COLR. Operation 11

(continued)

BASES

ACTIONS

A.1 (continued)

Setpoints

within the AXIAL POWER IMBALANCE ~~limits~~ given in the COLR is the acceptable region of operation. Operation in violation of the AXIAL POWER IMBALANCE ~~limits~~ given in the COLR is the restricted region of operation. Setpoints (11)

Operation with AXIAL POWER IMBALANCE in the restricted region shown on the AXIAL POWER IMBALANCE figures in the COLR potentially violates the LOCA LHR limits ~~($F_Q(Z)$ limits)~~ or the loss of flow accident DNB peaking limits ~~($F_{\Delta H}$ limits)~~ or both. For verification that ~~$F_Q(Z)$ and $F_{\Delta H}$~~ core local LHRs are within their specified limits, SR 3.2.5.1 is performed using the Incore Detector System to obtain a three dimensional power distribution map. Verification that ~~$F_Q(Z)$ and $F_{\Delta H}$~~ are within their specified limits ensures that operation with the AXIAL POWER IMBALANCE in the restricted region does not violate the ECCS or 95/95 DNB criteria. The required Completion Time of 2 hours provides reasonable time for the operator to obtain a power distribution map and to determine and verify that the ~~power peaking factors~~ are within their specified limits. The 2 hour Frequency provides reasonable time to ensure that continued verification of the ~~power peaking factors~~ is obtained as core conditions (primarily regulating rod insertion and induced xenon redistribution) change, because little rod motion occurs in 2 hours due to fuel burnup, the potential for xenon redistribution is limited, and the probability of an event occurring in this short time frame is low. (31)

A.2

Indefinite operation with the AXIAL POWER IMBALANCE in the restricted region is not prudent. Even if ~~power peaking~~ LHR monitoring per Required Action A.1 is continued, excessive AXIAL POWER IMBALANCE over an extended period of time may cause a potentially adverse xenon redistribution to occur. (31)
Therefore, ~~power peaking~~ monitoring is only allowed for a maximum of 24 hours. This required Completion Time is reasonable based on the low probability of a limiting event occurring simultaneously with the AXIAL POWER IMBALANCE outside the ~~limits~~ Setpoints of this LCO. In addition, this limited Completion Time precludes long term depletion of the reactor fuel with excessive AXIAL POWER IMBALANCE and gives the operator sufficient time to reposition the APSRs or (51) (11)

(continued)

BASES

ACTIONS

A.2 (continued)

regulating rods to reduce the AXIAL POWER IMBALANCE because adverse effects of xenon redistribution and fuel depletion are limited.

B.1

If the Required Actions are not and the associated Completion Times of Condition A cannot be met, the AXIAL POWER IMBALANCE may exceed its specified limits and the reactor may be operating with a global axial power distribution mismatch. Continued operation in this configuration may induce an axial xenon oscillation and may result in an increased linear heat generation rate when the xenon redistributes. Reducing THERMAL POWER to ≤ 40% RTP reduces the maximum LHR to a value that does not exceed the LHR ~~$F_Q(Z)$ and $F_{\Delta H}$~~ initial condition limits assumed in the accident analyses. The required Completion Time of 4 ~~2~~ hours is reasonable based on limiting a potentially adverse xenon redistribution, the low probability of an accident occurring in this relatively short time period, and the number of steps required to complete this Action.

edit

31

7

SURVEILLANCE REQUIREMENTS

SR 3.2.3.1

~~The AXIAL~~ POWER IMBALANCE can be monitored by both the Incore and Excore Detector Systems. The AXIAL POWER IMBALANCE maximum allowable setpoints are derived from their corresponding measurement system independent limits by adjusting for both the system observability errors and instrumentation errors. Although they may be based on the same measurement system independent limits, the setpoints for the different systems are not identical because of differences in the errors applicable for each of these systems. The uncertainty analysis that defines the required error adjustment to convert the measurement system independent limits to full incore detector system limits ~~alarm setpoints~~ assumes that 75% of the detectors in each quadrant are OPERABLE. Detectors located on the core major axes are assumed to contribute one half of their output to each quadrant; detectors in the center assembly are assumed to contribute one quarter of their output to each quadrant. For AXIAL POWER IMBALANCE measurements using the Incore Detector System, the Minimum

edit

11

(continued)

BASES

edit

SURVEILLANCE REQUIREMENTS (continued)

SR 3.2.3.1 (continued)

Incore Detector System consists of OPERABLE detectors configured as follows:

a. Nine detectors shall be arranged such that there are three detectors in each of three strings and there are three detectors lying in the same axial plane, with one plane at the core midplane and one plane in each axial core half;

b. The axial planes in each core half shall be symmetrical about the core midplane; and

c. The detector strings shall not have radial symmetry.

Figure B 3.2.3-1 (Minimum Incore Detector System for AXIAL POWER IMBALANCE Measurement) depicts an example of this configuration. This arrangement is chosen to reduce the uncertainty in the measurement of the AXIAL POWER IMBALANCE by the Minimum Incore Detector System. For example, the requirement for placing one detector of each of the three strings at the core midplane puts three detectors in the central region of the core where the neutron flux tends to be higher. It also helps prevent measuring an AXIAL POWER IMBALANCE that is excessively large when the reactor is operating at low THERMAL POWER levels. The third requirement for placement of detectors (i.e., radial asymmetry) reduces uncertainty by measuring the neutron flux at core locations that are not radially symmetric.

~~SR 3.2.3.1~~

~~If the plant computer becomes inoperable, then the Excore System or Minimum Incore Detector System may be used to monitor the AXIAL POWER IMBALANCE. Although these systems do not provide a direct calculation and display of the AXIAL POWER IMBALANCE, a 1 hour Frequency provides reasonable time between calculations for detecting any trends in the AXIAL POWER IMBALANCE that may exceed its alarm setpoint and for undertaking corrective action.~~ 8

~~When the Full Incore Detector System is OPERABLE, the operator receives an alarm if the AXIAL POWER IMBALANCE increases to its alarm setpoint. When the AXIAL POWER IMBALANCE is less then the alarm setpoint,~~ verification of the AXIAL POWER IMBALANCE indication every 12 hours ensures CAP

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.2.3.1 (continued)

Setpoints

that the AXIAL POWER IMBALANCE ~~limits~~ are not violated and ~~verifies that the alarm system is OPERABLE.~~ This Surveillance Frequency is acceptable because the mechanisms that can cause AXIAL POWER IMBALANCE, such as xenon redistribution or CONTROL ROD drive mechanism malfunctions that cause slow AXIAL POWER IMBALANCE increases, can be discovered by the operator before the specified limits are violated.

takes into account other information and alarms available in the control room.

11

8

edit

REFERENCES

1. 10 CFR 50.46.
2. ~~FSAR, Chapter [15].~~ 10 CFR 50.36.

10

~~This figure for illustration only. Do not use for operation.~~

edit

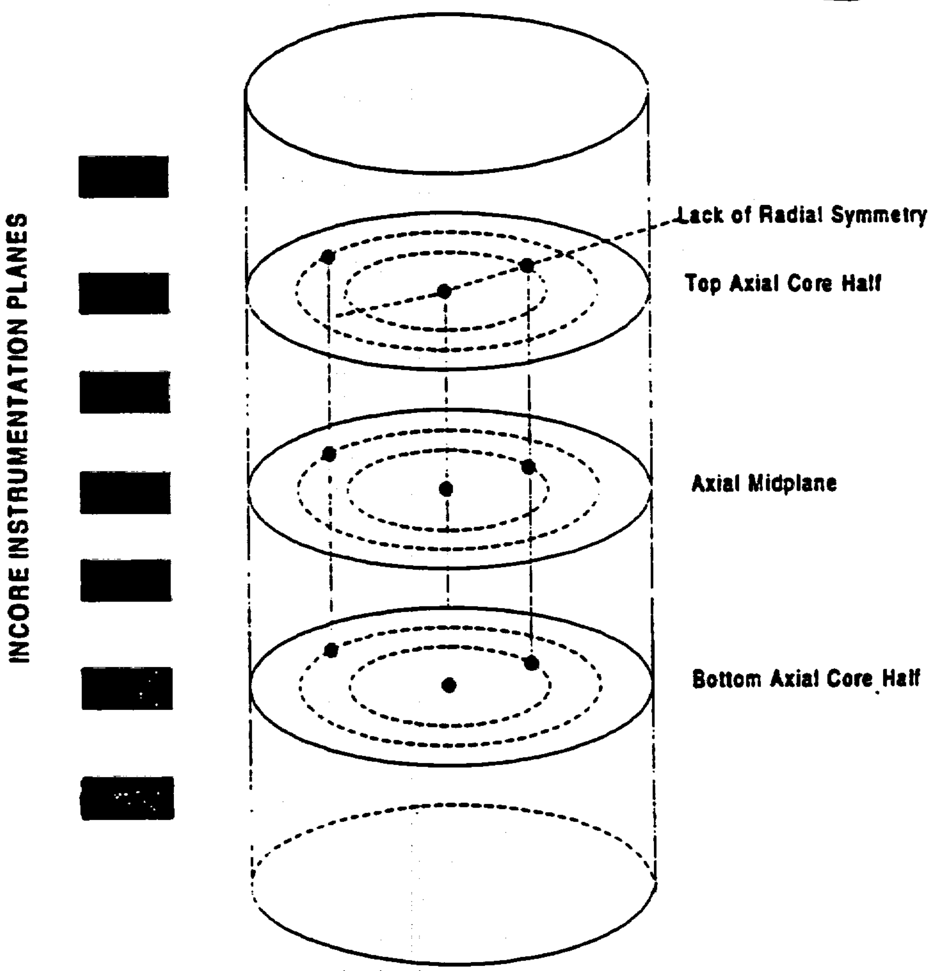

Figure B 3.2.3-1 (page 1 of 1)
Minimum Incore System for AXIAL POWER IMBALANCE Measurement

B 3.2 POWER DISTRIBUTION LIMITS

B 3.2.4 QUADRANT POWER TILT (QPT)

BASES

BACKGROUND

This LCO is required to limit the core power distribution based on accident initial condition criteria.

The power density at any point in the core must be limited to maintain specified acceptable fuel design limits, including limits that preserve the criteria specified in 10 CFR 50.46 (Ref. 1). Together, LCO 3.2.1, "Regulating Rod Insertion Limits," LCO 3.2.2, "AXIAL POWER SHAPING ROD (APSR) Insertion Limits," LCO 3.2.3, "AXIAL POWER IMBALANCE Operating Limits," and LCO 3.2.4, "QUADRANT POWER TILT (QPT)," provide limits on control component operation and on monitored process variables to ensure that the core operates within the ~~$F_Q(Z)$ and $F^N_{\Delta H}$~~ linear heat rate (LHR) limits given in the COLR. Operation within the ~~$F_Q(Z)$~~ LHR limits given in the COLR prevents power peaks that exceed the loss of coolant accident (LOCA) limits derived by Emergency Core Cooling Systems (ECCS) analysis~~. Operation within the $F^N_{\Delta H}$ limits given in the COLR~~ and prevents departure from nucleate boiling (DNB) during a loss of forced reactor coolant flow accident. 31

This LCO is required to limit fuel cladding failures that breach the primary fission product barrier and release fission products to the reactor coolant in the event of a LOCA, loss of forced reactor coolant flow, or other accident requiring termination by a Reactor Protection System trip function. This LCO limits the amount of damage to the fuel cladding during an accident by maintaining the validity of the assumptions used in the safety analysis related to the initial power distribution and reactivity.

Fuel cladding failure during a postulated LOCA is limited by restricting the maximum ~~linear heat rate (~~LHR~~)~~ so that ~~the~~ peak cladding temperature does not exceed 2200°F (Ref. ~~2~~ 1). edit edit
Peak cladding temperatures > 2200°F cause severe cladding failure by oxidation due to a Zircaloy water reaction. Other criteria must also be met (e.g., maximum cladding oxidation, maximum hydrogen generation, coolable geometry, and long term cooling). However, peak cladding temperature is usually most limiting.

(continued)

BASES

BACKGROUND (continued)

Proximity to the DNB condition is expressed by the departure from nucleate boiling ratio (DNBR), defined as the ratio of the cladding surface heat flux required to cause DNB to the actual cladding surface heat flux. The minimum DNBR value during both normal operation and anticipated transients is limited to the DNBR correlation limit for the particular fuel design in use, and is accepted as an appropriate margin to DNB. The DNBR correlation limit ensures that there is at least 95% probability at the 95% confidence level (the 95/95 DNB criterion) that the hot fuel rod in the core does not experience DNB.

The measurement system independent limits on QPT are determined analytically ~~directly~~ by the reload safety evaluation analysis without adjustment for measurement system error and uncertainty. Operation beyond these limits could invalidate core power distribution assumptions used in the accident analysis. The error adjusted maximum allowable ~~alarm~~ setpoints (measurement system dependent limits) for QPT are specified in the COLR. edit edit

APPLICABLE SAFETY ANALYSES

The fuel cladding must not sustain damage as a result of normal operation ~~(Condition 1)~~ and abnormalities ~~anticipated operational occurrences (Condition 2)~~. The LCOs based on power distribution (LCO 3.2.1, LCO 3.2.2, LCO 3.2.3, and LCO 3.2.4) preclude core power distributions that violate the following fuel design criteria:

a. During a large break LOCA, the peak cladding temperature must not exceed 2200°F (Ref. 1 ~~1~~). 14

b. During a loss of forced reactor coolant flow accident, there must be at least 95% probability at the 95% confidence level (the 95/95 DNB criterion) that the hot fuel rod in the core does not experience a DNB condition.

QPT is one of the process variables that characterize and control the three dimensional power distribution of the reactor core.

Fuel cladding damage does not occur when the core is operated outside this LCO during normal operation. However, fuel cladding damage could result if an accident occurs with

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

simultaneous violation of one or more of the LCOs governing the core power distribution. Changes in the power distribution can cause increased power peaking and correspondingly increased local LHRs.

The dependence of the core power distribution on burnup, regulating rod insertion, APSR position, and spatial xenon distribution is taken into account during the reload safety evaluation analysis. An allowance for QPT is accommodated in the analysis and resultant LCO limits. The increase in peaking taken for QPT is developed from a database of full core power distribution calculations (Ref. ~~4~~ 2). The calculations consist of simulations of many power distributions with tilt causing mechanisms (e.g., dropped or misaligned CONTROL RODS, broken APSR fingers fully inserted, misloaded assemblies, and burnup gradients). An increase of < 2% peak power per 1% QPT is supported by the analysis, therefore a value of 2% peak power increase per 1% QPT is used to bound peak power increases due to QPT. edit

Operation at the AXIAL POWER IMBALANCE or rod insertion limits must be interpreted as operating the core at the maximum allowable ~~$F_Q(Z)$ or $F^N_{\Delta H}$ peaking factors~~ LHR for accident initial conditions with the allowed QPT present. 31

QPT satisfies Criterion 2 of ~~the NRC Policy Statement~~ 10CFR 50.36 (Ref. 3). 10

LCO

The power distribution LCO limits have been established based on correlations between power peaking and easily measured process variables: regulating rod position, APSR position, AXIAL POWER IMBALANCE, and QPT. The regulating rod insertion ~~limits~~ setpoints and the AXIAL POWER IMBALANCE boundaries contained in the COLR represent the measurement system ~~in~~dependent limits at which the core power distribution either exceeds the LOCA LHR limits or causes a reduction in DNBR below the safety limit during a loss of flow accident with the allowable QPT present and with an APSR position consistent with the limitations on APSR ~~withdrawal~~ position determined by the fuel cycle design and specified by LCO 3.2.2. 11 edit

~~Operation beyond the power distribution based LCO limits for the corresponding allowable THERMAL POWER and simultaneous occurrence of one of a LOCA, loss of forced reactor coolant~~ 27

(continued)

BASES

LCO (continued)

~~flow accident, or ejected rod accident has an acceptably low probability. Therefore, if these LCO limits are violated, a short time is allowed for corrective action before a significant power reduction is required.~~ (27)

setpoints

The ~~maximum~~ allowable setpoints for steady state ~~transient, and maximum limits~~ for QPT applicable for the full symmetrical Incore Detector System, Minimum Incore Detector System, and Excore Detector System are provided ~~the setpoints are given~~ in the COLR. The setpoints for the three systems are derived by adjustment of the measurement system independent QPT limits, given in the COLR to allow for system observability and instrumentation errors. (9) (11)

also

~~Actual alarm setpoints implemented in the plant may be more restrictive than the maximum allowable setpoint values to allow for additional conservatism between the actual alarm setpoint and the measurement system independent limit.~~ (26)

~~It is desirable for an operator to retain the ability to operate the reactor when a QPT exists. In certain instances, operation of the reactor with a QPT may be helpful or necessary to discover the cause of the QPT. The combination of power level restriction with QPT in each Required Action statement restricts the local LHR to a safe level, allowing movement through the specified applicability conditions in the exception to Specification 3.0.3.~~ (27)

APPLICABILITY

In MODE 1, the limits on QPT must be maintained when THERMAL POWER is > 20% RTP to prevent the core power distribution from exceeding the design limits. The minimum power level of 20% RTP is large enough to obtain meaningful QPT indications without compromising safety. ~~Operation at or below 20% RTP with QPT up to 20% is acceptable because the resulting maximum LHR is not high enough to cause violation of the LOCA LHR limit ($F_Q(Z)$ limit) or the initial condition DNB allowable peaking limit ($F^N_{\Delta H}$ limit) during accidents initiated from this power level.~~ (27)

In MODE 2, the combination of QPT with maximum ALLOWABLE THERMAL POWER level does not result in LHRs sufficiently large to violate the fuel design limits, and therefore, applicability in this MODE is not required. Although not

(continued)

BASES

APPLICABILITY (continued)

specifically addressed in the LCO, QPTs ~~≤ 20%~~ greater than the maximum setpoint specified in the COLR in MODE 1 with THERMAL POWER < 20% RTP are allowed ~~for the same reasons~~ based on engineering judgment. (11) (12)

In MODES 3, 4, 5, and 6, this LCO is not applicable, because the reactor is not generating Significant THERMAL POWER and QPT is indeterminate. edit

~~In MODE 1, it may be necessary to suspend the QPT limits during PHYSICS TESTS per LCO 3.1.8, "PHYSICS TESTS Exceptions—MODE 1." Suspension of these limits is permissible because the reactor protection criteria are maintained by the remaining LCOs governing the three dimensional power distribution and by the Surveillances required by LCO 3.1.8.~~ (27)

ACTIONS

A.1.1

The steady state ~~limit~~ Setpoint specified in the COLR provides an allowance for QPT that may occur during normal operation. A peaking increase to accommodate QPTs up to the steady state ~~limit~~ Setpoint is allowed by the regulating rod insertion limits of LCO 3.2.1 and the AXIAL POWER IMBALANCE limits of LCO 3.2.3. (11)

Operation with QPT greater than the steady state ~~limit~~ Setpoint specified in the COLR potentially violates the LOCA LHR limits ~~($F_Q(Z)$ limits)~~, or loss of flow accident DNB peaking limits ~~($F_{\Delta H}^N$ limits)~~, or both. For verification that ~~$F_Q(Z)$ and $F_{\Delta H}^N$~~ core local LHRs are within their specified limits, SR ~~3.1.5.2~~ 3.2.5.1 is performed using the Incore Detector System to obtain a three dimensional power distribution map. Verification that ~~$F_Q(Z)$ and $F_{\Delta H}^N$~~ core local LHRs are within their limits ensures that operation with QPT greater than the steady state ~~limit~~ Setpoint does not violate the ECCS or 95/95 DNB criteria. The required Completion Time of once per 2 hours is a reasonable amount of time to allow the operator to obtain a power distribution map and to verify the ~~power peaking factors~~ core local LHRs. Repeating SR 3.2.5.1 every 2 hours is a reasonable Frequency at which to ensure that continued verification of the ~~power peaking factors~~ core local LHRs is obtained as core conditions that influence QPT change. (31) edit (31) (11) (31)

(continued)

BASES

ACTIONS (continued)

A.1.2.1

The safety analysis has shown that a conservative corrective action is to reduce THERMAL POWER by 2% RTP or more from the ALLOWABLE THERMAL POWER for each 1% of QPT in excess of the steady state ~~limit~~ Setpoint. This action limits the local LHR to a value corresponding to ~~steady state operation, thereby reducing it to a value within~~ the assumed accident initial condition limits. The required Completion Time of 2 hours is reasonable, based on limiting the potential for xenon redistribution, the low probability of an accident occurring, and the steps required to complete the Required Action. 11 edit

If QPT can be reduced to less than or equal to the steady state ~~limit~~ Setpoint in < 2 hours, the reactor may return to normal operation without undergoing a power reduction. Significant radial xenon redistribution does not occur within this amount of time. 11

The required Completion Time of 2 hours after the last performance of SR ~~3.5.2.1~~ 3.2.5.1 allows reduction of THERMAL POWER in the event the operators cannot or choose not to continue to perform SR ~~3.5.2.1~~ 3.2.5.1 as required by Required Action A.1.1. edit edit

A.1.2.2

Power operation is allowed to continue if THERMAL POWER is reduced in accordance with Required Action A.1.2.1. The same reduction (i.e., 2% RTP or more) is also applicable to ~~the nuclear overpower trip setpoint and~~ the nuclear overpower based on Reactor Coolant System (RCS) flow and AXIAL POWER IMBALANCE trip setpoint, for each 1% of QPT in excess of the steady state limit. This reduction maintains both core protection and ~~an OPERABILITY~~ thermal margins at the reduced THERMAL POWER level similar to that at RTP. The required Completion Time of 10 hours or 10 hours after the last performance of SR 3.2.5.1 is reasonable based on the need to limit the potentially adverse xenon redistribution, the low probability of an accident occurring while operating ~~out of specification~~ with the QPT limits not met, and the number of steps required to complete the Required Action. 15 edit edit 17

<INSERT B 3.2-31A> 18

(continued)

<INSERT B 3.2-31A>

A.1.2.3

Power operation is allowed to continue if restrictions are imposed on the allowed degree of regulating group insertion. This Required Action requires a reduction in the regulating group insertion setpoints given in the COLR by ≥ 2% RTP from the ALLOWABLE THERMAL POWER for each 1% of QPT greater than the steady state setpoint. Based on engineering judgment, this action is intended to reduce the potential power peaking associated with regulating rod group insertion into the core.

The Completion Time of 10 hours is reasonable based on the need to limit the potentially adverse xenon redistribution, the low probability of an accident occurring while operating with QPT limits not met, and the number of steps required to complete the Required Action. The second Completion Time of 10 hours after the last performance of SR 3.2.5.1 is based on the same reasoning and is provided in the event the operators cannot or choose not to continue to perform SR 3.2.5.1 as required by Required Action A.1.1.

A.1.2.4

Power operation is allowed to continue if restrictions are imposed on the allowed Operational Power Imbalance Setpoints given in the COLR. This Required Action results in a reduction in the allowed THERMAL POWER level as a function of AXIAL POWER IMBALANCE by ≥ 2% RTP from the ALLOWABLE THERMAL POWER for each 1% of QPT greater than the steady state limit. Based on engineering judgment, this action is intended to reduce the potential power peaking associated with the combined affects of operating with an AXIAL POWER IMBALANCE and a QPT.

The Completion Time of 10 hours is reasonable based on the need to limit the potentially adverse xenon redistribution, the low probability of an accident occurring while operating with QPT limits not met, and the number of steps required to complete the Required Action. The second Completion Time of 10 hours after the last performance of SR 3.2.5.1 is based on the same reasoning and is provided in the event the operators cannot or choose not to continue to perform SR 3.2.5.1 as required by Required Action A.1.1.

BASES

ACTIONS (continued)

A.2

Although the actions directed by Required Action A.1.2.1 restore thermal margins, if the source of the QPT is not established and corrected, it is prudent to establish increased margins. A required Completion Time of 24 hours to reduce QPT to less than the steady state limit is a reasonable time for investigation and corrective measures. edit

~~B.1~~

~~If QPT exceeds the transient limit but is equal to or less than the maximum limit due to a misaligned CONTROL ROD or APSR, then power operation is allowed to continue if the THERMAL POWER is reduced 2% RTP or more from the ALLOWABLE THERMAL POWER for each 1% of QPT in excess of the steady state limit. Thus, the transient limit is the upper bound within which the 2% for 1% power reduction rule may be applied, but only for QPTs caused by CONTROL ROD or APSR misalignment. The required Completion Time of 30 minutes ensures that the operator completes the THERMAL POWER reduction before significant xenon redistribution occurs.~~ 9

~~B.2~~

~~When a misaligned CONTROL ROD or APSR occurs, a local xenon redistribution may occur. The required Completion Time of 2 hours allows the operator sufficient time to relatch or realign a CONTROL ROD or APSR, but is short enough to limit xenon redistribution so that large increases in the local LHR do not occur due to xenon redistribution resulting from the QPT.~~

B ~~C~~.1 9

If the Required Action(s) and associated Completion Time(s) of Condition A ~~or B~~ are not met, a further power reduction is required. Power reduction to < 60% ~~RTP~~ of ALLOWABLE THERMAL POWER provides conservative protection from increased peaking due to xenon redistribution. The required Completion Time of 2 hours is reasonable to allow the operator to reduce THERMAL POWER to < 60% of ALLOWABLE THERMAL POWER without challenging ~~plant~~ unit systems. edit edit

(continued)

BASES

ACTIONS (continued)

B.2 ~~D.2~~ 9

Reduction of the nuclear overpower trip setpoint to ≤ 65.5% of ALLOWABLE THERMAL POWER after THERMAL POWER has been reduced to < 60% of ALLOWABLE THERMAL POWER maintains both core protection and OPERABILITY margin at reduced power similar to that at full power. The required Completion Time of 10 hours allows the operator sufficient time to reset the trip setpoint and is reasonable based on operating experience.

~~D.1~~

~~Power reduction to 60% of the ALLOWABLE THERMAL POWER is a conservative method of limiting the maximum core LHR for QPTs up to 20%. Although the power reduction is based on the correlation used in Required Actions A.1.2.1 and B.1, the database for a power peaking increase as a function of QPT is less extensive for tilt mechanisms other than misaligned CONTROL RODS and APSRs. Because greater uncertainty in the potential power peaking increase exists with the less extensive database, a more conservative action is taken when the tilt is caused by a mechanism other than a misaligned CONTROL ROD or APSR. The required Completion Time of 2 hours allows the operator to reduce THERMAL POWER to < 60% of the ALLOWABLE THERMAL POWER without challenging plant systems.~~ 9

~~D.2~~

~~Reduction of the nuclear overpower trip setpoint to ≤ 65.5% of the ALLOWABLE THERMAL POWER after THERMAL POWER has been reduced to < 60% of the ALLOWABLE THERMAL POWER maintains both core protection and an operating margin at reduced power similar to that at full power. The required Completion Time of 10 hours allows the operator sufficient time to reset the trip setpoint and is reasonable based on operating experience.~~ 9

C.1 ~~E.1~~

and associated

If the Required Actions ~~for Condition C or D cannot be met within the required~~ Completion ~~Time~~ Times of Condition B are not met, then the reactor will 21

(continued)

BASES

(9)

ACTIONS

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

continue in power operation with significant QPT. Either the power level has not been reduced to comply with the Required Action or the nuclear overpower trip setpoint has not been reduced within the required Completion Time. To preclude risk of fuel damage in any of these conditions, THERMAL POWER is reduced further. ~~Specification 3.0.3 normally requires a shutdown to MODE 2. However,~~ operation ~~at~~ below 20% RTP allows the operator to investigate the cause of the QPT and to correct it. Local LHRs with a large QPT do not violate the fuel design limits at or below 20% RTP. The required Completion Time of ~~2~~ 4 hours is acceptable based on limiting the potential increase in local LHRs that could occur due to xenon redistribution with the QPT out of specification.

(32) edit

(7)

(9)

D.1 ~~D.2~~

~~The maximum limit of 20% QPT is set as the upper bound within which power reduction to 60% of ALLOWABLE THERMAL POWER or power reduction of 2% for 1% (for misaligned CONTROL RODS only) applies [Ref. 4].~~ (9)

~~The maximum limit of 20% QPT is consistent with allowing power operation up to 60% of ALLOWABLE THERMAL POWER when QPT setpoints are exceeded.~~ (9) QPT in excess of the maximum ~~limit~~ Setpoint specified in the COLR can be an indication of a severe power distribution anomaly, and a power reduction to at most 20% RTP ensures local LHRs do not exceed allowable limits while the cause is being determined and corrected.

(11)

(7)

The required Completion Time of ~~2~~ 4 hours is reasonable to allow the operator to reduce THERMAL POWER to ≤ 20% RTP without challenging ~~plant~~ unit systems.

edit

SURVEILLANCE REQUIREMENTS

QPT can be monitored by both the CTP incore and CTP excore CTP detector systems. The QPT setpoints are derived from their corresponding measurement system independent limits by adjustment for system observability errors and instrumentation errors. Although they may be based on the same measurement system independent limit, the setpoints for the different systems are not identical because of

edit

(continued)

BASES

SURVEILLANCE REQUIREMENTS (continued)

differences in the errors applicable for these systems. For QPT measurements using the Incore Detector System, the Minimum Incore Detector System consists of OPERABLE detectors configured as follows:

a. Two sets of four detectors shall lie in each core half. Each set of detectors shall lie in the same axial plane. The two sets in the same core half may lie in the same axial plane.

b. Detectors in the same plane shall have quarter core radial symmetry.

edit

Figure B 3.2.4-~~2~~ 1 (Minimum Incore Detector System for QPT Measurement) depicts an example of this configuration. The full symmetric incore system for QPT uses the Incore Detector System as ~~described above~~ and is configured such that at least 75% of the Detector detectors in each core quadrant are OPERABLE.

edit
edit

SR 3.2.4.1

~~Should the plant computer become inoperable, then the Excore System or Minimum Incore Detector System may be used to monitor the QPT. Because these systems do not provide a direct calculation and display of the QPT, performing the calculations at a 12 hour Frequency is sufficient to follow any changes in the QPT that may approach the setpoint because with the exception of CONTROL ROD related effects detected by other systems, QPT changes are slow. This Frequency also provides operators sufficient time to undertake corrective actions if QPT approaches the setpoints.~~

~~When the full symmetrical Incore Detector System is in use, the operator receives an alarm, if QPT increases to the alarm setpoint. When QPT is less than the alarm setpoint,~~ CAP checking the QPT indication every 7 days ensures that the operator can determine whether the plant computer software and Incore Detector System inputs for monitoring QPT are functioning properly, and ~~that the monitoring and alarm system remains OPERABLE.~~ takes into account other information and alarms available to the operator in the control room. This procedure allows the QPT mechanisms, such as xenon redistribution, burnup gradients, and CONTROL ROD drive mechanism malfunctions, which can cause slow development of a QPT, to be detected. Operating

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(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.2.4.1 (continued)

experience has confirmed the acceptability of a Surveillance Frequency of 7 days.

Following restoration of the QPT to within the ~~steady state~~ ~~limit~~, operation at ≥ 95% RTP may proceed provided the QPT is determined to remain within the ~~steady state~~ ~~limit~~ at the increased THERMAL POWER level. In case QPT exceeds the ~~steady state~~ ~~limit~~ for more than 24 hours ~~or exceeds the transient limit~~ (Condition A~~, B, or D~~), the potential for xenon redistribution is greater. Therefore, the QPT is monitored for 12 consecutive hourly intervals to determine whether the period of any oscillation due to xenon redistribution causes the QPT to exceed the ~~steady state~~ ~~limit~~ again.

Setpoint

11

9

11

11

REFERENCES

1. 10 CFR 50.46.
~~2. FSAR, Section [ ].~~ edit
~~3. ANSI N18.2-1973, American National Standards Institute, August 6, 1973.~~ 14
2. ~~4.~~ BAW 10122A, ~~Rev. 1, May 1984.~~ "Normal Operating Controls" edit
3. 10 CFR 50.36. 10

~~This figure for illustration only.~~
~~Do not use for operation~~

edit

edit

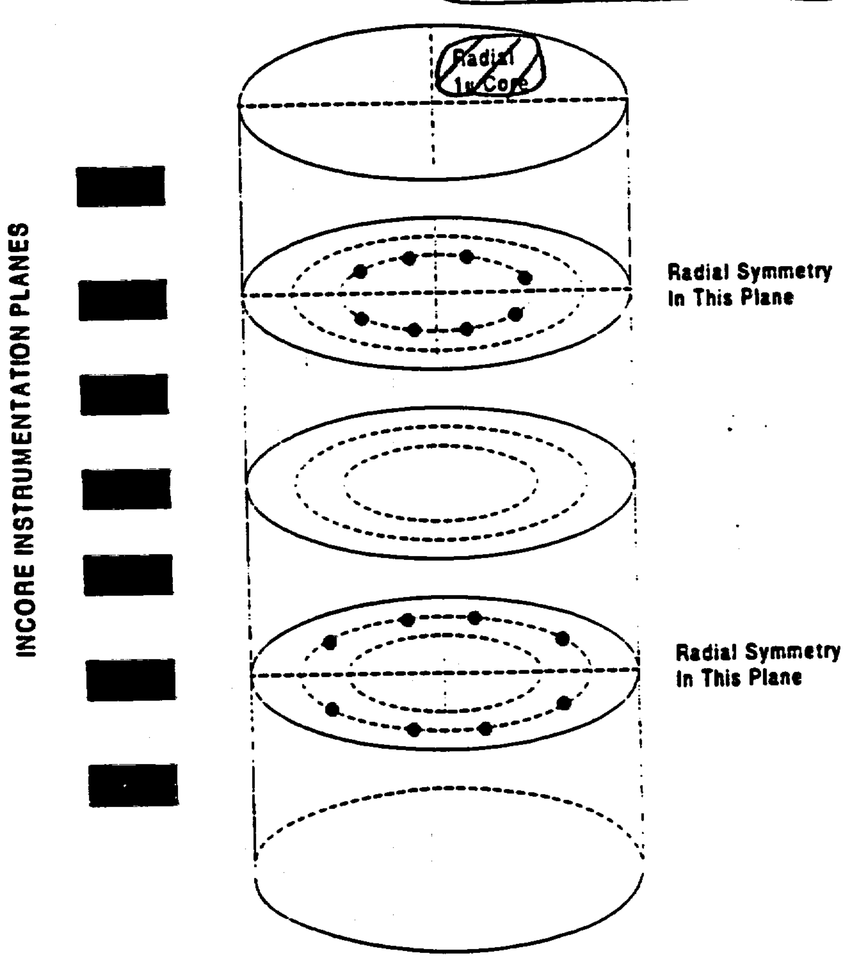

Figure B 3.2.4-1 (page 1 of 1)
Minimum Incore System for QUADRANT POWER TILT Measurement

B 3.2 POWER DISTRIBUTION LIMITS

B 3.2.5 Power Peaking Factors

BASES

BACKGROUND

The purpose of this ~~MODE 1~~ LCO is to establish limits that constrain the core power distribution within design limits during normal operation ~~(Condition 1) and~~ during ~~anticipated operational occurrences (Condition 2)~~ such that accident initial condition protection criteria are preserved. The accident initial condition criteria are preserved by bounding operation ~~at THERMAL POWER~~ within specified acceptable fuel design limits.

abnormalities and

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<INSERT B 3.2-38A>

The LOCA-limited LHR

~~$F_Q(Z)$~~ is a specified acceptable fuel design limit that preserves the initial conditions for the Emergency Core Cooling Systems (ECCS) analysis. ~~$F_Q(Z)$ is defined as the maximum local fuel rod linear power density divided by the average fuel rod linear power density, assuming nominal fuel pellet and rod dimensions.~~ ~~Because $F_Q(Z)$ is a ratio of local power densities, it is related to the maximum local (pellet) power density in a fuel rod.~~ Operation within the ~~$F_Q(Z)$~~ limits given in the COLR prevents power ~~peaking~~ that would exceed the ~~loss of coolant accident (LOCA)~~ linear heat rate (LHR) limits derived from the analysis of the ECCS.

<INSERT B3.2-38B>

by the LOCA LHR figure

The LOCA-limited LHR bounds the fuel centerline melt LHR limit. Thus compliance with the LOCA-limited LHR ensures compliance with the fuel centerline melt LHR

generation rates

-limited

31

The ~~$F^N_{\Delta H}$ limit~~ is a specified acceptable fuel design limit that preserves the initial conditions for the limiting loss of flow transient. ~~$F^N_{\Delta H}$ is defined as the ratio of the integral of linear power along the fuel rod on which the minimum departure from nucleate boiling ratio (DNBR) occurs to the average integrated rod power.~~ ~~Because $F^N_{\Delta H}$ is a ratio of integrated powers, it is related to the maximum total power produced in a fuel rod.~~ Operation within the ~~$F^N_{\Delta H}$~~ limits given in the COLR prevents ~~departure from nucleate boiling (DNB)~~ during a postulated loss of forced reactor coolant flow accident.

DNBR-limited LHR

<INSERT B 3.2-38C>

DNBR-limited LHR limits

Measurement of the core power peaking factors using the Incore Detector System to obtain a three dimensional power distribution map provides direct confirmation that ~~$F_Q(Z)$ and $F^N_{\Delta H}$~~ are within their limits, and may be used to verify that the ~~power peaking factors~~ remain bounded when one or more normal operating parameters exceed their limits.

LHRs

Core local LHRs

(continued)

<INSERT B 3.2-38A>

This is accomplished by limiting the local linear heat rate (LHR) to three general constraints: 1) the LHR may not exceed a value that results in fuel centerline melt, 2) the LHR may not exceed a value that would result in peak cladding temperatures of greater than 2200°F during a loss of coolant accident (LOCA), and 3) the LHR may not exceed a value that would result in the minimum departure from nucleate boiling ratio (DNBR) dropping below the specified acceptable fuel design limits in the event of the limiting loss of flow transient.

<INSERT B 3.2-38B>

The LOCA-limited LHR is dependent upon core axial location and fuel batch design. The LOCA-limited LHR may be designated as LHR in units of kW/ft or as a power peaking factor. When expressed as a power peaking factor, the LOCA-limited LHR is designated as $F_Q(Z)$.

<INSERT B 3.2-38C>

DNBR is defined as the ratio of the heat flux that would cause departure from nucleate boiling (DNB) at a particular core location to the actual heat flux at that core location. The DNBR-limited LHR represents the linear power generation rate along the fuel rod on which the minimum DNBR occurs. Compliance with this LHR value may be accomplished: 1) by correlating the LHR at the limiting location to the critical heat flux (expressed as a LHR) for the limiting location, 2) by correlating the LHR to DNBR or DNB margin for the limiting location, or 3) by correlating the LHR to a power peaking factor (designated as $F^N_{\Delta H}$) for the limiting location.

The relationship between the observable parameters of neutron power, reactor coolant flow, temperature and pressure and the critical heat flux, DNBR or DNB margin is provided through the use of a critical heat flux correlation. The critical heat flux correlations used to determine the critical heat flux for uniform and non-uniform heat flux distributions are described in the Bases for Safety Limit 2.1.1.

BASES (continued)

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LOCA-limited LHR limits

APPLICABLE SAFETY ANALYSES

The limits on $F_Q(Z)$ are determined by the ECCS analysis in order to limit peak cladding temperatures to 2200°F during a LOCA. The maximum acceptable cladding temperature is specified by 10 CFR 50.46 (Ref. 1). Higher cladding temperatures could cause severe cladding failure by oxidation due to a Zircaloy water reaction. Other criteria must also be met (e.g., maximum cladding oxidation, maximum hydrogen generation, coolable geometry, and long term cooling). However, peak cladding temperature is usually most limiting.

DNBR-limited LHR

The limits on $F^N_{\Delta H}$ provide protection from DNB during a limiting loss of flow transient. Proximity to the DNB condition is expressed by the DNBR, defined as the ratio of the ~~cladding surface~~ heat flux ~~required to~~ cause DNB to the actual ~~cladding surface~~ heat flux. The minimum DNBR value during both normal operation and anticipated transients is limited to the DNBR correlation limit for the particular fuel design in use, and is accepted as an appropriate margin to DNB. The DNBR correlation limit ensures that there is at least 95% probability at the 95% confidence level (the 95/95 DNB criterion) that the hot fuel rod in the core does not experience DNB.

That would at a particular core location

at that core location

The critical heat flux correlations used to determine the critical heat flux for uniform and non-uniform heat flux distributions are described in the Bases for SL 2.1.1.

This LCO precludes core power distributions that violate the following fuel design criteria:

a. During a large break LOCA, peak cladding temperature must not exceed 2200°F (Ref. 1).

b. During a loss of forced reactor coolant flow accident, there must be at least 95% probability at the 95% confidence level (the 95/95 DNB criterion) that the hot fuel rod in the core does not experience a DNB condition.

The reload safety evaluation analysis determines limits on global core parameters that characterize the core power distribution. The primary parameters used to monitor and control the core power distribution are the regulating rod position, the APSR position, the AXIAL POWER IMBALANCE, and the QPT. These parameters are normally used to monitor and control the core power distribution because their measurements are continuously observable. Limits are placed on these parameters to ensure that the core power peaking factors remain bounded during operation in MODE 1 with THERMAL Power greater than 20% RTP. Nuclear

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(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

design model calculational uncertainty, manufacturing tolerances (e.g., the engineering hot channel factor), effects of fuel densification and rod bow, and modeling simplifications (such as treatment of the spacer grid effects) are accommodated as necessary through use of peaking augmentation factors in the reload safety evaluation analysis (Ref. 2). edit 31

LHR limitations ~~$F_Q(Z)$ and $F^N_{\Delta H}$~~ satisfy Criterion 2 of ~~the NRC Policy Statement~~ 10 CFR 50.36 (Ref. 3). 10

LCO

This LCO for ~~the~~ power peaking ~~factors $F_Q(Z)$ and $F^N_{\Delta H}$~~ LHR ensures that the core operates within the bounds assumed for the ECCS and thermal hydraulic analyses. Verification that LHR is ~~$F_Q(Z)$ and $F^N_{\Delta H}$ are~~ within the limits of this LCO as specified in the COLR allows continued operation ~~at THERMAL POWER~~ when the Required Actions of LCO 3.1.4, "CONTROL ROD Group Alignment Limits," LCO 3.2.1, "Regulating Rod Insertion Limits," LCO 3.2.2, "AXIAL POWER SHAPING ROD Insertion Limits," LCO 3.2.3, "AXIAL POWER IMBALANCE Operating Limits," and LCO 3.2.4, "QUADRANT POWER TILT," are entered. Conservative THERMAL POWER reductions are required if the limits on LHR ~~$F_Q(Z)$ and $F^N_{\Delta H}$~~ are exceeded. Verification that LHR is ~~$F_Q(Z)$ and $F^N_{\Delta H}$ are~~ within limits is also required during MODE 1 PHYSICS TESTS per LCO 3.1.8, "PHYSICS TESTS Exceptions—MODE 1." 31

Measurement uncertainties are applied when LHR is ~~$F_Q(Z)$ and $F^N_{\Delta H}$ are~~ determined using the Incore Detector System. The measurement uncertainties applied to the measured values ~~of $F_Q(Z)$ and $F^N_{\Delta H}$~~ account for uncertainties in observability and instrument string signal processing.

APPLICABILITY

In MODE 1 with THERMAL POWER > 20% RTP, the limits on LHR ~~$F_Q(Z)$ and $F^N_{\Delta H}$~~ must be maintained in order to prevent the core power distribution from exceeding the limits assumed in the analyses of the LOCA and loss of forced reactor coolant flow accidents. In MODE 1 with THERMAL POWER ≤ 20% RTP and in ~~In~~ MODES 2, 3, 4, 5, and 6, this LCO is not applicable because the reactor has insufficient stored energy in the fuel or energy being transferred to the coolant to require a limit on the distribution of core power. The minimum THERMAL POWER level of 20% was chosen based on the ability of the incore detector system to satisfactorily obtain meaningful power distribution data. 23 31 23 23

(continued)

BASES (continued)

ACTIONS

LHRs, DNBRs and

ing

as an LHR, DNBR, margin to DNB or as power peaking factors

<Insert B3.2-41A>

The operator must take care in interpreting the relationship of the power peaking factors ~~$F_Q(Z)$ and $F^N_{\Delta H}$~~ to their limits. ~~Limit values of $F_Q(Z)$ and $F^N_{\Delta H}$ in the COLR may be expressed in either LHR or in peaking units. Because $F_Q(Z)$ and $F^N_{\Delta H}$ are power peaking factors, constant LHR is maintained as THERMAL POWER is reduced, thereby allowing power peaking to be increased in inverse proportion to THERMAL POWER.~~

~~Therefore, the $F_Q(Z)$ and $F^N_{\Delta H}$ limits increase as THERMAL POWER decreases (assuming $F_Q(Z)$ and $F^N_{\Delta H}$ are expressed in peaking units) so that a constant LHR limit is maintained.~~ (3)

A.1

the LHR

When ~~$F_Q(Z)$~~ is determined not to be within its specified limit as determined by a three dimensional power distribution map, a THERMAL POWER reduction is taken to reduce the ~~maximum~~ limiting LHR in the core. ~~Design calculations have verified that a conservative THERMAL POWER reduction is 1% RTP or more for each 1% by which $F_Q(Z)$ exceeds its limit (Ref. [ ]).~~ The Completion Time of ~~15 minutes~~ 2 hours provides an acceptable time to reduce power in an orderly manner and without allowing the ~~plant~~ unit to remain in an unacceptable condition for an extended period of time.

edit

~~A.2~~

~~Power operation is allowed to continue by Required Action A.1 if THERMAL POWER is reduced by 1% RTP or more from the ALLOWABLE THERMAL POWER for each 1% by which $F_Q(Z)$ exceeds its limit. The same reduction in nuclear overpower trip setpoint and nuclear overpower based on the Reactor Coolant System (RCS) flow and the AXIAL POWER IMBALANCE trip setpoint is required for each 1% by which $F_Q(Z)$ is in excess of its limit. These reductions maintain both core protection and OPERABILITY margin at the reduced THERMAL POWER. The required Completion Time of 8 hours is reasonable based on the low probability of an accident occurring in this short time period and the number of steps required to complete the Required Action.~~ (3)

(continued)

<INSERT B 3.2-41A>

When expressed as power peaking factors, the value must be adjusted in inverse proportion to the THERMAL POWER level of the core as the power is reduced from RTP. Thus, the allowable peaking factors will increase as THERMAL POWER decreases.

BASES

ACTIONS (continued)

A.3

Continued operation with $F_Q(Z)$ exceeding its limit is not permitted, because the initial conditions assumed in the accident analyses are no longer valid. The required Completion Time of 24 hours to restore $F_Q(Z)$ within its limits at the reduced THERMAL POWER level is reasonable based on the low probability of a limiting event occurring simultaneously with $F_Q(Z)$ exceeding its limit. In addition, it precludes long term depletion with local LHRs higher than the limiting values, and limits the potential for inducing an adverse perturbation in the axial xenon distribution.

B.1

When $F^N_{\Delta H}$ is determined not to be within its acceptable limit as determined by a three dimensional power distribution map, a THERMAL POWER reduction is taken to reduce the maximum LHR in the core. The parameter RH by which THERMAL POWER is decreased per 1% increase in $F^N_{\Delta H}$ above the limit has been verified to be conservative by design calculations, and is defined in the COLR. The parameter RH is the inverse of the increase in $F^N_{\Delta H}$ allowed as THERMAL POWER decreases by 1% RTP, and is based on an analysis of the DNBR during the limiting loss of forced reactor coolant flow transient from various initial THERMAL POWER levels. The required Completion Time of 15 minutes is reasonable for the operator to take the actions necessary to reduce the unit power. 31

B.2

When a decrease in THERMAL POWER is required because $F^N_{\Delta H}$ has exceeded its limit, Required Action B.2 requires reduction of the high flux trip setpoint and the nuclear overpower based on RCS flow and AXIAL POWER IMBALANCE trip setpoint. The amount of reduction of these trip setpoints is governed by the same factor (RH(%) for each 1% that $F^N_{\Delta H}$ exceeds its limit) that determines the THERMAL POWER reduction. This process maintains core protection by providing margin to the trip setpoints at the reduced THERMAL POWER similar to that at RTP. The parameter RH is specified in the COLR. The required Completion Time of 8 hours is reasonable based on the low probability of an accident occurring in this short

(continued)

BASES

ACTIONS

B.2 (continued)

time period and the number of steps required to complete this Action.

B.3

Continued operation with $F^N_{\Delta H}$ exceeding its limit is not permitted, because the initial conditions assumed in the accident analyses are no longer valid. The required Completion Time of 24 hours to restore $F^N_{\Delta H}$ within its limit at the reduced THERMAL POWER level is reasonable based on the low probability of a limiting event occurring simultaneously with $F^N_{\Delta H}$ exceeding its limit. In addition, this Completion Time precludes long term depletion with an unacceptably high local power and limits the potential for inducing an adverse perturbation in the radial xenon distribution.

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B.1 ~~C.1~~

If a THERMAL POWER reduction is not sufficient to restore $F_Q(Z)$ or $F^N_{\Delta H}$ within its limit (i.e., the Required Actions and associated Completion Times for Condition A or B are not met), then THERMAL POWER operation should cease. The reactor is placed in MODE 2 in which this LCO does not apply. The required Completion Time of 2 hours is a reasonable amount of time for the operator to reduce THERMAL POWER in an orderly manner and without challenging plant systems.

be significantly reduced

with THERMAL POWER ≤ 20% RTP where

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unit

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SURVEILLANCE REQUIREMENTS

SR 3.2.5.1

power distribution

Core monitoring is performed using the Incore Detector System to obtain a three dimensional power distribution map. Maximum values of $F_Q(Z)$ and $F^N_{\Delta H}$ obtained from this map may then be compared with the $F_Q(Z)$ and limits in the COLR to verify that the limits have not been exceeded. Measurement of the core power peaking factors in this manner may be used to verify that the measured values of $F_Q(Z)$ and $F^N_{\Delta H}$ remain within their specified limits when one or more of the limits specified by LCO 3.1.4, LCO 3.2.1, LCO 3.2.2, LCO 3.2.3, or

LHR

distribution

LHR

Minimum DNBR values or DNB margins determined from the core power distribution mapping may also be compared to their limits or correlated to LHR values to verify that the limits have not been exceeded.

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(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.2.5.1 (continued)

LCO 3.2.4 is exceeded, or when LCO 3.1.8 is applicable. If the local LHRs ~~$F_Q(Z)$ and $F_{\Delta H}^N$~~ remain within their limits when one or more of these parameters exceed their limits, operation at THERMAL POWER may continue because the true initial conditions (the power Core distribution ~~peaking factors~~) remain within their specified limits.

Because the limits on ~~$F_Q(Z)$ and $F_{\Delta H}^N$~~ LHR are preserved when the parameters specified by LCO 3.1.4, LCO 3.2.1, LCO 3.2.2, LCO 3.2.3, and LCO 3.2.4 are within their limits, a Note is provided in the SR to indicate that monitoring of the ~~power peaking factors~~ Core local LHRs is required only when complying with the Required Actions of these LCOs and when LCO 3.1.8 is applicable.

Frequencies for monitoring of the ~~power peaking factors~~ Core local LHRs are specified in the Action statements of the individual LCOs. These Frequencies are reasonable based on the low probability of a limiting event occurring simultaneously with ~~either $F_Q(Z)$ or $F_{\Delta H}^N$~~ LHR exceeding its limit, and they provide sufficient time for the operator to obtain a power distribution map from the Incore Detector System. Indefinite THERMAL POWER operation in a Required Action of LCO 3.1.4, LCO 3.2.1, LCO 3.2.2, LCO 3.2.3, or LCO 3.2.4 is ~~not~~ permitted, because the core local LHRs are within ~~in order to limit the potential for exceeding both the power peaking factors assumed in the accident analyses due to operation with unanalyzed core power distributions and spatial xenon distributions beyond their~~ analyzed ~~ranges~~ core power distributions and spatial xenon distributions. 31

REFERENCES

1. 10 CFR 50.46.
2. BAW-10179 P-A, "Safety Criteria and Methodology for Acceptable Cycle Reload Analyses," Rev 2, October 1997. 31
3. 10 CFR 50.36. 10

This Section Addresses the Following Specifications:

NUREG-1430	ANO-1 ITS	Title
3.3.1	3.3.1	Reactor Protection System (RPS) Instrumentation
3.3.2	3.3.2	Reactor Protection System (RPS) Manual Reactor Trip
3.3.3	3.3.3	Reactor Protection System (RPS) - Reactor Trip Module (RTM)
3.3.4	3.3.4	CONTROL ROD Drive (CRD) Trip Devices
3.3.9	3.3.9	Source Range Neutron Flux
3.3.10	3.3.10	Intermediate Range Neutron Flux

3.3 INSTRUMENTATION

3.3.1 Reactor Protection System (RPS) Instrumentation

LCO 3.3.1 Four channels of RPS instrumentation for each Function in Table 3.3.1-1 shall be OPERABLE.

APPLICABILITY: According to Table 3.3.1-1.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One channel inoperable.	A.1 Place channel in bypass or trip. OR A.2 Prevent bypass of remaining channels.	1 hour 1 hour
B. Two channels inoperable.	B.1 Place one channel in trip. AND B.2.1 Place second channel in bypass. OR B.2.2 Prevent bypass of remaining channels.	1 hour 1 hour 1 hour
C. Three or more channels inoperable. OR Required Action and associated Completion Time of Condition A or B not met.	C.1 Enter the Condition referenced in Table 3.3.1-1 for the Function.	Immediately

CONDITION	REQUIRED ACTION	COMPLETION TIME
D. As required by Required Action C.1 and referenced in Table 3.3.1-1.	D.1 Be in MODE 3. AND D.2 Open all control rod drive (CRD) trip breakers.	6 hours 6 hours
E. As required by Required Action C.1 and referenced in Table 3.3.1-1.	E.1 Open all CRD trip breakers.	6 hours
F. As required by Required Action C.1 and referenced in Table 3.3.1-1.	F.1 Reduce THERMAL POWER < 45% RTP.	6 hours
G. As required by Required Action C.1 and referenced in Table 3.3.1-1.	G.1 Reduce THERMAL POWER < 10% RTP.	6 hours

SURVEILLANCE REQUIREMENTS

-----NOTE-----

Refer to Table 3.3.1-1 to determine which SRs apply to each RPS Function.

SURVEILLANCE		FREQUENCY
SR 3.3.1.1	Perform CHANNEL CHECK.	12 hours

SURVEILLANCE		FREQUENCY
SR 3.3.1.2	---NOTE--- Not required to be performed until 24 hours after THERMAL POWER is ≥ 20% RTP. --- Verify calorimetric heat balance is ≤ 2% RTP greater than power range channel output. Adjust power range channel output if calorimetric exceeds power range channel output by ≥ 2% RTP.	96 hours AND Once within 24 hours after a THERMAL POWER change of ≥ 10% RTP
SR 3.3.1.3	---NOTE--- Not required to be performed until 24 hours after THERMAL POWER is ≥ 20% RTP. --- Compare out of core measured AXIAL POWER IMBALANCE to incore measured AXIAL POWER IMBALANCE.	31 days
SR 3.3.1.4	Perform CHANNEL FUNCTIONAL TEST.	31 days
SR 3.3.1.5	---NOTE--- Not required to be performed until 24 hours after THERMAL POWER is ≥ 20% RTP. --- Calibrate the power range channels to the incore channels.	31 days
SR 3.3.1.6	---NOTE--- Neutron detectors are excluded from CHANNEL CALIBRATION. --- Perform CHANNEL CALIBRATION.	18 months

Table 3.3.1-1 (page 1 of 1)
Reactor Protection System Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	CONDITIONS REFERENCED FROM REQUIRED ACTION C.1	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE
1. Nuclear Overpower -				
a. High Setpoint	1,2[a],3[d]	D	SR 3.3.1.1 SR 3.3.1.2 SR 3.3.1.4 SR 3.3.1.5 SR 3.3.1.6	≤ 104.9% RTP
b. Low Setpoint	2[b],3[b] 4[b],5[b]	E	SR 3.3.1.1 SR 3.3.1.4 SR 3.3.1.6	≤ 5% RTP
2. RCS High Outlet Temperature	1,2	D	SR 3.3.1.1 SR 3.3.1.4 SR 3.3.1.6	≤ 618°F
3. RCS High Pressure	1,2[a],3[d]	D	SR 3.3.1.1 SR 3.3.1.4 SR 3.3.1.6	≤ 2355 psig
4. RCS Low Pressure	1,2[a]	D	SR 3.3.1.1 SR 3.3.1.4 SR 3.3.1.6	≥ 1800 psig
5. RCS Variable Low Pressure	1,2[a]	D	SR 3.3.1.1 SR 3.3.1.4 SR 3.3.1.6	As specified in the COLR
6. Reactor Building High Pressure	1,2,3[c]	D	SR 3.3.1.1 SR 3.3.1.4 SR 3.3.1.6	≤ 18.7 psia
7. Reactor Coolant Pump to Power	1,2[a]	D	SR 3.3.1.1 SR 3.3.1.4 SR 3.3.1.6	≤ 55% RTP with one pump operating in each loop.
8. Nuclear Overpower RCS Flow and Measured AXIAL POWER IMBALANCE	1,2[a]	D	SR 3.3.1.1 SR 3.3.1.3 SR 3.3.1.4 SR 3.3.1.5 SR 3.3.1.6	As specified in the COLR
9. Main Turbine Trip (Oil Pressure)	≥ 45% RTP	F	SR 3.3.1.1 SR 3.3.1.4 SR 3.3.1.6	≥ 40.5 psig
10. Loss of Main Feedwater Pumps (Control Oil Pressure)	≥ 10% RTP	G	SR 3.3.1.1 SR 3.3.1.4 SR 3.3.1.6	≥ 55.5 psig
11. Shutdown Bypass RCS High Pressure	2[b],3[b] 4[b],5[b]	E	SR 3.3.1.1 SR 3.3.1.4 SR 3.3.1.6	≤ 1720 psig

(a) When not in shutdown bypass operation.

(b) During shutdown bypass operation with any CRD trip breaker in the closed position and the CRD System capable of rod withdrawal.

(c) With any CRD trip breaker in the closed position and the CRD System capable of rod withdrawal.

(d) With any CRD trip breaker in the closed position, the CRD system capable of rod withdrawal, and not in shutdown bypass operation.

3.3 INSTRUMENTATION

3.3.2 Reactor Protection System (RPS) Manual Reactor Trip

LCO 3.3.2 The RPS Manual Reactor Trip Function shall be OPERABLE.

APPLICABILITY: MODES 1 and 2,
MODES 3, 4, and 5 with any control rod drive (CRD) trip breaker in the closed position and the CRD System capable of rod withdrawal.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Manual Reactor Trip Function inoperable.	A.1 Restore Function to OPERABLE status.	1 hour
B. Required Action and associated Completion Time not met in MODE 1, 2, or 3.	B.1 Be in MODE 3. AND B.2 Open all CRD trip breakers.	6 hours 6 hours
C. Required Action and associated Completion Time not met in MODE 4 or 5.	C.1 Open all CRD trip breakers.	6 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.3.2.1	Perform CHANNEL FUNCTIONAL TEST.	Once prior to each reactor startup if not performed within the previous 7 days

3.3 INSTRUMENTATION

3.3.3 Reactor Protection System (RPS) - Reactor Trip Module (RTM)

LCO 3.3.3 Four RTMs shall be OPERABLE.

APPLICABILITY: MODES 1 and 2,
MODES 3, 4, and 5 with any control rod drive (CRD) trip breaker in the closed position and the CRD System capable of rod withdrawal.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One RTM inoperable.	A.1.1 Open the associated CRD trip breaker.	1 hour
	<u>OR</u>	
	A.1.2 Remove power from the associated CRD trip breaker.	1 hour
	<u>AND</u>	
	A.2 Physically remove the inoperable RTM.	1 hour
B. Two or more RTMs inoperable in MODE 1, 2, or 3. <u>OR</u> Required Action and associated Completion Time not met in MODE 1, 2, or 3.	B.1 Be in MODE 3.	6 hours
	<u>AND</u>	
	B.2.1 Open all CRD trip breakers.	6 hours
	<u>OR</u>	
	B.2.2 Remove power from all CRD trip breakers.	6 hours

CONDITION	REQUIRED ACTION		COMPLETION TIME
C. Two or more RTMs inoperable in MODE 4 or 5. OR Required Action and associated Completion Time not met in MODE 4 or 5.	C.1	Open all CRD trip breakers.	6 hours
	OR		
	C.2	Remove power from all CRD trip breakers.	6 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.3.3.1	Perform CHANNEL FUNCTIONAL TEST.	92 days

3.3 INSTRUMENTATION

3.3.4 Control Rod Drive (CRD) Trip Devices

LCO 3.3.4 The following CRD trip devices shall be OPERABLE:

a. Two AC CRD trip breakers;

b. Two DC CRD trip breaker pairs; and

c. Eight electronic trip assembly (ETA) relays.

APPLICABILITY: MODES 1 and 2,
MODES 3, 4, and 5 with any CRD trip breaker in the closed position and the CRD System capable of rod withdrawal.

ACTIONS

---NOTE---

Separate Condition entry is allowed for each CRD trip device.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more CRD trip breaker(s) or breaker pair undervoltage or shunt trip Functions inoperable.	A.1 Open the CRD trip breaker. <u>OR</u> A.2 Remove power from the CRD trip breaker.	48 hours 48 hours
B. One or more CRD trip breaker(s) or breaker pair inoperable for reasons other than those in Condition A.	B.1 Open the CRD trip breaker. <u>OR</u> B.2 Remove power from the CRD trip breaker.	1 hour 1 hour

CONDITION	REQUIRED ACTION		COMPLETION TIME
C. One or more required ETA relays inoperable.	C.1	Transfer affected CONTROL ROD group to power supply with OPERABLE or open ETA relays.	1 hour
	<u>OR</u>		
	C.2	Transfer affected CONTROL ROD group to a DC hold power supply.	1 hour
	<u>OR</u>		
	C.3	Place the SCRs associated with the inoperable ETA relay in trip.	1 hour
	<u>OR</u>		1 hour
	C.4	Open corresponding AC CRD trip breaker.	
D. Required Action and associated Completion Time not met in MODE 1, 2, or 3.	D.1	Be in MODE 3.	6 hours
	<u>AND</u>		
	D.2.1	Open all CRD trip breakers.	6 hours
	<u>OR</u>		
	D.2.2	Remove power from all CRD trip breakers.	6 hours
E. Required Action and associated Completion Time not met in MODE 4 or 5.	E.1	Open all CRD trip breakers.	6 hours
	<u>OR</u>		
	E.2	Remove power from all CRD trip breakers.	6 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.3.4.1	Perform CHANNEL FUNCTIONAL TEST.	92 days

3.3 INSTRUMENTATION

3.3.9 Source Range Neutron Flux.3.9 Source Range Neutron Flux

LCO 3.3.9 One source range neutron flux channel shall be OPERABLE.

APPLICABILITY: MODES 2, 3, 4, and 5.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Required source range neutron flux channel inoperable with ≤ 1E-10 amp on the intermediate range neutron flux channel.	A.1 Suspend operations involving positive reactivity changes.	Immediately
	AND	
	A.2 Initiate action to insert all CONTROL RODS.	Immediately
	AND	
	A.3 Open control rod drive trip breakers.	1 hour
	AND	
	A.4 Verify SDM to be within the limit provided in the COLR.	1 hour AND Once per 12 hours thereafter
B. Required source range neutron flux channel inoperable with > 1E-10 amp on the intermediate range neutron flux channel.	B.1 Initiate action to restore required channel to OPERABLE status.	1 hour

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.3.9.1	Perform CHANNEL CHECK.	12 hours
SR 3.3.9.2	-----NOTE----- Neutron detectors are excluded from CHANNEL CALIBRATION. ----- Perform CHANNEL CALIBRATION.	18 months

3.3 INSTRUMENTATION

3.3.10 Intermediate Range Neutron Flux

LCO 3.3.10 One intermediate range neutron flux channel shall be OPERABLE.

APPLICABILITY: MODE 2,
MODES 3, 4, and 5 with any control rod drive (CRD) trip breaker in the closed position and the CRD System capable of rod withdrawal.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Required channel inoperable.	A.1 Suspend operations involving positive reactivity changes.	Immediately
	AND	
	A.2 Open CRD trip breakers.	1 hour

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.3.10.1	Perform CHANNEL CHECK.	12 hours
SR 3.3.10.2	Perform CHANNEL FUNCTIONAL TEST.	31 days
SR 3.3.10.3	-----NOTE----- Neutron detectors are excluded from CHANNEL CALIBRATION. ----- Perform CHANNEL CALIBRATION.	18 months

SURVEILLANCE		FREQUENCY
SR 3.3.10.4	Verify at least one decade overlap between source range and intermediate range neutron flux channels.	Once each reactor startup prior to source range counts exceeding 10^5 cps

B 3.3 INSTRUMENTATION

B 3.3.1 Reactor Protection System (RPS) Instrumentation

BASES

BACKGROUND

The RPS initiates a reactor trip, if necessary, to protect against violating the core fuel design limits and the Reactor Coolant System (RCS) pressure boundary during abnormalities. By tripping the reactor, the RPS also assists the Engineered Safety Feature (ESF) Systems in mitigating accidents.

The protection and monitoring systems have been designed to assure safe operation of the reactor. This is achieved by identifying limiting safety system settings (LSSS) in terms of parameters directly monitored by the RPS, as well as the LCOs and administrative controls on other parameters and equipment performance.

The LSSS, defined in this Specification as the Allowable Value, in conjunction with the LCOs and administrative controls, establishes the threshold for protective system action to prevent exceeding specified acceptable limits during Design Basis Accidents (DBAs). Acceptable consequences for accidents are that the offsite dose shall be maintained within 10 CFR 100 limits or other limits approved by the NRC.

During abnormalities, one or more of the following the limits is maintained:

a. For accidents other than locked rotor, the departure from nucleate boiling ratio (DNBR) shall be maintained above the Safety Limit (SL) value. For the locked rotor accident, the minimum DNBR shall not be less than the applicable critical heat flux correlation limit, or fuel cladding shall be shown to experience no significant temperature excursions;

b. Fuel centerline temperature shall be maintained below the SL value;

c. The RCS pressure SL of 2750 psig shall not be exceeded; and

d. Reactor power shall not exceed 112% RTP.

Maintaining the parameters within the above values ensures that the offsite dose will be within the 10 CFR 100 criteria during abnormalities.

RPS Overview

The RPS consists of four separate redundant protection channels that receive inputs of neutron flux, RCS pressure, RCS flow, reactor outlet temperature, RCS pump status, reactor building (RB) pressure, main feedwater (MFW) pump turbine status, and main turbine status.

Figure 7.1, SAR, Chapter 7 (Ref. 1), shows the arrangement of the RPS protection channels. A protection channel is composed of measurement channels, a manual trip channel, a reactor trip module (RTM), and control rod drive (CRD) trip devices. LCO 3.3.1 provides requirements for the individual measurement channels. These channels encompass all equipment and electronics from the point at which the measured parameter is sensed through the bistable relay contacts in the trip string. LCO 3.3.2, "Reactor Protection System (RPS) Manual Reactor Trip," LCO 3.3.3, "Reactor Protection System (RPS) - Reactor Trip Module (RTM)," and LCO 3.3.4, "Control Rod Drive (CRD) Trip Devices," discuss the remaining RPS elements.

The RPS instrumentation measures critical unit parameters and compares these to predetermined setpoints. If the setpoint is exceeded, a channel trip signal is generated. The generation of trip signals in any two of the four RPS channels will result in the trip of the reactor.

The Reactor Trip System (RTS) contains multiple CRD trip devices, two AC trip breakers, and two DC trip breaker pairs that provide a path for power to the CRD System. In addition to the safety rods, the power for the regulating rods and APSRs may be interrupted by the electronic trip assembly (ETA) relays. The system has two separate paths (or channels), with each path having either two breakers in series or a breaker and an ETA relay controlled silicon controlled rectifier (SCR) in series. Each path provides independent power to the CRDs. Either path can provide sufficient power to operate all CRDs. Two separate power paths to the CRDs ensure that a single failure that opens one path will not cause an unwanted reactor trip.

The RPS consists of four independent protection channels, each containing an RTM. Each RTM receives signals from its own measurement channels that indicate a protection channel trip is required. The RTM transmits this signal to its own two-out-of-four trip logic and to the two-out-of-four logic of the RTMs in the other three RPS channels. Whenever any two RPS channels trip, the RTM in each channel actuates to remove 120 VAC power from its associated CRD trip breaker.

The reactor is tripped by opening circuit breakers and de-energizing ETA relays that interrupt the control power supply to the CRDs. Six breakers are installed to increase reliability and allow testing of the trip system. A one-out-of-two taken twice logic is used to interrupt power to the rods.

The RPS has two manual bypasses: a shutdown bypass and a channel bypass. Shutdown bypass allows the withdrawal of safety rods to provide the availability of rapidly insertable negative reactivity during unit cooldowns or heatups. Channel bypass is typically used for maintenance and testing. Test circuits in the trip strings allow testing of RPS trip Functions. Also, an automatic bypass is provided at low power levels for the Main Turbine Trip and the Loss of Main Feedwater Pump Functions.

The RPS receives input from the instrumentation channels discussed next. The specific relationship between measurement channels and protection channels differs from parameter to parameter.

These arrangements and the relationship of instrumentation channels to trip Functions are discussed below to assist in understanding the overall effect of instrumentation channel failure.

Power Range Nuclear Instrumentation

Power Range Nuclear Instrumentation channels provide inputs to the following RPS trip Functions:

1. Nuclear Overpower
    a. Nuclear Overpower - High Setpoint;
    b. Nuclear Overpower - Low Setpoint;
7. Reactor Coolant Pump to Power;
8. Nuclear Overpower RCS Flow and Measured AXIAL POWER IMBALANCE (Power Imbalance Flow);
9. Main Turbine Trip (Oil Pressure); and
10. Loss of Main Feedwater Pumps (Control Oil Pressure).

The Main Turbine Trip and Loss of Main Feedwater Pumps Functions utilize the Power Range Nuclear Instrumentation only for enabling/disabling the operating bypass at low power levels.

The power range instrumentation has four linear channels, one for each core quadrant. Each channel feeds one RPS protection channel. Each channel originates in a detector assembly containing two uncompensated ion chambers. The ion chambers are positioned to represent the top half and bottom half of the core. The individual currents from the chambers are fed to individual linear amplifiers. The summation of the top and bottom is the total reactor power. The top minus the bottom neutron signal is the measured AXIAL POWER IMBALANCE of the reactor core.

Reactor Outlet Temperature

The Reactor Outlet Temperature provides input to the following Functions:

2. Reactor Outlet High Temperature; and

5. RCS Variable Low Pressure.

The Reactor Outlet Temperature is measured by two resistance elements in each hot leg, for a total of four. One temperature detector is associated with each protection channel.

Reactor Coolant System Pressure

The Reactor Coolant System Pressure provides input to the following Functions:

3. RCS High Pressure;

4. RCS Low Pressure;

5. RCS Variable Low Pressure; and

11. Shutdown Bypass RCS High Pressure.

The RPS inputs of reactor coolant pressure are provided by two pressure transmitters in each hot leg, for a total of four. One sensor is associated with each protection channel.

Reactor Building Pressure

The Reactor Building Pressure measurements provide input only to the Reactor Building High Pressure trip, Function 6. There are four RB High Pressure sensors, one associated with each protection channel.

Reactor Coolant Pump Power Monitoring

Reactor coolant pump power monitors are inputs to the Reactor Coolant Pump to Power trip, Function 7. Each RCP's operating current is measured by a current transformer providing the current input to the associated RCP underpower relay, and the bus voltage is measured by a potential transformer providing the voltage input to the associated RCP underpower relays. Each RCP underpower relay provides individual RCP status to each protection channel.

Reactor Coolant System Flow

The Reactor Coolant System Flow measurements are an input to the Nuclear Overpower RCS Flow and Measured AXIAL POWER IMBALANCE trip, Function 8. The reactor coolant flow inputs to the RPS are provided by eight differential pressure transmitters, four on each loop, which measure flow through calibrated flow tubes. One flow input in each loop is associated with each protection channel.

Main Turbine Automatic Stop Oil Pressure

Main Turbine Automatic Stop Oil Pressure is an input to the Main Turbine Trip (Oil Pressure) reactor trip, Function 9. Each of the four protection channels receives turbine status information from one of four pressure switches monitoring main turbine automatic stop oil pressure. Contact buffers in each protection channel continuously monitor the status of the contact inputs and initiate an RPS trip when a main turbine trip is indicated.

Feedwater Pump Control Oil Pressure

Feedwater Pump Control Oil Pressure is an input to the Loss of Main Feedwater Pumps (Control Oil Pressure) trip, Function 10. Control oil pressure is measured by four switches on each feedwater pump. One switch on each pump is associated with each protection channel.

RPS Bypasses

The RPS is designed with two types of manual bypasses: channel bypass and shutdown bypass.

Channel bypass provides a method of placing all Functions in one RPS protection channel in a bypassed condition, and shutdown bypass provides a method of leaving the safety rods withdrawn during cooldown and depressurization of the RCS. Each bypass is discussed next.

Channel Bypass

A channel bypass provision is provided to allow for maintenance and testing of the RPS. The use of channel bypass keeps the protection channel trip relay energized regardless of the status of the instrumentation channel bistable relay contacts. To place a protection channel in channel bypass, the key switch must be operated, and the other three channels must not be in channel bypass. This is ensured by contacts from the other channels being in series with the channel bypass relay. If any contact is open, the second channel cannot be bypassed. When the bypass relay is energized, the bypass contact closes, maintaining the channel trip relay in an energized condition. An indicator light remains lit while the channel is in bypass. All RPS trips are reduced to a two-out-of-three logic in channel bypass. Only one channel bypass key is accessible for use in the control room.

Shutdown Bypass

During unit cooldown, it is allowable to leave some safety rods withdrawn to provide shutdown capabilities in the event of unusual positive reactivity additions (moderator dilution, etc.).

However, the unit is also depressurized as coolant temperature is decreased. If the safety rods are withdrawn and coolant pressure is decreased, an RCS Low Pressure trip will occur at 1800 psig and the rods will fall into the core. To avoid this, the protection system allows the operator to bypass the low pressure trip and maintain shutdown capabilities. During the cooldown and depressurization, the safety rods are inserted prior to the low pressure trip of 1800 psig. The RCS pressure is decreased to less than 1720 psig, then each RPS channel is placed in shutdown bypass.

When an RPS channel is placed in shutdown bypass, the RCS Low Pressure trip, Nuclear Overpower RCS Flow and Measured AXIAL POWER IMBALANCE trip, Reactor Coolant Pump to Power trip, and the RCS Variable Low Pressure trip, are bypassed and a RCS High Pressure, ≤ 1720 psig trip and a Nuclear Overpower Low Setpoint trip, ≤ 5% RTP, are inserted. The operator can now withdraw the safety rods for additional rapidly insertable negative reactivity.

The insertion of the high pressure trip with a trip setpoint of ≤ 1720 psig prevents operation at normal system pressure, approximately 2155 psig, with a portion of the RPS bypassed, and ensures that the bypass is removed prior to normal operation. When the RCS pressure is increased during a unit heatup, the safety rods are inserted prior to reaching 1720 psig. The shutdown bypass is removed, which returns the RPS to normal, and system pressure is increased to greater than 1800 psig. All or some of the safety rods are then withdrawn and normally remain at the full out condition for the rest of the heatup.

The insertion of the Nuclear Overpower Low Setpoint Trip provides a backup to the Shutdown Bypass RCS High Pressure trip while preventing the generation of any significant amount of power.

Module Interlock and Test Trip Relay

Each channel and each trip module is capable of being individually tested. When a module is placed into the test mode, it causes the test trip relay to open and to indicate an RPS channel trip. Under normal conditions, the channel to be tested is placed in bypass before a module is tested.

Trip Setpoints/Allowable Value

The trip setpoints are the nominal values at which the bistables are set. Any bistable is considered to be properly adjusted when the "as left" value is within the band for CHANNEL CALIBRATION accuracy.

The trip setpoints used in the bistables are based on the analytical limits used in the safety analysis described in SAR, Chapter 14 and Chapter 3A (Ref. 2). The selection of these trip setpoints is such that adequate protection is provided when appropriate sensor and processing time delays are taken into account. To allow for calibration tolerances, instrumentation uncertainties, instrument drift, and environment errors, the Allowable Values specified in Table 3.3.1-1 are equal to or conservatively adjusted with respect to the analytical limits. Guidance used to calculate the uncertainty associated with the trip setpoints is provided in Instrument Loop Error Analysis and Setpoint Methodology Manual, Design Guide, IDG-001 (Ref. 3). The explicit uncertainties are addressed in the individual design calculations as required. The trip setpoint entered into the bistable may be more conservative than that specified by the Allowable Value to account for changes in instrument error detectable by a CHANNEL FUNCTIONAL TEST. A channel is inoperable if its as-found trip setpoint is not within its required Allowable Value.

Setpoints in accordance with the Allowable Value ensure that the limits of Chapter 2.0, "Safety Limits," in the Technical Specifications are not violated during abnormalities and that the consequences of DBAs will be acceptable, providing the unit is operated from within the LCOs at the onset of the abnormality or DBA and the equipment functions as analyzed. Note that in LCO 3.3.1 the Allowable Values listed in Table 3.3.1-1 are the LSSS.

Each channel can be tested online to verify that the signal and trip setpoint are within the specified allowance requirements of approved calibration procedures. Once a designated channel is taken out of service for testing, a simulated signal may be injected in place of the field instrument signal. The process equipment for the channel may then be tested, verified, and calibrated.

APPLICABLE SAFETY ANALYSES, LCO, AND APPLICABILITY

Analyzed accidents and transients can be detected by one or more RPS Functions. The accident analysis contained in the SAR, Chapter 14 and Chapter 3A (Ref. 2), takes credit for most RPS trip Functions. Functions not specifically credited in the accident analysis were qualitatively credited in the NRC staff approved licensing basis for the unit. These Functions are high RB pressure, high RCS temperature, turbine trip, loss of main feedwater, the shutdown bypass nuclear overpower low setpoint, and shutdown bypass high pressure. These Functions may provide protection for conditions that do not require dynamic transient analysis to demonstrate Function performance. These Functions also serve as backups to Functions that were credited in the safety analysis.

The LCO requires all instrumentation performing an RPS Function to be OPERABLE. Failure of any instrument renders the affected channel(s) inoperable. The four channels of each Function in Table 3.3.1-1 of the RPS instrumentation shall be OPERABLE during its specified Applicability to ensure that a reactor trip will be actuated if needed. Additionally, during shutdown bypass with any CRD trip breaker closed, the applicable RPS Functions must also be OPERABLE. This ensures the capability to trip the withdrawn CONTROL RODS exists at all times that rod motion is possible. The trip Function channels specified in Table 3.3.1-1 are considered OPERABLE when all channel components necessary to provide a reactor trip are functional and in service for the required MODE or Other Specified Condition listed in Table 3.3.1-1.

Required Actions allow maintenance (protection channel) bypass of individual channels, but the bypass activates interlocks that prevent operation with a second channel bypass. Bypass effectively places the unit in a two-out-of-three logic configuration that can still initiate a reactor trip, even with a single failure within the system.

Only the Allowable Values are specified for each RPS trip Function in the LCO. Trip setpoints are specified in the setpoint calculations or calibration procedures. The setpoints are selected such that the setpoint measured by CHANNEL FUNCTIONAL TESTS is not expected to exceed the Allowable Value if the bistable is performing as required.

For most RPS Functions, the Allowable Value is to ensure that the departure from nucleate boiling (DNB) or RCS pressure SLs are not challenged. Cycle specific figures for use during operation are contained in the COLR.

Certain RPS trips function to indirectly protect the SLs by detecting specific conditions that do not immediately challenge SLs but will eventually lead to challenge if no action is taken. These trips function to minimize the consequences of unit transients caused by the specific conditions. The Allowable Value for these Functions is selected at the specified deviation from normal values that will indicate the condition, without risking spurious trips due to normal fluctuations in the measured parameter.

The Allowable Values for bypass removal Functions are stated in the Applicable MODE or Other Specified Condition column of Table 3.3.1-1.

The safety analyses applicable to each RPS Function are discussed next.

1. Nuclear Overpower

a. Nuclear Overpower - High Setpoint

The Nuclear Overpower - High Setpoint trip provides protection for the design thermal overpower condition based on the measured out of core fast neutron leakage flux.

The Nuclear Overpower - High Setpoint trip initiates a reactor trip when the neutron power reaches a predefined setpoint at the design overpower limit. Because THERMAL POWER lags the neutron power, tripping when the neutron power reaches the design overpower will limit THERMAL POWER to a maximum value of the design overpower.

Thus, the Nuclear Overpower - High Setpoint trip protects against violation of the DNBR and fuel centerline melt SLs. However, the RCS Variable Low Pressure, and Nuclear Overpower RCS Flow and Measured AXIAL POWER IMBALANCE, also provide protection. The role of the Nuclear Overpower - High Setpoint trip is to limit reactor THERMAL POWER below the highest power at which the other two trips are known to provide protection.

The Nuclear Overpower - High Setpoint trip also provides transient protection for rapid positive reactivity excursions during power operations. These events include the rod withdrawal accident, the rod ejection accident, and the steam line break accident. By providing a trip during these events, the Nuclear Overpower - High Setpoint trip protects the unit from excessive power levels and also serves to reduce reactor power to prevent violation of the RCS pressure SL.

Rod withdrawal accident analyses cover a large spectrum of reactivity insertion rates (rod worths), which exhibit slow and rapid rates of power increases. At high reactivity insertion rates, the Nuclear Overpower - High Setpoint trip provides the primary protection. At low reactivity insertion rates, the high pressure trip provides primary protection.

The specified Allowable Value is selected to initiate a trip at or before reactor power exceeds the highest point at which the RCS Variable Low Pressure and the Nuclear Overpower RCS Flow and Measured AXIAL POWER IMBALANCE trips are analyzed to provide protection against DNB and fuel centerline melt. The Allowable Value does not account for harsh environment induced errors, because the trip will actuate prior to degraded environmental conditions being reached.

b. Nuclear Overpower - Low Setpoint

While in shutdown bypass, the Nuclear Overpower - Low Setpoint is instated with a trip setpoint of ≤ 5% RTP. The low power setpoint, in conjunction with the Shutdown Bypass RCS High Pressure setpoint, protect the unit from excessive power conditions when other RPS trips are bypassed.

The Allowable Value was chosen to be as low as practical and still lie within the range of the out of core instrumentation.

2. Reactor Outlet High Temperature

The Reactor Outlet High Temperature trip, in conjunction with the RCS Low Pressure and RCS Variable Low Pressure trips, provides protection for the DNBR SL. A trip is initiated whenever the reactor outlet temperature approaches the conditions necessary for DNB. Portions of each Reactor Outlet High Temperature trip channel are common with the RCS Variable Low Pressure trip. The Reactor Outlet High Temperature trip provides steady state protection for the DNBR SL.

The Reactor Outlet High Temperature trip limits the maximum RCS temperature to below the highest value for which DNB protection by the Variable Low Pressure trip is ensured. The trip setpoint Allowable Value is selected to initiate a trip before hot leg temperatures reach the point beyond which the RCS Low Pressure and Variable Low Pressure trips are analyzed. Above the high temperature trip, the variable low pressure trip need not provide protection, because the unit would have tripped already. The Allowable Value does not reflect errors induced by harsh environmental conditions that the equipment is expected to experience because the trip will actuate prior to degraded environmental conditions being reached.

3. RCS High Pressure

The RCS High Pressure trip works in conjunction with the pressurizer safety valves to prevent RCS overpressurization, thereby protecting the RCS High Pressure SL.

The RCS High Pressure trip has been credited in the accident analysis calculations for slow positive reactivity insertion transients (rod withdrawal accidents and moderator dilution). The rod withdrawal accidents cover a large spectrum of reactivity insertion rates and rod worths that exhibit slow and rapid rates of power increases. At high reactivity insertion rates, the Nuclear Overpower - High Setpoint trip provides the primary protection. At low reactivity insertion rates, the RCS High Pressure trip provides the primary protection.

The Allowable Value is selected such that the RCS High Pressure SL is not exceeded during steady state operation or slow power increasing transients. The Allowable Value does not reflect errors induced by harsh environmental conditions because the trip will actuate prior to degraded environmental conditions being reached.

4. RCS Low Pressure

The RCS Low Pressure trip, in conjunction with the Reactor Outlet High Temperature and Variable Low Pressure trips, provides protection for the DNBR SL. A trip is initiated prior to reactor outlet temperature exceeding the conditions necessary for DNB. The RCS Low Pressure trip provides the DNB low pressure limit for the RCS Variable Low Pressure trip.

The RCS Low Pressure Allowable Value is selected to initiate a reactor trip before RCS pressure is reduced below the lowest point at which the RCS Variable Low Pressure trip is analyzed. The RCS Low Pressure trip provides protection for primary system depressurization events and has been credited in the accident analysis calculations for small break loss of coolant accidents (LOCAs). Consequently, harsh RB conditions created by small break LOCAs can affect performance of the RCS pressure sensors and transmitters. Therefore, degraded environmental conditions are considered in the Allowable Value determination.

5. RCS Variable Low Pressure

The RCS Variable Low Pressure trip, in conjunction with the Reactor Outlet High Temperature and RCS Low Pressure trips, provides protection for the DNBR SL. A trip is initiated prior to the system parameters of pressure and temperature exceeding the conditions necessary for DNB. The RCS Variable Low Pressure trip provides a floating low pressure trip based on the reactor outlet temperature expressed in degrees Fahrenheit within the range specified by the Reactor Outlet High Temperature and RCS Low Pressure trips.

The RCS Variable Low Pressure Allowable Value is selected to initiate a trip prior to temperature and pressure exceeding the conditions necessary for DNB while operating in a temperature pressure region constrained by the low pressure and high temperature trips. The RCS Variable Low Pressure trip is not assumed for transient protection in the unit safety analysis. Therefore, the Allowable Value does not account for errors induced by a harsh RB environment.

6. Reactor Building High Pressure

The Reactor Building High Pressure trip provides an early indication of a high energy line break (HELB) inside the RB. By detecting changes in the RB pressure, the RPS can provide a reactor trip before the other system parameters have varied significantly. Thus, this trip acts to minimize accident consequences.

The Allowable Value for RB High Pressure trip is set at the lowest value consistent with avoiding spurious trips during normal operation. Even in the case where this trip is a backup for other RPS trips for LOCA or MSLB, it is assumed to occur before degraded building conditions have an appreciable effect on RB High Pressure trip components. Therefore, determination of the Allowable Value does not account for errors induced by a harsh environment.

7. Reactor Coolant Pump to Power

The Reactor Coolant Pump to Power trip provides protection for changes in the reactor coolant flow due to the loss of multiple RCPs. Because the flow reduction lags loss of power indications due to the inertia of the RCPs, the trip initiates protective action earlier than a trip based on a measured flow signal.

The trip also prevents operation with both pumps in either coolant loop tripped. Under these conditions, core flow and core fluid mixing may be insufficient for adequate heat transfer. Thus, the Reactor Coolant Pump to Power trip functions to protect the DNBR and fuel centerline temperature SLs.

The Reactor Coolant Pump to Power trip has been credited in the accident analysis calculations for the loss of four RCPs. The trip also provides protection for the loss of a pump or pumps which would result in both pumps in a single steam generator loop being tripped.

The Allowable Value for the Reactor Coolant Pump to Power trip setpoint is selected to prevent normal power operation unless at least one RCP is operating in each loop. RCP status is monitored by power transducers associated with each pump. These relays indicate a loss of an RCP on underpower. The underpower Allowable Value is selected to reliably trip on loss of voltage to the RCPs. Neither the reactor power nor the pump power Allowable Value account for instrumentation errors caused by harsh environments because the trip Function is not required to respond to events that could create harsh environments around the equipment.

8. Nuclear Overpower RCS Flow and Measured AXIAL POWER IMBALANCE

The Nuclear Overpower RCS Flow and Measured AXIAL POWER IMBALANCE trip provides steady state protection for the reactor core SLs. A reactor trip is initiated prior to the core power, AXIAL POWER IMBALANCE, and reactor coolant flow conditions exceeding the DNB or fuel centerline temperature limits.

This trip supplements the protection provided by the Reactor Coolant Pump to Power trip, through the power to flow ratio, for loss of reactor coolant flow events. The power to flow ratio provides direct protection for the limiting loss of flow transient which is the loss of two RCPs from four pump operation. The imbalance portion of the trip is credited for steady state protection only.

The power to flow ratio of the Nuclear Overpower RCS Flow and Measured AXIAL POWER IMBALANCE trip also provides steady state protection to prevent reactor power from exceeding the allowable power when the primary system is operating with two or three pump flow. Thus, the power to flow ratio prevents overpower conditions similar to the Nuclear Overpower trip. This protection ensures that during reduced flow conditions the core power is maintained below that required to begin DNB.

The Allowable Value is selected to ensure that a trip occurs prior to core power, axial power peaking, and reactor coolant flow conditions reaching DNB or fuel centerline temperature limits. The Allowable Value for this Function is given in the unit COLR because the cycle specific core peaking changes affect the Allowable Value.

9. Main Turbine Trip (Oil Pressure)

The Main Turbine Trip Function trips the reactor when the main turbine is tripped at high power levels. The Main Turbine Trip Function provides an early reactor trip in anticipation of the loss of heat sink associated with a turbine trip. The Main Turbine Trip Function was added to the B&W designed units in accordance with NUREG-0737 (Ref. 4) following the Three Mile Island Unit 2 accident. The trip lowers the probability of an RCS electromatic relief valve (ERV) actuation for turbine trip cases.

Each of the four turbine oil pressure switches feeds one of the four protection channels through a buffer that continuously monitors the status of the contacts. Therefore, failure of any pressure switch affects only one protection channel.

For the Main Turbine Trip (Oil Pressure) bistable, the Allowable Value of ≥ 40.5 psig is selected to provide a trip whenever main turbine oil pressure drops below the normal operating range. To ensure that the trip is enabled as required by the LCO, the reactor power bypass is set to a value of < 45% RTP. The turbine trip is not required to protect against events that can create a harsh environment in the turbine building. Therefore, errors induced by harsh environments are not included in the determination of the setpoint Allowable Value.

10. Loss of Main Feedwater Pumps (Control Oil Pressure)

The Loss of Main Feedwater Pumps (Control Oil Pressure) trip provides a reactor trip at high power levels when both MFW pumps are tripped. The trip provides an early reactor trip in anticipation of the loss of heat sink associated with a loss of main feedwater. This trip was added in accordance with NUREG-0737 (Ref. 4) following the Three Mile Island Unit 2 accident. This trip provides a reactor trip for a loss of main feedwater to minimize challenges to the ERV.

For the feedwater pump control oil pressure bistable, the Allowable Value of ≥ 55.5 psig is selected to provide a trip whenever feedwater pump control oil pressure drops below the normal operating range. To ensure that the trip is enabled as required by the LCO, the reactor power bypass is set to a value of < 10% RTP. The Loss of Main Feedwater Pumps (Control Oil Pressure) trip is not required to protect against events that can create a harsh environment in the turbine building. Therefore, errors caused by harsh environments are not included in the determination of the setpoint Allowable Value.

11. Shutdown Bypass RCS High Pressure

The RPS Shutdown Bypass is provided to allow for withdrawing the CONTROL RODS while operating below the normal RCS Low Pressure trip setpoint. The shutdown bypass allows the operator to withdraw the safety groups of CONTROL RODS. This makes their negative reactivity available to terminate inadvertent reactivity excursions. Because the shutdown bypass high pressure trip setpoint is below the normal RCS low pressure trip setpoint, the reactor must be tripped while passing between these two setpoints. This ensures that RPS trips cannot be bypassed unless the CONTROL RODS are all inserted.

Accidents analyzed in the SAR, Chapter 14 and Chapter 3A (Ref. 2), do not include events that occur during shutdown bypass operation.

During shutdown bypass operation with the Shutdown Bypass RCS High Pressure trip active with a setpoint of ≤ 1720 psig and the Nuclear Overpower - Low Setpoint active with a setpoint of ≤ 5% RTP, the trips listed below are bypassed..

4. RCS Low Pressure;

5. RCS Variable Low Pressure;

7. Reactor Coolant Pump to Power; and

8. Nuclear Overpower RCS Flow and Measured AXIAL POWER IMBALANCE.

The Shutdown Bypass Nuclear Overpower - Low Setpoint Allowable Value is selected to initiate a trip before producing significant THERMAL POWER.

General Discussion

In MODES 1 and 2, the RPS satisfies Criterion 3 of 10 CFR 50.36 (Ref. 5). In MODES 3, 4, and 5 with any CRD trip breaker in the closed position and the CRD System capable of rod withdrawal, the RPS satisfies Criterion 4 of 10 CFR 50.36.

In MODE 1; in MODE 2, when not operating in shutdown bypass; and in MODE 3, when not operating in shutdown bypass but with any CRD trip breaker in the closed position and the CRD system capable of rod withdrawal, the following trips are required to be OPERABLE. These trips function to ensure that any withdrawn CONTROL RODS can be automatically inserted to make or maintain the reactor subcritical.

1.a. Nuclear Overpower-High Setpoint; and

3. RCS High Pressure.

In MODES 1 and 2, the following trips are required to the OPERABLE. These trips function as primary or as back-up trips to ensure that any withdrawn CONTROL RODS can be automatically inserted to make or maintain the reactor subcritical.

2. Reactor Outlet High Temperature; and

6. Reactor Building High Pressure.

In addition, Function 6, Reactor Building High Pressure, is required to be OPERABLE in MODE 3, whenever any CRD trip breaker is closed and the CRD system is capable of rod withdrawal. In this MODE, this Function serves purely as a back-up to other required Functions.

In MODE 1 and in MODE 2, when not in shutdown bypass operation, the following trips are required to be OPERABLE. These Functions operate to ensure that any withdrawn CONTROL RODS can be automatically inserted to make or maintain the reactor subcritical. These functions are all bypassed when the channel is placed in a shutdown bypass condition. Therefore, they are not required to be OPERABLE during shutdown bypass operation.

4. RCS Low Pressure;

5. RCS Variable Low Pressure;

7. Reactor Coolant Pump to Power; and

8. Nuclear Overpower RCS Flow and Measured AXIAL POWER IMBALANCE.

Two Functions are required to be OPERABLE only during portions of MODE 1. These are the Main Turbine Trip (Oil Pressure) and the Loss of Main Feedwater Pumps (Control Oil Pressure) trip. These Functions are required to be OPERABLE at ≥ 45% RTP and ≥ 10% RTP, respectively. Analyses presented in BAW-1893 (Ref. 6) have shown that for operation below these power levels, these trips are not necessary to minimize challenges to the ERV as required by NUREG-0737 (Ref. 4).

Because the safety function of the RPS is to trip the CONTROL RODS, the RPS is not required to be OPERABLE in MODE 3, 4, or 5, if either the reactor trip breakers are open, or the CRD System is incapable of rod withdrawal. Similarly, the RPS is not required to be OPERABLE in MODE 6 because the CONTROL RODS are normally decoupled from the CRDs.

However, during shutdown bypass operation, in MODE 2, 3, 4, or 5, the Shutdown Bypass RCS High Pressure and Nuclear Overpower - Low setpoint trips are required to be OPERABLE if the CRD trip breakers are closed and the CRD System is capable of rod withdrawal. Under these conditions, the Shutdown Bypass RCS High Pressure and Nuclear Overpower - Low setpoint trips sufficiently reduce the potential for conditions that could challenge SLs.

ACTIONS

Conditions A, B, and C are applicable to all RPS protection Functions. If a channel's trip setpoint is found nonconservative with respect to the required Allowable Value in Table 3.3.1-1, or the transmitter, instrument loop, signal processing electronics or bistable is found inoperable, the channel must be declared inoperable and all applicable Conditions entered immediately.

A.1 and A.2

If one or more Functions in one protection channel become inoperable, the affected protection channel must be placed in bypass or trip, or the bypass of the remaining channels prevented. If the channel is bypassed, all RPS Functions are placed in a two-out-of-three logic configuration and the bypass of any other channel is prevented. In this configuration, the RPS can still perform its safety function in the presence of a random failure of any single channel. Alternatively, the inoperable channel can be placed in trip. Tripping the affected protection channel places all RPS Functions in a one-out-of-three configuration.

Another option is to maintain the channel, which contains one or more inoperable Functions, in an untripped and unbypassed state. In this case, bypass of the remaining three channels must be prevented. This is accomplished by tagging them, under administrative controls, to prevent their being bypassed. This option assumes that the inoperability of the Function(s) does not require the channel containing the inoperable Function(s) to remain in a tripped condition, and that the channel contains other Functions which remain OPERABLE.

By maintaining the channel in an untripped and unbypassed state, the inoperable Function (s) are in a two-out-of-three logic configuration. This configuration is equivalent to bypassing the channel. However, by maintaining the channel in an untripped and unbypassed condition, the OPERABLE Functions within that channel remain in service in a normal two-out-of-four logic configuration.

Operation in these configurations may continue indefinitely because the RPS is capable of performing its trip Function in the presence of any single random failure. The 1 hour Completion Time is sufficient to perform Required Action A.1 or Required Action A.2.

B.1, B.2.1, and B.2.2

For Required Action B.1 and Required Action B.2, if one or more Functions in two protection channels become inoperable, one of two inoperable protection channels must be placed in trip. The second inoperable channel may be bypassed or may be maintained in an untripped and unbypassed condition. If the channel is not bypassed, bypass of the remaining channels must be prevented. This is accomplished by tagging them, under administrative controls, to prevent their being bypassed. This option assumes that the inoperability of the Function(s) in the second channel does not require that channel to remain in a tripped condition, and that the channel contains one or more Function(s) which remains OPERABLE. These Required Actions place all RPS Functions in either a one-out-of-two or one-out-of-three logic configuration. In either of these configurations, the RPS can still perform its safety functions in the presence of a random failure of any single channel. The 1 hour Completion Time is sufficient time to perform Required Action B.1, Required Action B.2.1, and Required Action B.2.2.

C.1

Required Action C.1 directs entry into the appropriate Condition referenced in Table 3.3.1-1. The applicable Condition referenced in the table is Function dependent. If the Required Action and associated Completion Time of Condition A or B are not met or if more than two channels are inoperable, Condition C is entered to provide for transfer to the appropriate subsequent Condition.

D.1 and D.2

If Required Action C.1 and Table 3.3.1-1 direct entry into Condition D, the unit must be brought to a MODE in which the specified RPS trip Functions are not required to be OPERABLE. The allowed Completion Time of 6 hours is reasonable, based on operating experience, to reach MODE 3 from full power conditions in an orderly manner and to open all CRD trip breakers without challenging unit systems.

E.1

If Required Action C.1 and Table 3.3.1-1 direct entry into Condition E, the unit must be brought to a MODE in which the specified RPS trip Functions are not required to be OPERABLE. To achieve this status, all CRD trip breakers must be opened. The allowed Completion Time of 6 hours is reasonable, based on operating experience, to open CRD trip breakers without challenging unit systems.

F.1

If Required Action C.1 and Table 3.3.1-1 direct entry into Condition F, the unit must be brought to a MODE in which the specified RPS trip Function is not required to be OPERABLE. To achieve this status, THERMAL POWER must be reduced to < 45% RTP. The allowed Completion Time of 6 hours is reasonable, based on operating experience, to reach 45% RTP from full power conditions in an orderly manner without challenging unit systems.

G.1

If Required Action C.1 and Table 3.3.1-1 direct entry into Condition G, the unit must be brought to a MODE in which the specified RPS trip Function is not required to be OPERABLE. To achieve this status, THERMAL POWER must be reduced to < 10% RTP. The allowed Completion Time of 6 hours is reasonable, based on operating experience, to reach 10% RTP from full power conditions in an orderly manner without challenging unit systems.

SURVEILLANCE REQUIREMENTS

The SRs for each RPS Function are identified by the SRs column of Table 3.3.1-1 for that Function. Most Functions are subject to CHANNEL CHECK, CHANNEL FUNCTIONAL TEST, and CHANNEL CALIBRATION testing.

The SRs are modified by a Note which directs the reader to Table 3.3.1-1 to determine the correct SRs to perform for each RPS Function.

SR 3.3.1.1

Performance of the CHANNEL CHECK once every 12 hours provides reasonable assurance of prompt identification of a gross failure of instrumentation. A CHANNEL CHECK is normally a comparison of the parameter indicated on one channel to the same parameter on other channels. It is based on the assumption that instrument channels monitoring the same parameter should read approximately the same value. Significant deviations between two instrument channels could be an indication of excessive instrument drift in one of the channels or of something even more serious. CHANNEL CHECK will detect gross channel failure; therefore, it is key in verifying that the instrumentation continues to operate properly between each CHANNEL CALIBRATION.

Agreement criteria are determined by the unit staff based on a combination of factors including channel instrument uncertainties. If a channel is outside the criteria, it may be an indication that the transmitter or the signal processing equipment has drifted outside its limit. If the channels are within the criteria, it is an indication that the channels are OPERABLE. If the channels are normally off scale during times when surveillance is required, the CHANNEL CHECK will only verify that they are off scale in the same direction. Off scale low current loop channels are, where practical, verified to be reading at the bottom of the range and not failed downscale.

The Frequency is based on operating experience that demonstrates channel failure is rare. Since the probability of two random failures in redundant channels in any 12 hour period is extremely low, the CHANNEL CHECK minimizes the chance of loss of protective function due to failure of redundant channels. The CHANNEL CHECK supplements less formal but more frequent checks of channel OPERABILITY during normal operational use of the displays associated with the LCO's required channels.

For Functions that trip on a combination of several measurements, such as the Nuclear Overpower RCS Flow and Measured AXIAL POWER IMBALANCE Function, the CHANNEL CHECK must be performed on each input.

SR 3.3.1.2

This SR is the performance of a heat balance calibration for the power range channels every 96 hours and once within 24 hours after a THERMAL POWER change of ≥ 10% RTP in one direction, when reactor power is ≥ 20% RTP. The heat balance calibration consists of a comparison of the results of the calorimetric with the power range channel output. The outputs of the power range channels are calibrated to the calorimetric. If the calorimetric exceeds the Nuclear Instrumentation System (NIS) channel output by ≥ 2% RTP, the NIS channel is not declared inoperable but must be adjusted. If the NIS channel cannot be properly adjusted, the channel is declared inoperable. A Note clarifies that this Surveillance is required only if reactor power is ≥ 20% RTP and that 24 hours is allowed for performing the first Surveillance after reaching 20% RTP.

Two calorimetric calculations are routinely performed. One relies upon primary system parameters and the other relies upon secondary system parameters. The primary calorimetric is generally less accurate than the secondary calorimetric at higher power levels and more accurate at lower power levels. For comparison to the nuclear instrumentation, between 0 and 15% power, only the primary calorimetric (heat balance) is considered. From 15 to 100% power the calorimetric is weighted linearly with only the secondary heat balance being considered at 100% power.

The power range channel's output shall be adjusted consistent with the calorimetric results if the calorimetric exceeds the power range channel's output by ≥ 2% RTP. The value of 2% is adequate because this value is assumed in the safety analyses of SAR, Chapter 14 (Ref. 2). These checks and, if necessary, the adjustment of the power range channels ensure that channel accuracy is maintained within the analyzed error margins. The 96 hour Frequency is adequate, based on unit operating experience, which demonstrates the change in the difference between the power range indication and the calorimetric results rarely exceeds 2% in any 96 hour period. Furthermore, the control room operators monitor redundant indications and alarms to detect deviations in channel outputs.

SR 3.3.1.3

A comparison of power range nuclear instrumentation channels against incore detectors shall be performed at a 31 day Frequency when reactor power is ≥ 20% RTP. A Note clarifies that 24 hours is allowed for performing the first Surveillance after reaching 20% RTP. If the absolute difference between the power range and incore AXIAL POWER IMBALANCE measurements is greater than the procedural limit, the power range channel is not inoperable, but a CHANNEL CALIBRATION that adjusts the measured imbalance to agree with the incore measurements is necessary. If the power range channel cannot be properly recalibrated, the channel is declared inoperable. The calculation of the Allowable Value envelope assumes a difference in out of core to incore AXIAL POWER IMBALANCE measurements of 2.5%. Additional inaccuracies beyond those that are measured are also included in the setpoint envelope calculation. The 31 day Frequency is adequate, considering that long term drift of the excore linear amplifiers is small and burnup of the detectors is slow. Also, the excore readings are a strong function of the power produced in the peripheral fuel bundles, and do not represent an integrated reading across the core. The slow changes in neutron flux during the fuel cycle can also be detected at this interval.

SR 3.3.1.4

A CHANNEL FUNCTIONAL TEST is performed to ensure that the entire channel will perform the intended function. Setpoints must be found within the Allowable Values specified in Table 3.3.1-1. Any setpoint adjustment shall be consistent with the assumptions of the current setpoint analysis.

The Frequency of 31 days is based on operating experience, which has demonstrated through high reliability of the instrumentation, that failure of more than one channel, of a given Function, in any 31 day interval is rare.

Testing in accordance with this SR is normally performed on a rotational basis, with one channel being tested each week. Testing one channel each week reduces the probability of an undetected failure existing within the system and minimizes the likelihood of the same systematic test errors being introduced into each redundant channel.

SR 3.3.1.5

This SR calibrates the power range (excore) channels to the incore channels every 31 days. This calibration adjusts the power range channel output to the calorimetric heat balance coincident with the imbalance output being calibrated to the imbalance condition predicted by the incore neutron detector system.

The 31 day Frequency specified for the Nuclear Overpower trip string is consistent with the drift assumptions made in the calculation of the setpoint. Furthermore, operating experience shows the reliability of the trip string is acceptable when calibrated on this interval. A Note clarifies that 24 hours is allowed for performing the first Surveillance after reaching 20% RTP.

SR 3.3.1.6

A Note to the Surveillance indicates that neutron detectors are excluded from CHANNEL CALIBRATION. This Note is necessary because of the difficulty in generating an appropriate detector input signal. Excluding the detectors is acceptable because the principles of detector operation ensure a virtually instantaneous response.

A CHANNEL CALIBRATION is a complete check of the instrument channel, including the sensor. The test verifies that the channel responds to the measured parameter within the necessary range and accuracy. CHANNEL CALIBRATION leaves the channel adjusted to account for instrument drift to ensure that the instrument channel remains operational between successive tests. CHANNEL CALIBRATION shall find that instrument errors are within the assumptions of the setpoint analysis. CHANNEL CALIBRATION must be performed consistent with the assumptions of the setpoint analysis. Whenever a sensing element is replaced, the next required CHANNEL CALIBRATION of the resistance temperature (RTD) sensors is accomplished by an inplace cross calibration that compares the other sensing elements with the recently installed sensing element.

The Frequency is justified by the assumption of at least an 18 month calibration interval in the determination of the allowable magnitude of equipment drift in the setpoint analysis.

REFERENCES

1. SAR, Chapter 7.
2. SAR, Chapter 14 and Chapter 3A.
3. Instrument Loop Error Analysis and Setpoint Methodology Manual, Design Guide, IDG-001.
4. NUREG-0737, "Clarification of TMI Action Plan Requirements," November 1980.
5. 10 CFR 50.36.
6. BAW-1893, "Basis for Raising Arming Threshold for Anticipatory Reactor Trip on Turbine Trip," October 1985.

B 3.3 INSTRUMENTATION

B 3.3.2 Reactor Protection System (RPS) Manual Reactor Trip

BASES

BACKGROUND

The RPS Manual Reactor Trip provides the operator with the capability to trip the reactor from the control room in the absence of, or coincident with, any other trip condition. Manual trip is provided by a trip push button on the main control board. This push button operates four electrically independent switches. This trip is independent of the automatic trip system. As shown in Figure 7.1, SAR, Chapter 7 (Ref. 1), control power for the control rod drive (CRD) breakers and electronic trip assembly (ETA) relays comes from the reactor trip modules (RTMs). The manual trip switches are located between the RTM output and the breaker undervoltage coils, breaker undervoltage relays, and ETA relays. The switches also initiate actuation of the breaker shunt trip mechanisms. These are separate switches which are actuated through a mechanical linkage from a single push button. Opening of the switches opens the circuits to the breakers, tripping them.

APPLICABLE SAFETY ANALYSES

The Manual Reactor Trip ensures that the control room operator can initiate a reactor trip at any time. The Manual Reactor Trip Function is required as a backup to the automatic trip functions.

Operating experience has shown the Manual Reactor Trip Function to be significant to public health and safety, and therefore satisfy Criterion 4 of 10 CFR 50.36 (Ref. 2).

LCO

The LCO on the RPS Manual Reactor Trip requires that the trip shall be OPERABLE whenever the reactor is critical or any time any CRD breaker is closed and rods are capable of being withdrawn, including shutdown bypass. This enables the operator to terminate any reactivity excursion that in the operator's judgment requires protective action, even if no automatic trip condition exists.

The Manual Reactor Trip Function is composed of four electrically independent trip switches sharing a common mechanical push button.

APPLICABILITY

The Manual Reactor Trip Function is required to be OPERABLE in MODES 1 and 2. It is also required to be OPERABLE in MODES 3, 4, and 5 if any CRD trip breaker is in the closed position and if the CRD System is capable of rod withdrawal. The primary safety function of the RPS is to trip the CONTROL RODS; therefore, the Manual Reactor Trip Function is not needed in MODE 3, 4, or 5 if the reactor trip breakers are open or if the CRD System is incapable of rod withdrawal. Similarly, the RPS Manual Reactor Trip is not needed in MODE 6 because the CONTROL RODS are normally decoupled from the CRDs.

ACTIONS

A.1

Condition A applies when the Manual Reactor Trip Function is found inoperable. One hour is allowed to restore the Function to OPERABLE status. The automatic functions and various alternative manual trip methods, such as removing power to the RTMs, are still available. The 1 hour Completion Time is sufficient time to correct minor problems.

B.1 and B.2

If the Required Action and associated Completion Time are not met in MODE 1, 2, or 3, the unit must be placed in a MODE in which manual trip is not required. Required Action B.1 and Required Action B.2 place the unit in at least MODE 3 with all CRD trip breakers open within 6 hours. The allowed 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.

C.1

If the Required Action and associated Completion Time are not met in MODE 4 or 5, the unit must be placed in a MODE in which manual trip is not required. To achieve this status, all CRD trip breakers must be opened. The allowed Completion Time of 6 hours is reasonable, based on operating experience, to open all CRD trip breakers without challenging unit systems.

SURVEILLANCE REQUIREMENTS

SR 3.3.2.1

This SR requires the performance of a CHANNEL FUNCTIONAL TEST of the Manual Reactor Trip Function. This test verifies the OPERABILITY of the Manual Reactor Trip by actuation of the CRD trip breakers. The Frequency shall be once prior to each reactor startup if not performed within the preceding 7 days to ensure the OPERABILITY of the Manual Reactor Trip Function prior to achieving criticality. The Frequency was developed in consideration that this Surveillance is only performed during a unit outage.

REFERENCES

1. SAR, Chapter 7.
2. 10 CFR 50.36.

B 3.3 INSTRUMENTATION

B 3.3.3 Reactor Protection System (RPS) - Reactor Trip Module (RTM)

BASES

BACKGROUND

The RPS consists of four independent protection channels, each containing an RTM. Figure 7.1, SAR, Chapter 7 (Ref. 1), shows a typical RPS protection channel and the relationship of the RTM to the RPS instrumentation, manual trip, and control rod drive (CRD) trip devices. The RTM receives bistable trip signals from the functions in its own channel and channel trip signals from the other three RPS - RTMs. The RTM provides these signals to its own two-out-of-four trip logic and transmits its own channel trip signal to the two-out-of-four logic of the RTMs in the other three RPS channels. Whenever any two RPS channels transmit channel trip signals, the RTM logic in each channel actuates to remove 120 VAC power from its associated CRD trip device.

The RPS trip scheme consists of series contacts that are operated by bistables. During normal unit operations, all contacts are closed and the RTM channel trip relay remains energized. However, if any trip parameter exceeds its setpoint, its associated contact opens, which de-energizes the channel trip relay.

When an RTM channel trip relay de-energizes, several things occur:

a. Each of the four (4) output logic relays "informs" its associated RPS channel that a reactor trip signal has occurred in the tripped RPS channel;

b. The contacts in the trip device circuitry, powered by the tripped channel, open, but the trip device remains energized through the closed contacts from the other RTMs. (This condition exists in each RPS - RTM. Each RPS - RTM controls power to a trip device.); and

c. The contact in parallel with the channel reset switch opens and the trip is sealed in. To re-energize the channel trip relay, the channel reset switch must be depressed after the trip condition has cleared.

When the second RPS channel senses a reactor trip condition, the output logic relays for the second channel de-energize and open contacts that supply power to the trip devices. With contacts opened by two separate RPS channels, power to the trip devices is interrupted and the CONTROL RODS fall into the core.

A minimum of two out of four RTMs must sense a trip condition to cause a reactor trip. Also, two channel trips caused by different trip functions can result in a reactor trip.

APPLICABLE SAFETY ANALYSES

Accident analyses rely on a reactor trip for protection of reactor core integrity, reactor coolant pressure boundary integrity, and reactor building OPERABILITY. A reactor trip must occur when needed to prevent accident conditions from exceeding those calculated in the accident analyses. More detailed descriptions of the applicable accident analyses are found in the bases for each of the RPS trip Functions in LCO 3.3.1, "Reactor Protection System (RPS) Instrumentation."

In MODES 1 and 2, the RTMs satisfy Criterion 3 of 10 CFR 50.36 (Ref. 2). In MODES 3, 4, and 5 with any CRD trip breaker in the closed position and the CRD System capable of rod withdrawal, the RTMs satisfy Criterion 4 of 10 CFR 50.36.

LCO

The RTM LCO requires all four RTMs to be OPERABLE. Failure of any RTM renders a portion of the RPS inoperable.

To be considered OPERABLE, an RTM must be able to receive and interpret trip signals from its own and other OPERABLE RPS channels and to open its associated trip device.

The requirement for four channels to be OPERABLE ensures that a minimum of two RPS channels will remain OPERABLE if a single failure has occurred in one channel and if a second channel has been bypassed. This two-out-of-four trip logic also ensures that a single RPS channel failure will not cause an unwanted reactor trip. Violation of this LCO could result in a trip signal not causing a reactor trip when needed.

APPLICABILITY

The RTMs are required to be OPERABLE in MODES 1 and 2. They are also required to be OPERABLE in MODES 3, 4, and 5 if any CRD trip breakers are in the closed position and the CRD System is capable of rod withdrawal. The RTMs are designed to ensure a reactor trip would occur, if needed. This need may exist in any of these MODES; therefore, the RTMs must be OPERABLE.

ACTIONS

<u>A.1.1, A.1.2, and A.2</u>

When an RTM is inoperable, the associated CRD trip breaker must then be placed in a condition that is equivalent to a tripped condition for the RTM. Required

Action A.1.1 or Required Action A.1.2 requires this either by opening (tripping) the CRD trip breaker or by removing power to the CRD trip device. Tripping one RTM or removing power opens one set of CRD trip devices. Power to hold up CONTROL RODS is still provided via the parallel CRD trip device(s). Therefore, a reactor trip will not occur until a second protection channel trips.

To ensure the trip signal is registered in the other channels, Required Action A.2 requires that the inoperable RTM be removed from the cabinet. This action causes the electrical interlocks to indicate a tripped channel in the remaining three RTMs. Operation in this condition is allowed indefinitely because the actions put the RPS into a one-out-of-three configuration. The 1 hour Completion Time is sufficient time to perform the Required Actions.

B.1, B.2.1, and B.2.2

Condition B applies if two or more RTMs are inoperable in MODE 1, 2, or 3, or if the Required Actions and associated Completion Time of Condition A are not met in MODE 1, 2, or 3. In this case, the unit must be placed in a MODE in which the LCO does not apply. This is done by placing the unit in at least MODE 3 with all CRD trip breakers open or with power to all CRD trip breakers removed within 6 hours. The allowed 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.

C.1 and C.2

Condition C applies if two or more RTMs are inoperable in MODE 4 or 5, or if the Required Actions and associated Completion Times are not met in MODE 4 or 5. In this case, the unit must be placed in a MODE in which the LCO does not apply. This is done by opening all CRD trip breakers or removing power from all CRD trip breakers. The allowed Completion Time of 6 hours is reasonable, based on operating experience, to open all CRD trip breakers or remove all power to the CRD System without challenging unit systems.

SURVEILLANCE REQUIREMENTS

SR 3.3.3.1

The SRs include performance of a CHANNEL FUNCTIONAL TEST every 92 days. This test shall verify the OPERABILITY of the RTM and its ability to receive and properly respond to channel trip and reactor trip signals.

The Frequency of 92 days is based on operating experience, which has demonstrated through high reliability of the instrumentation, that failure of more than one RTM in any 92 day interval is rare (Ref. 3).

Testing in accordance with this SR is normally performed on a rotational basis, with one RTM being tested each 23 days. Testing one RTM each 23 days reduces the probability of an undetected failure existing within the system and minimizes the likelihood of the same systematic test errors being introduced into each redundant RTM.

REFERENCES

1. SAR, Chapter 7.

2. 10 CFR 50.36.

3. BAW-10167A, "Justification for Increasing the Reactor Trip System On-Line Test Intervals," Supplement 3, "Justification for Increasing the Trip Device Test Interval," February 1998.

B 3.3 INSTRUMENTATION

B 3.3.4 Control Rod Drive (CRD) Trip Devices

BASES

BACKGROUND

The Reactor Protection System (RPS) contains multiple CRD trip devices: two AC trip breakers, two DC trip breaker pairs, and ten electronic trip assembly (ETA) relays. The system has two separate paths (or channels), with each path having one AC breaker either in series with a pair of DC breakers or functionally in series with five ETA relays in parallel. Each path provides independent power to the CRDs. Either path can provide sufficient power to operate the entire CRD System.

Figure 7-10 SAR, Chapter 7 (Ref. 1), illustrates the configuration of CRD trip devices. To trip the reactor, power to the CRDs must be removed. Loss of power causes the CRD's mechanisms to release the CONTROL RODS, which then fall by gravity into the core.

Power to CRDs is supplied from two separate unit sources through the AC trip circuit breakers. These breakers are designated A and B, and their undervoltage and shunt trip coils are controlled by RPS channels A and B, respectively. From the circuit breakers, the CRD power travels through voltage regulators and stepdown transformers. These devices in turn supply redundant buses that feed the DC holding power supplies and the regulating rod, APSR and auxiliary power supplies.

The DC holding power supplies rectify the AC input and supply power to hold the safety rods in their fully withdrawn position. One of the redundant power sources supplies phase A; the other, phase CC. Either phase being energized is sufficient to hold the rod. Two breakers are located on the output of each power supply. Each breaker controls half of the power to two of the four safety rod groups. The undervoltage and shunt trip coils on the two circuit breakers on the output of one of the power supplies is controlled by RPS channel C. The other two breakers are controlled by RPS channel D.

In addition to the DC holding power supplies, the redundant buses also supply power to the regulating rod, APSR and auxiliary power supplies. These power supplies contain silicon controlled rectifiers (SCRs), which are gated on and off to provide power to, and remove power from, the phases of the CRD mechanisms. The gating control signal for these SCRs is supplied through the closed contacts of the ETA relays. These contacts are referred to as E and F contactors, and are controlled by the C and D RPS channels, respectively.

The AC breaker and DC breakers, or gated SCRs, are in series in one of the power supplies; whereas, the redundant AC breaker and DC breakers or gated SCRs are in series in the other power supply to the CONTROL RODS. The logic required to cause a reactor trip is the opening of a circuit breaker or ETA relay in each of the redundant power supplies. (The pair of DC circuit breakers on the output of the power supply are treated as one breaker.) This is known as a one-out-of-two taken twice logic. The following examples illustrate the operation of the reactor trip circuit breakers.

a. If the A AC circuit breaker opens:

   1. the input power to associated DC power supply is lost, and

   2. the SCR supply from the associated power source is lost.

b. If the D DC circuit breaker(s) and F contactors open:

   1. the output of the redundant DC power supply is lost and the safety rods de-energize, and

   2. when the F contactor opens, SCR gating power is lost and the regulating rods will be de-energized.

c. The combination of (a) and (b) causes a reactor trip.

Any other combination of at least one circuit breaker opening in each power supply will cause a reactor trip.

In summary, two tripped RPS channels will cause a reactor trip. For example, a reactor trip occurs if RPS channel B senses a low Reactor Coolant System (RCS) pressure condition and if RPS channel C senses a variable low RCS pressure condition. When the channel B bistable relay de-energizes, the channel trip relay de-energizes and opens its associated contacts. The same thing occurs in channel C, except the variable low pressure bistable relay de-energizes the channel C trip relay. When the output logic relays in channels B and C de-energize, the B and C contacts in the trip logic of each channel's reactor trip module (RTM) open causing an undervoltage to each trip breaker. All trip breakers and the ETA relay contactors open, and power is removed from all CRD mechanisms. All rods fall into the core, resulting in a reactor trip.

APPLICABLE SAFETY ANALYSES

Accident analyses rely on a reactor trip for protection of reactor core integrity, reactor coolant pressure boundary integrity, and reactor building OPERABILITY. A reactor trip must occur when needed to prevent accident consequences from exceeding those calculated in the accident analyses. The control rod insertion limits ensure that adequate rod worth is available upon reactor trip to shut down the reactor to the required SDM. Further, OPERABILITY of the CRD trip devices ensures that all CONTROL RODS will trip when required. More detailed descriptions of the applicable accident analyses are found in the Bases for each of the individual RPS trip Functions in LCO 3.3.1, "Reactor Protection System (RPS) Instrumentation."

In MODES 1 and 2, the CRD trip devices satisfy Criterion 3 of 10 CFR 50.36 (Ref. 2). In MODES 3, 4, and 5 with any CRD trip breaker in the closed position and the CRD System capable of rod withdrawal, the CRD trip devices satisfy Criterion 4 of 10 CFR 50.36.

LCO

The LCO requires all of the specified CRD trip devices to be OPERABLE in MODES 1 and 2, and in MODES 3, 4, and 5 when any CRD trip breaker is in the closed position and the CRD System is capable of rod withdrawal. Failure of any required CRD trip device renders a portion of the RPS inoperable. Without reliable CRD reactor trip circuit breakers and associated support circuitry, a reactor trip may not occur when initiated either automatically or manually.

All required CRD trip devices shall be OPERABLE to ensure that the reactor remains capable of being tripped any time it is critical. OPERABILITY is defined as the CRD trip device being able to receive a reactor trip signal and to respond to this trip signal by interrupting power to the CRDs. Both of a CRD trip breaker's diverse trip devices and the breaker itself must be functioning properly for the breaker to be OPERABLE.

Both ETA relays associated with each of the three regulating rod groups and the two ETA relays associated with the auxiliary power supply must be OPERABLE to satisfy the LCO. The ETA relays associated with the APSR power supply are not required to be OPERABLE because the APSRs are not designed to fall into the core upon initiation of a reactor trip.

Requiring all breakers and ETA relays to be OPERABLE ensures that at least one device in each of the two power paths to the CRDs will remain OPERABLE even with a single failure.

APPLICABILITY

The CRD trip devices are required to be OPERABLE in MODES 1 and 2, and in MODES 3, 4, and 5 when any CRD trip breaker is in the closed position and the CRD System is capable of rod withdrawal.

The CRD trip devices are designed to ensure that a reactor trip would occur if needed. Since a trip may be required in all of these MODES, the CRD trip devices must be OPERABLE.

ACTIONS

A Note has been added to the ACTIONS indicating separate Condition entry is allowed for each CRD trip device.

A.1 and A.2

Condition A represents reduced redundancy in the CRD trip Function. Condition A applies when:

- One diverse trip Function (undervoltage or shunt trip device) is inoperable in one or more CRD trip breaker(s) or breaker pair; or
- One diverse trip Function is inoperable in both DC trip breakers associated with one protection channel. In this case, the inoperable trip Function does not need to be the same for both breakers.

If one of the diverse trip Functions on a CRD trip breaker or breaker pair becomes inoperable, actions must be taken to preclude the inoperable CRD trip device from preventing a reactor trip when needed. This is done by manually opening the inoperable CRD trip breaker or by removing power from the inoperable CRD trip breaker. Either of these actions places the affected CRDs in a one-out-of-two trip configuration, which precludes a single failure, which in turn could prevent tripping of the reactor. The 48 hour Completion Time has been shown to be acceptable through operating experience.

B.1 and B.2

Condition B represents a loss of redundancy for the CRD trip Function. Condition B applies when both diverse trip Functions are inoperable in one or more trip breaker(s) or breaker pairs.

Required Action B.1 and Required Action B.2 are the same as Required Action A.1 and Required Action A.2, but the Completion Time is shortened. The 1 hour Completion Time allowed to open or remove power from the CRD trip breaker allows the operator to take all the appropriate actions for the inoperable breaker and still ensures that the risk involved is acceptable.

C.1, C.2, C.3, and C.4

Condition C represents a loss of redundancy for the CRD trip Function. Condition C applies when one or more ETA relays are inoperable. The preferred action is to restore the ETA relay to OPERABLE status. If this cannot be done, the operator can perform one of four actions to eliminate reliance on the failed ETA relay. The first option is to switch the affected CONTROL ROD group to an alternate power supply which has two OPERABLE or one OPERABLE and one open ETA relay.

This removes the failed ETA relay from the trip sequence, and the unit can operate indefinitely. The second option is to transfer the affected CONTROL ROD group to a DC holding power supply. This option is only available if the affected group is a safety rod group and the affected power supply is the auxiliary power supply. The third option is to open the inoperable ETA contacts. This option results in the safety function being performed. The fourth option is to open the corresponding AC CRD trip breaker. This also results in the safety function being performed, thereby eliminating the failed ETA relay from the trip sequence.

The 1 hour Completion Time is sufficient to perform the Required Action.

D.1, D.2.1, and D.2.2

If the Required Actions and associated Completion Times of Condition A, B, or C are not met in MODE 1, 2, or 3, 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, with all CRD trip breakers open or with power to all CRD trip breakers removed within 6 hours. The allowed 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.

E.1 and E.2

If the Required Actions and associated Completion Times of Condition A, B, or C are not met in MODE 4 or 5, the unit must be brought to a MODE in which the LCO does not apply. To achieve this status, all CRD trip breakers must be opened or power to all CRD trip breakers removed within 6 hours. The allowed Completion Time of 6 hours is reasonable, based on operating experience, to open all CRD trip breakers or remove power from all CRD trip breakers without challenging unit systems.

SURVEILLANCE REQUIREMENTS

SR 3.3.4.1

SR 3.3.4.1 is to perform a CHANNEL FUNCTIONAL TEST every 92 days. This test verifies the OPERABILITY of the trip devices by actuation of the end devices. Also, this test independently verifies the undervoltage and shunt trip mechanisms of the trip breakers. The Frequency of 92 days is based on operating experience, which has demonstrated that failure of more than one channel of a given function in any 92 day interval is a rare event (Ref. 3).

Testing in accordance with this SR is normally performed on a rotational basis with one channel being tested each 23 days. Testing one channel each 23 days reduces the probability of an undetected failure existing within the system and minimizes the likelihood of the same systematic test errors being introduced into each redundant trip device.

REFERENCES

1. SAR, Chapter 7.
2. 10 CFR 50.36.
3. BAW-10167A, "Justification for Increasing the Reactor Trip System On-Line Test Intervals," Supplement 3, "Justification for Increasing the Trip Device Test Interval," February 1998.

B 3.3 INSTRUMENTATION

B 3.3.9 Source Range Neutron Flux

BASES

BACKGROUND

The source range neutron flux channels provide the operator with an indication of the approach to criticality. These channels also provide the operator with a flux indication that reveals changes in reactivity.

The source range instrumentation has two redundant count rate channels originating in two high sensitivity fission chambers. Two source range detectors are externally located on opposite sides of the core. These channels are used over a counting range of 0.1 cps to 1E5 cps and are displayed on the operator's control console in terms of log count rate. The channels also measure the rate of change of the neutron flux level, which is displayed for the operator in terms of startup rate from -1 decades to +7 decades per minute. An interlock provides a control rod withdraw "inhibit" on a high startup rate of +2 decades per minute in either channel. This interlock is bypassed when the intermediate range neutron flux channels reach 1E-9 amps or power range neutron flux channels reach 10% RTP.

APPLICABLE SAFETY ANALYSES

The source range neutron flux channels are necessary to monitor core reactivity changes. They are the primary means for detecting reactivity changes and triggering operator actions to respond to reactivity transients initiated from conditions in which the Reactor Protection System (RPS) is not required to be OPERABLE. They also trigger operator actions to anticipate RPS actuation in the event of reactivity transients starting from shutdown or low power conditions.

The source range neutron flux channels satisfy Criterion 4 of 10 CFR 50.36 (Ref. 1).

LCO

One source range neutron flux channel shall be OPERABLE to provide the operator with source range neutron instrumentation. The source range instrumentation provides the primary power indication at ≤ 1E-10 amp on intermediate range instrumentation and must remain OPERABLE for the operator to continue increasing power.

APPLICABILITY

One source range neutron flux channel shall be OPERABLE in MODE 2 to provide indication during an approach to criticality. Neutron flux level is sufficient for monitoring on the intermediate range and on the power range instrumentation prior to entering MODE 1; therefore, source range instrumentation is not required in MODE 1.

In MODES 3, 4, and 5, source range neutron flux instrumentation shall be OPERABLE to provide the operator with a means of monitoring neutron flux and to provide an early indication of reactivity changes.

The requirements for source range neutron flux instrumentation during MODE 6 refueling operations are addressed in LCO 3.9.2, "Nuclear Instrumentation."

ACTIONS

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

With the required source range neutron flux channel inoperable with ≤ 1E-10 amp on the intermediate range neutron flux instrumentation, the operators must take actions to limit the possibilities for adding positive reactivity. This is done by immediately suspending positive reactivity additions, initiating action to insert all CONTROL RODS, and opening the control rod drive trip breakers within 1 hour. Periodic SDM verification is then required to provide a means for detecting the slow reactivity changes that could be caused by mechanisms other than CONTROL ROD withdrawal or operations involving positive reactivity changes. Since the source range instrumentation provides the only reliable direct indication of power in these MODES, the operators must continue to verify the SDM every 12 hours until at least one channel of the source range instrumentation is returned to OPERABLE status. Required Action A.1, Required Action A.2, and Required Action A.3 preclude rapid positive reactivity additions. The 1 hour Completion Time for Required Action A.3 and Required Action A.4 provides sufficient time for operators to accomplish the actions. The 12 hour Frequency for performing the SDM verification provides reasonable assurance that the reactivity changes possible with CONTROL RODS inserted are detected before SDM limits are challenged.

If no indication of intermediate range flux is available, these Required Actions are also appropriate.

B.1

With > 1E-10 amp in MODE 2, 3, 4, or 5 on the intermediate range neutron flux instrumentation, continued operation is allowed with the required source range neutron flux channel inoperable. The ability to continue operation is justified because the instrumentation does not provide a safety function during high power operation. However, actions are initiated within 1 hour to restore the required channel to OPERABLE status for future availability. The Completion Time of 1 hour is sufficient to initiate the action. The action must continue until the channel is restored to OPERABLE status.

SURVEILLANCE REQUIREMENTS

SR 3.3.9.1

Performance of the CHANNEL CHECK once every 12 hours provides reasonable assurance of prompt identification of a gross failure of instrumentation. A CHANNEL CHECK is normally a comparison of the parameter indicated on one channel to the same parameter on other channels. It is based on the assumption that instrument channels monitoring the same parameter should read approximately the same value. Significant deviations between the two instrument channels could be an indication of excessive instrument drift in one of the channels or of something even more serious. CHANNEL CHECK will detect gross channel failure; therefore, it is key in verifying that the instrumentation continues to operate properly between each CHANNEL CALIBRATION.

Agreement criteria are determined by the unit staff, based on a combination of factors including channel instrument uncertainties. If a channel is outside the criteria, it may be an indication that the signal processing equipment has drifted outside its limit. If the channels are within the criteria, it is an indication that the channels are OPERABLE. If the channels are normally off scale during times when surveillance is required, the CHANNEL CHECK will only verify that they are off scale in the same direction.

The Frequency is based on operating experience that demonstrates channel failure is rare. Since the probability of two random failures in redundant channels in any 12 hour period is extremely low, the CHANNEL CHECK minimizes the chance of loss of protective function due to failure of redundant channels. The CHANNEL CHECK supplements less formal, but more frequent, checks of channel OPERABILITY during normal operational use of the displays associated with the LCO's required channel. When operating with only one channel OPERABLE, CHANNEL CHECK is still required. However, in this condition, a redundant source range may not be available for comparison. CHANNEL CHECK may still be performed via comparison with intermediate range detectors, if available, and verification that the OPERABLE source range channel is energized and indicating a value consistent with current unit status.

SR 3.3.9.2

For a source range neutron flux channel, CHANNEL CALIBRATION is a complete check and readjustment of the channel from the preamplifier input to the indicator.

This test verifies the channel responds to measured parameters within the necessary range and accuracy. CHANNEL CALIBRATION leaves the channel at a setpoint which accounts for instrument drift to ensure that the instrument channel remains operational between successive tests.

The SR is modified by a Note excluding neutron detectors from CHANNEL CALIBRATION. It is not necessary to test the detectors because generating a meaningful test signal is difficult, and there is no adjustment that can be made to the detectors. Furthermore, adjustment of the detectors is unnecessary because they are passive devices with minimal drift. Finally, the detectors are of simple construction, and any failures in the detectors will be apparent as change in channel output.

The Frequency of 18 months is based on demonstrated instrument CHANNEL CALIBRATION reliability over an 18 month interval, such that the instrument is not adversely affected by drift.

REFERENCES

1. 10 CFR 50.36.

B 3.3 INSTRUMENTATION

B 3.3.10 Intermediate Range Neutron Flux

BASES

BACKGROUND

The intermediate range neutron flux channels provide the operator with an indication of reactor power at higher power levels than the source range instrumentation and lower power levels than the power range instrumentation.

The intermediate range instrumentation has two channels originating in two gamma compensated ion chambers. Each channel provides eight decades of flux level information in terms of the log of ion chamber current from 1E-11 amp to 1E-3 amp. The channels also measure the rate of change of the neutron flux level, which is displayed for the operator in terms of startup rate from -0.5 decades to +5 decades per minute. A high startup rate of +3 decades per minute in either channel will initiate a control rod withdrawal inhibit while below 10% RTP.

The intermediate range compensated ion chambers are of the electrically adjustable gamma compensating type. Each detector has a separate adjustable high voltage power supply and an adjustable compensating voltage supply.

APPLICABLE SAFETY ANALYSES

Intermediate range neutron flux channels are necessary to monitor core reactivity changes and provide the primary indication to trigger operator actions to anticipate Reactor Protection System actuation in the event of reactivity transients starting from low power conditions.

The intermediate range neutron flux channels satisfy Criterion 4 of 10 CFR 50.36 (Ref. 1).

LCO

One intermediate range neutron flux instrumentation channel shall be OPERABLE to provide the operator with neutron flux indication. This enables operators to control the increase in power and to detect neutron flux transients. This indication is used until the power range instrumentation is on scale. Violation of this requirement could prevent the operator from detecting and controlling neutron flux transients that could result in reactor trip during power escalation.

APPLICABILITY

The required intermediate range neutron flux channel shall be OPERABLE in MODE 2 and in MODES 3, 4, and 5 with any control rod drive (CRD) trip breaker in the closed position and the CRD System capable of rod withdrawal.

The intermediate range instrumentation is designed to detect power changes when the power range and source range instrumentation cannot provide reliable indications, e.g., during initial criticality and power escalation. Since those conditions can exist in, or propagate from, all of these MODES, the intermediate range instrumentation must be OPERABLE.

ACTIONS

A.1 and A.2

With the required intermediate range neutron flux channel inoperable when THERMAL POWER is ≤ 5% RTP, the operators must place the reactor in the next lowest condition for which the intermediate range instrumentation is not required. This involves providing power level indication on the source range instrumentation by immediately suspending operations involving positive reactivity changes and, within 1 hour, placing the reactor in the tripped condition with the CRD trip breakers open. The Completion Times are based on unit operating experience and allow the operators sufficient time to manually insert the CONTROL RODS prior to opening the CRD breakers.

SURVEILLANCE REQUIREMENTS

SR 3.3.10.1

Performance of the CHANNEL CHECK once every 12 hours provides reasonable assurance of prompt identification that a gross failure of instrumentation has not occurred. A CHANNEL CHECK is normally a comparison of the parameter indicated on one channel to the same parameter on other channels. It is based on the assumption that instrument channels monitoring the same parameter should read approximately the same value. Significant deviations between the two instrument channels could be an indication of excessive instrument drift in one of the channels or of something even more serious. CHANNEL CHECK will detect gross channel failure; therefore, it is key in verifying that the instrumentation continues to operate properly between each CHANNEL CALIBRATION.

Agreement criteria are determined by the unit staff based on a combination of factors including channel instrument uncertainties. If a channel is outside the criteria, it may be an indication that the signal processing equipment has drifted outside its limit. If the channels are within the criteria, it is an indication that the channels are OPERABLE. Off scale low current loop channels are verified, where practical to be reading at the bottom of the range and not failed low.

The Frequency, about once every shift, is based on operating experience that demonstrates channel failure is rare. Since the probability of two random failures in redundant channels in any 12 hour period is extremely low, the CHANNEL CHECK minimizes the chance of loss of protective function due to failure of redundant channels. The CHANNEL CHECK supplements less formal, but more frequent, checks of channel OPERABILITY during normal operational use of the displays associated with the LCO's required channel.

When operating with only one channel OPERABLE, CHANNEL CHECK is still required. However, in this condition, a redundant intermediate range is not available for comparison. CHANNEL CHECK may still be performed via comparison with power or source range detectors, if available, and verification that the OPERABLE intermediate range channel is energized and indicates a value consistent with current unit status.

SR 3.3.10.2

A CHANNEL FUNCTIONAL TEST, of the required intermediate range instrument channel, verifies proper operation of the channel each 31 days. Monthly testing provides reasonable assurance that the instrument channel will function, if required, to provide indication during MODE 2 and during unanticipated reactivity excursions from MODES 3, 4, or 5.

SR 3.3.10.3

For intermediate range neutron flux channels, CHANNEL CALIBRATION is a complete check and readjustment of the channels, from the preamplifier input to the indicators. This test verifies the channel responds to a measured parameter within the necessary range and accuracy. CHANNEL CALIBRATION leaves the channel at a setpoint which accounts for instrument drift to ensure that the instrument channel remains operational between successive tests.

The SR is modified by a Note excluding neutron detectors from CHANNEL CALIBRATION. It is not necessary to test the detectors because generating a meaningful test signal is difficult. In addition, the detectors are of simple construction, and any failures in the detectors will be apparent as a change in channel output. The Frequency is based on operating experience and consistency with the typical industry refueling cycle and is justified by demonstrated instrument reliability over an 18 month interval such that the instrument is not adversely affected by drift.

SR 3.3.10.4

SR 3.3.10.4 is the verification performed each reactor startup of one decade of overlap with the source range neutron flux instrumentation. This ensures a continuous source of power indication during the approach to criticality. Failure to perform this Surveillance leaves the unit in a condition where the source range channels provide adequate protection until the verification can be made.

REFERENCES

1. 10 CFR 50.36.

CTS DISCUSSION OF CHANGES

ITS Section 3.3A: Instrumentation - RPS

Note: ITS Section 3.3A package includes the following ITS:

ITS 3.3.1	Reactor Protection System (RPS) Instrumentation
ITS 3.3.2	RPS Manual Reactor Trip
ITS 3.3.3	RPS—Reactor Trip Module (RTM)
ITS 3.3.4	Control Rod Drive (CRD) Trip Devices
ITS 3.3.9	Source Range Neutron Flux
ITS 3.3.10	Intermediate Range Neutron Flux

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 B&W 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 RSTS Bases. The CTS Bases will be reviewed for technical content that will be identified for retention in the ITS Bases.

A3 CTS 3.5.1.1, 3.5.1.2, and 3.5.1.3 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, but CTS 3.5.1.3 provides limitations for any one or two channels inoperable. Similarly, CTS Specifications 4.1.a, 4.1.b, and 4.1.c 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.

A4 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 (See DOC L2 and M4)
Q - Quarterly	92 days
P - Prior to 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

Each of these changes is consistent with the current application of the CTS frequencies at ANO-1. These changes maintain requirements consistent with both CTS and NUREG-1430. These changes are administrative in nature because they represent a change in presentation format only with no change of actual requirements.

A5 The CTS requirement to perform heat balance calibrations "daily under non-steady state operating conditions" has been retained in ITS in the Frequency of SR 3.3.1.2. This portion of the ITS Frequency is stated as "Once within 24 hours after a THERMAL POWER change of ≥ 10% RTP in one direction." This change in wording provides requirements consistent with the ANO-1 application of the CTS requirements. This change provides a change in format and a clarification of existing requirements. No new requirements are added and no existing requirements are removed by this change.

A6 The power range nuclear instruments at ANO-1 provide the nuclear power input to the Reactor Protection System (RPS) channels. This input is used in several reactor trip functions including the nuclear overpower trip function. CTS Table 3.5.1-1 and Table 4.1-1 provide specific requirements for these power range instrument channels, as well as requirements for all but one of the reactor trip functions which depend on these instruments for input. The nuclear overpower trip function is not specifically addressed in CTS Table 3.5.1-1 or Table 4.1-1. The power range nuclear instrument channels will not be specifically dealt with in ITS. Rather, ITS Specification 3.3.1 including Table 3.3.1-1 will deal individually with each RPS trip function which receives input from the power range instrument channels, including the nuclear overpower trip function. This change is administrative in nature because it represents a change, in the presentation of these requirements, with no actual change in requirements.

CTS DISCUSSION OF CHANGES

A7 The term Minimum Degree of Redundancy as presented in CTS, i.e., Table 3.5.1-1 Column 4, will not be retained in ITS. Omission of this term is not considered to result in any changes in requirements since the intent of this column is consistent with application of Table 3.5.1-1 Column 3, "Minimum Channels Operable," which is retained (although the format is changed per DOC A3). Removal of this term and its usage from the CTS does not represent any actual change in requirements, only a change in presentation.

A8 CTS Table 3.5.1-1 RPS Functional Unit 13 "Electronic (SCR) Trip Relay" indicates that there are two channels of electronic trip relays and that they must both be OPERABLE. Each of these channels consists of five electronic trip relays, four of which are required. The four required relays are associated with the three regulating rod power supplies and the auxiliary power supply which can be manually selected to power rods from any of the eight groups. The two which are not required are associated with the power supply for the AXIAL POWER SHAPING RODS (APSRs). OPERABILITY of the relays in the APSR power supply are not required because these rods are designed not to insert upon a reactor trip. The actions associated with these CTS requirements, which are found in Table 3.5.1-1 Note 23, deal specifically with one inoperable relay and with two or more inoperable relays.

NUREG-1430 treats each of these required relays individually by specifying in LCO 3.3.4 that eight "electronic trip assembly (ETA) relays" must be OPERABLE. (Note: CTS term "Electronic (SCR) Trip Relay" is considered equivalent to ITS term "electronic trip assembly (ETA) relays.") The change from specifying two channels, each of which by design contains four required relays, to specifying eight individual relays is administrative in nature. No new requirements are added by this change nor are any existing requirements removed by it. This change provides requirements consistent with NUREG-1430.

A9 NUREG-1430 3.3.4 ACTIONS NOTE has been adopted in ITS. This note allows separate Condition entry for each CRD trip device. The adoption of this Note maintains flexibility similar to that provided by the CTS. The CTS allows application of the action requirements found in Table 3.5.1-1 Notes 24 and 25 separately to each type of CRD trip breakers (AC and DC). This flexibility is retained in the ITS by application of the ACTIONS NOTE to 3.3.4 Conditions A and B. CTS Table 3.5.1-1 Note 23 provides specific action requirements for two or more inoperable Electronic Trip Relays. Application of the ACTIONS NOTE to 3.3.4 Condition C retains these requirements. No new requirements are added and no existing requirements deleted by this change.

A10 The Note modifying ITS SR 3.3.1.6 has been adopted. This Note specifically excludes neutron detectors from CHANNEL CALIBRATIONS. While the allowance provided by this Note was not specifically expressed in CTS, the application of the ANO-1 CTS definition of Instrument Channel Calibration has in practice excluded the neutron detectors. This exclusion has been made due to the passive design of the detectors, the extreme difficulty in both accessing the detectors and in generating an appropriate input signal to the detectors, and the fact that no specific adjustments can be made to the detectors. Although no specific exceptions, as allowed by this Note, exist in CTS, its adoption is administrative in nature because no actual change in requirements is made by adopting it.

A11 The CTS Table 4.1-1 Item 4 specifies the testing requirements for the Power Range Channel and CTS Table 4.1-1 Item 10 specifies the testing requirements for the Flux-Reactor Coolant Flow Comparator. These testing requirements have been retained in ITS Table 3.3.1-1 and are specifically applied to each appropriate Function in ITS Table 3.3.1-1.

As applied at ANO-1, CTS Table 4.1-1 requires a monthly CHANNEL CALIBRATION and monthly CHANNEL FUNCTIONAL TEST of both the Nuclear Overpower Function and the Nuclear Overpower RCS Flow and Measured AXIAL IMBALANCE Function. Both the CTS and ITS definitions specify that the required calibration includes the CHANNEL FUNCTIONAL TEST. Therefore, the specific requirements to perform CHANNEL FUNCTIONAL TESTS on the same Frequency as the CHANNEL CALIBRATIONS are not retained in the ITS. This change represents no actual change in requirements, only a change in presentation of requirements.

A12 CTS Table 2.3-1 indicates the Reactor Protection System Trip Setting Limit for the high reactor building pressure trip function as a maximum of 4 psig and parenthetically indicates that this is equivalent to 18.7 psia. This limit has been incorporated into the ITS as the Allowable Value for the Reactor Building High Pressure Function in Table 3.3.1-1. This value has been specified as ≤18.7 psia to be consistent with the actual design of the instrumentation used for this function. The removal of the reference to the equivalent value of 4 psig is administrative in nature and represents no change in requirements, only a change in presentation.

A13 The allowances and requirements of CTS Table 3.5.1-1 Note 24 part "b." are not specifically retained in ITS. (Note that the word "operable" in the first sentence of Note 24 part "b." was inadvertently changed from "inoperable," following the original insertion of this Note into CTS.) Based on the history of CTS, Table 3.5.1-1 Note 24 appears to have been adopted from similar requirements found in NUREG-0103 Rev. 4, Standard Technical Specifications for Babcock and Wilcox Pressurized Water Reactors, Table 3.3-1 Action 7. Part "b." of CTS Table 3.5.1-1 Note 24, and NUREG-0103 Table 3.3-1 Action 7 both provide allowances to bypass a channel, to allow testing, while operating with one channel inoperable. The design of the CRD trip breakers, at ANO-1, does not contain a bypass feature. Therefore, the allowance provided by CTS Table 3.5.1-1, Note 24, part "b.," is not appropriate for the CRD trip breakers and cannot be implemented at ANO-1.

The removal of the allowances of CTS Table 3.5.1-1, Note 24, part "b." is administrative in nature because, due to the design of the equipment to which they are applicable, these allowances cannot be used. Removal of these requirements will result in no actual change in the application of CTS Table 3.5.1-1 Note 24 requirements.

A14 Not used.

A15 Not used.

A16 Not used.

TECHNICAL CHANGE – MORE RESTRICTIVE

M1 CTS Table 3.5.1-1 Reactor Protection System section Item 1 requirements for the manual push-button have been replaced with ITS 3.3.2.

ITS 3.3.2 has an Applicability of MODES 1 and 2, and MODES 3, 4, and 5 with any CONTROL ROD drive (CRD) trip breaker in the closed position and the CRD System capable of rod withdrawal. The equivalent CTS requirements were, by implication, applicable while above hot shutdown (MODE 3). This implied Applicability was based on the action requirements of Table 3.5.1-1 Note 1. The adoption of ITS 3.3.2 Applicability will specifically require this function to be OPERABLE in MODES 1 and 2. In addition, the RPS Manual Trip Function will be required to be OPERABLE while in MODES 3, 4 and 5 with any CRD trip breaker in the closed position and the CRD System capable of rod withdrawal. The adoption of this Applicability has been made to provide requirements consistent with those presented in NUREG-1430.

The adoption of ITS 3.3.2 ACTION A provides for a one hour restoration period prior to entry into Condition B or C. Adoption of this new ACTION provides a specific period of time for restoration, which did not exist in CTS. The one hour Completion Time of Required Action A.1, when added to the Completion Times of Required Actions B.1, B.2, and C.1, in each case, results in total times which are more restrictive than CTS requirements. ITS 3.3.2 ACTION A was adopted to provide requirements consistent with NUREG-1430.

Due to the adoption of an Applicability which includes MODES other than MODES 1 and 2, the adoption of ITS 3.3.2 ACTION C was required. This ACTION provides requirements appropriate for exiting the MODES of Applicability for this LCO. This ITS ACTION requirement, which is more restrictive than current requirements, was adopted to provide requirements consistent with NUREG-1430.

M2 Testing requirements on the anticipatory trip functions within the Reactor Protection System (RPS) have been changed to provide requirements consistent with the testing of other Functions within RPS. These changes have been made for both the turbine trip and the loss of main feedwater pump anticipatory reactor trips. These changes provide requirements consistent with those of NUREG-1430.

Specifically, the CHANNEL CHECK Frequency was changed from monthly to 12 hours. This 12 hour Frequency is consistent with the CHANNEL CHECK requirements, in CTS, for other RPS trip functions and is consistent with NUREG-1430. Additionally, the CHANNEL FUNCTIONAL TEST Frequency has been changed from "PC - Prior to going Critical if not done within previous 31 days" to 31 days. This change ensures that each of these channels is functionally tested each month, not just once prior to criticality. This monthly testing is consistent with the CTS testing requirements for other RPS trip functions and is consistent with the requirements of NUREG-1430.

M3 The adoption of ITS 3.3.1 ACTION D and ITS 3.3.2 ACTION B provides requirements more restrictive than those in CTS Table 3.5.1-1 Note 1. The adoption of these ACTIONS will require entry into MODE 3 within 6 hours as opposed to the current 12 hours. Additionally, these ACTIONS will require all CRD trip breakers to be opened within 6 hours. These changes are being made to improve consistency between the requirements of the ITS and NUREG-1430.

M4 Heat balance calibration requirements found in CTS Table 4.1-1 Item 3 have been replaced by ITS SR 3.3.1.2. This change includes the addition of details of when an adjustment of the power range instruments is required. This detail was not provided in CTS and therefore its addition represents more restrictive requirements. This change was made to provide testing requirements consistent with those found in NUREG-1430.

The Frequency for performance of heat balance calibrations was also changed. The CTS twice weekly requirement was replaced by a 96 hour Frequency in ITS SR 3.3.1.2. Because no specific requirement exists in CTS to perform this twice weekly calibration at equal intervals, testing could be performed with more than a 96 hour interval between them. This change to a 96 hour Frequency provides requirements consistent in format with NUREG-1430, while maintaining testing on a Frequency roughly equivalent to CTS requirements.

M5 The actions for inoperable Electronic (SCR) Trip Relays in CTS Table 3.5.1-1 Note 23 have been replaced by ITS 3.3.4 ACTION C. (Note: CTS term "Electronic (SCR) Trip Relay" is considered equivalent to ITS term "electronic trip assembly (ETA) relays.") CTS allows up to 48 hours to restore a single inoperable electronic (SCR) trip relay after which, the device is required to be in the trip (open) condition within the next hour. In the event more than one electronic trip relay, in a channel, is inoperable, all electronic (SCR) trip devices, in the channel, are to be tripped within one hour. These requirements have been replaced by ITS 3.3.4 ACTION C. This ACTION contains no provision for a 48 hour delay prior to requiring additional action to be taken. Because action to compensate for a single inoperable ETA is required sooner by ITS than by CTS, this change is more restrictive.

Additionally, because no actions were specified in CTS in the event the actions of Table 3.5.1-1 Note 23 were not completed, the addition of ITS 3.3.4 ACTION D is also more restrictive. With no additional actions specified, entry into CTS 3.0.3 would be appropriate, upon failure to comply with Table 3.5.1-1 Note 23. CTS 3.0.3 would allow up to 13 hours to reach hot shutdown (MODE 3) conditions. ITS 3.3.4 ACTION D requires the unit to be placed in MODE 3, with either the CRD trip breakers open, or power removed from the CRD system, within 6 hours.

ITS 3.3.4 ACTIONS C and D are being adopted to provide requirements consistent with NUREG-1430 and to provide ACTIONS for the ETA relays which are consistent with the ACTIONS required for the other CRD trip devices.

CTS DISCUSSION OF CHANGES

M6 ITS 3.3.4 Condition A has been adopted replacing CTS Table 3.5.1-1 Note 25. More restrictive requirements are represented by Condition A in that 48 hours were previously allowed for restoration of the inoperability followed by one hour in which to trip the breaker. Condition A will allow a total of 48 hours for restoration and either tripping (opening) the breaker or removing power from it. This reduction in total Completion Time from 49 hours to 48 hours is adopted to provide requirements consistent with NUREG-1430.

M7 CTS Table 3.5.1-1 Note 3 allowed continued operation above hot shutdown with the required source range instrument channel inoperable provided at least one intermediate range instrument was indicating greater than 1E-10 amps. No specific requirement existed in CTS to initiate repairs on this inoperable instrument. This CTS requirement has been replaced by ITS 3.3.9 ACTION B. The adoption of ACTION B will continue to allow operation above MODE 3 with the required source range instrument channel inoperable. However, the additional requirement to initiate action to repair the inoperable instrument channel within 1 hour is included. This additional requirement has been adopted to provide requirements consistent with NUREG-1430.

M8 CTS Table 3.5.1-1 RPS Functional Units 3 and 4, intermediate range instrument channels and source range instrument channels, both indicate that the actions of Note 1 are required in the event that the required instrument channel is inoperable. Note 1 requires that the unit be placed in hot shutdown (MODE 3) within 12 hours. No actions are specified in CTS to deal with their inoperability while in MODE 3 or below.

The requirements of Table 3.5.1-1 Note 1, as applied to the source range and intermediate range instrument channels, have been replaced by ITS 3.3.9 ACTION A and ITS 3.3.10 ACTION A, respectively. These new requirements are more restrictive in that they provide additional ACTIONS not required by CTS. These new ACTIONS provide requirements which ensure that the unit is placed in an acceptable condition to compensate for the inoperability of either the required source range instrument or the required intermediate range instrument. These additional ACTIONS are being adopted to provide requirements which are consistent with NUREG-1430 requirements.

M9 CHANNEL CALIBRATION requirements for the source range and intermediate range instruments on an 18 month Frequency have been adopted. Adoption of ITS SR 3.3.9.2 and ITS SR 3.3.10.3 represent more restrictive requirements because no equivalent requirements exist in CTS. These CHANNEL CALIBRATION requirements have been adopted to provide testing requirements consistent with NUREG-1430.

M10 The Required Action to be performed in the event one decade of overlap between the source range and intermediate range instruments is not achieved was presented in CTS 3.5.1.5. This has been replaced by ITS 3.3.10 ACTION A. The adoption of this ITS ACTION presents more restrictive requirements in that unlimited continued operation in the source range will no longer be allowed. ITS 3.3.10 ACTION A has been adopted to provide requirements consistent with NUREG-1430.

CTS DISCUSSION OF CHANGES

M11 Requirements for the Reactor Protection System (RPS) Nuclear Overpower Low Setpoint function and the Shutdown Bypass RCS High Pressure function have been adopted in ITS Table 3.3.1-1. The adoption of these requirements includes the ACTION requirements of ITS 3.3.1 ACTION E. This ACTION has been added to provide an appropriate ACTION, where none existed in CTS, to deal with the inoperability of the shutdown bypass functions within the RPS. The adoption of ITS 3.3.1 ACTION E provides a Condition and Required Action consistent with NUREG-1430 and provides a Completion Time consistent with the Completion Times for other RPS ACTIONS in ITS 3.3.1. The Surveillance Requirements for these shutdown bypass functions were also adopted. These SRs were adopted to provide testing requirements consistent with those presented in NUREG-1430 for these functions.

M12 The "Applicable MODES or Other Specified Conditions" column of ITS Table 3.3.1-1 has been adopted to provide specific Applicability requirements for the individual RPS functions where no specific requirement existed in the ANO-1 CTS. Each of the RPS functions in CTS Table 3.5.1-1 was by implication required to be OPERABLE while above hot shutdown (MODE 3). This implied Applicability was based on the action requirements of Table 3.5.1-1 Note 1. The addition of specific Applicability requirements while in other than MODES 1 and 2, where none existed previously, is consistent with NUREG 1430.

M13 CTS 3.5.1.3 is revised to reflect ITS 3.3.1 ACTIONS A and B. The requirements are essentially equivalent except that no Completion Time is included in the CTS. This addition of a specific Completion Time is considered more restrictive because no specific requirements similar to these exist in CTS. These changes have been made to provide requirements consistent with NUREG-1430.

M14 ITS 3.3.3 Reactor Protection System (RPS)--Reactor Trip Module (RTM) LCO, Applicability, and ACTIONS have been adopted. ANO-1 CTS requirements for the RTMs were presented only as testing requirements in Table 4.1-1 Item 1, Protective Channel Coincidence Logic. The adoption of these ITS requirements represent more restrictive requirements because no specific requirements similar to these exist in CTS. These changes have been made to provide requirements consistent with NUREG-1430.

M15 ITS 3.3.4, CONTROL ROD Drive (CRD) Trip Devices, Applicability, and ACTIONS D and E, and their respective Bases, have been adopted. Each of these items was adopted to provide requirements consistent with NUREG-1430. The adoption of these items represents more restrictive requirements because no specific requirements similar to these exist in CTS.

M16 ITS 3.3.9, Source Range Neutron Flux, Applicability has been adopted. While the MODE 2 Applicability was implied by the action requirements of Table 3.5.1-1 Note 1 and 2, no specific statement of Applicability for these instrument channels exists in CTS. Although at least one source range instrument channel is now maintained OPERABLE while operating in MODES 2, 3 , 4, and 5, the adoption of the Applicability statement in ITS 3.3.9 represents more restrictive requirements in that specific requirements will now exist where none existed previously. Appropriate ACTION requirements were also adopted in the form of ITS 3.3.9 ACTIONS A. These additional requirements have been adopted to provide requirements consistent with NUREG-1430 while maintaining the CTS requirement to require only one source range instrument channel while in these MODES.

M17 ITS 3.3.10, Intermediate Range Neutron Flux, Applicability has been adopted. While the MODE 2 Applicability was implied by the action requirements of Table 3.5.1-1 Note 1 and 2, no specific statement of Applicability for these instrument channels exists in CTS. Although at least one intermediate range instrument channel is now maintained OPERABLE while operating in MODE 2, the adoption of the Applicability statement in ITS 3.3.10 represents more restrictive requirements in that specific requirements will now exist where none existed previously. Appropriate ACTION requirements were also adopted in the form of ITS 3.3.10 ACTION A. These additional requirements have been adopted to provide requirements consistent with NUREG-1430 while maintaining the CTS requirement to require only one intermediate range instrument channel while in these MODES.

M18 ITS 3.3.1 Condition C has been adopted. This adoption provides for specific Required Actions in the event ITS 3.3.1 Required Action and Completion Times of Condition A or B are not met. The proposed Condition C also provides Required Actions in the event that more than 2 channels are inoperable.

ITS 3.3.1 ACTIONS A and B are essentially the requirements of CTS 3.5.1.3 (with the exception of Completion Times: see DOC M13). Failure to comply with CTS 3.5.1.3 would have required entry into CTS 3.0.3 which would have allowed continued operation above hot shutdown (MODE 3) for up to 13 hours. The adoption of ITS 3.3.1 Condition C will in many cases result in requirements to enter MODE 3 within 6 hours. These more restrictive requirements are being adopted to provide requirements consistent with NUREG-1430 requirements.

When applied to the specific condition of more than 2 channels inoperable, the adoption of ITS 3.3.1 Condition C is administrative in nature. The adoption of Condition C requires entry into specific ACTION requirements. Similarly, CTS Table 3.5.1-1 column 5 specifies that the actions of CTS Table 3.5.1-1 Note 1 apply when less than 2 channels are OPERABLE (more than 2 channels inoperable). The changes in Completion Times related to CTS Table 3.5.1-1 Note 1 are discussed in other Discussions of Changes.

M19 The Allowable Values for ITS Table 3.3.1 Function 9 Main Turbine Trip (Control Oil Pressure) and Function 10 Loss of Main Feedwater Pumps (Control Oil Pressure) have been added. Because these values were not specified in CTS their adoption in ITS represents additional restrictions These additions has been made to provide requirements consistent with NUREG-1430 requirements.

TECHNICAL CHANGE – LESS RESTRICTIVE

L1 The requirements of CTS 3.5.1.6 have been replaced by those of ITS 3.3.4 Condition B. The requirements of ITS 3.3.4 Condition B are less restrictive than those found in CTS 3.5.1.6 and are consistent with the actions required by CTS Table 3.5.1-1 Note 24.

Adoption of the requirements of ITS 3.3.4 Condition B allows a time period of up to one hour to either trip or remove power from an inoperable CONTROL ROD Drive (CRD) trip breaker as opposed to the 30 minutes allowed by CTS 3.5.1.6 for removing power from a trip device failed in the untripped state. The change from 30 minutes to one hour is consistent with the requirements of CTS Table 3.5.1-1 Note 24 for an inoperable CRD trip breaker (not specifically failed in the untripped state), and is consistent with NUREG-1430 requirements. The change from 30 minutes to one hour removes the confusion provided by having different times specified in CTS for completing identical actions for very similar failures. The additional 30 minutes only slightly increases the time the unit is allowed to operate in this condition and therefore, the likelihood of an additional failure occurring during this time which could prevent the CONTROL RODS from inserting, is very slight.

Additionally, ITS 3.3.4 will allow unrestricted operation for an unlimited period of time with an inoperable CRD trip breaker, even if the breaker is in the untripped state, provided power is removed from that breaker. CTS 3.5.1.6 requires that the untripped breaker be repaired and the remaining trip devices be tested within 8 hours of detection or the unit be placed in Hot Shutdown (MODE 3) within the next 4 hours. Operation for an indefinite period of time, with inoperable untripped CRD trip devices, from which power has been removed, is acceptable because removal of power from the untripped device accomplishes the same result as tripping the device. Because the function of the of the CRD trip device has been accomplished, continued operation is acceptable. Removal of the requirement to test the remaining trip devices when one is found failed in the untripped state is also appropriate. Failure of one of these devices in the untripped state in no way increases the likelihood of the failure of another of these devices. Further, any indication of a common mode failure results (through administrative controls) in timely evaluation of the condition and appropriate determination of the need for additional testing.

L2 Heat balance calibration requirements found in CTS Table 4.1-1 Item 3 have been replaced by ITS SR 3.3.1.2. The Frequency for performance of heat balance calibrations of twice weekly was changed to a 96 hour frequency in ITS SR 3.3.1.2. Although no specific requirement exists in CTS to perform this twice-weekly calibration at equal intervals, if this twice weekly requirement is interpreted as being required on equal intervals, the change to 96 hours represents a less restrictive change. Based on this interpretation of current requirements, the 96 hour Frequency would provide an additional 12 hours between required performances of this calibration. This additional time between performances of heat balance calibrations is acceptable, based on operating experience, due to the very slow divergence between power range instruments and heat balance power level during steady state operation. This change to a 96 hour Frequency provides requirements consistent in format with NUREG-1430, while maintaining testing on a Frequency which approximates CTS requirements.

L3 The Applicability of the requirements to have OPERABLE RPS anticipatory trip functions on a loss of main feedwater and on a turbine trip has been changed from the CTS Applicability of greater than 5% reactor power for both, to ≥ 10% RTP for Loss of Main Feedwater Pumps and ≥ 45% RTP for Main Turbine Trip. ANO-1 CTS allows the trip on loss of main feedwater function to be bypassed up to 10% reactor power and the trip on turbine trip function to be bypassed up to 45% reactor power. The adoption of the ITS Applicability will maintain requirements consistent with CTS in that the trip function for Loss Main Feedwater will be required to be in service (not bypassed) and OPERABLE prior to exceeding 10% RTP. Similarly, the trip function for Main Turbine Trip will be required to be in service (not bypassed) and OPERABLE prior to exceeding 45% RTP. The ITS requirements are less restrictive than the CTS requirements in that the trip on Loss of Main Feedwater Pumps will no longer be required to be OPERABLE between 5% and 10% power and the trip on Main Turbine Trip will no longer be required to be OPERABLE between 5% and 45% power.

This change in Applicability is acceptable because it provides OPERABILITY requirements for the anticipatory trip functions which are consistent with their safety function. By requiring these trip functions to be OPERABLE while operating at reactor power levels at which they were allowed to be bypassed, CTS provided OPERABILITY requirements which were inconsistent with their safety function.

Additionally, the action requirements provided by CTS 3.5.1.9.3 have been replaced by ITS 3.3.1 Conditions F and G. This change provides for ACTION requirements which are consistent with the Applicability of ITS 3.3.1 for these functions and which are consistent with NUREG-1430. This change will provide requirements to exit the MODES of Applicability in the event these functions are INOPERABLE. Removal of the requirements to go to hot shutdown (MODE 3), which is well below even the CTS MODES of Applicability, is consistent with the philosophy presented in NUREG-1430. This philosophy is to require that either the equipment be restored or the unit exit the MODES of Applicability.

L4 Not Used.

L5 Three channels of power range instruments within RPS are required by CTS Table 3.5.1-1. This requirement is modified by Note 4 which allows a reduction to two OPERABLE channels for up to four hours to allow testing, calibration or maintenance. This reduction to two OPERABLE channels is allowed provided the degree of redundancy is maintained at one. This requirement will be replaced by the Nuclear Overpower High Setpoint specified in ITS Table 3.3.1-1 and the Conditions of ITS 3.3.1. This change will retain the requirement to have a minimum of two OPERABLE channels of power range instrumentation. Operation with only two OPERABLE channels will be allowed to continue indefinitely provided the requirements of ITS 3.3.1 Condition B are met. The Required Actions of Condition B place the system in the same configuration as required by CTS Table 3.5.1-1 Note 4. By tripping one inoperable channel and bypassing the other, the system is reduced to a one out of two logic for the remaining OPERABLE channels. This is consistent with CTS Table 3.5.1-1 Note 4, which requires two OPERABLE channels with a degree of redundancy of one.

Adoption of these ITS requirements is less restrictive in that operation with two inoperable power range instrument channels will now be allowed indefinitely whereas by CTS this condition was limited to four hours. Indefinite operation with only two OPERABLE power range instruments is equally as appropriate as operation with only two OPERABLE channels of any other RPS function, provided the system is placed in a one out of two logic for the remaining two OPERABLE channels, as required by ITS 3.3.1 Condition B. This change provides requirements consistent with NUREG-1430 and provides consistent requirements for each of the trip functions within RPS.

L6 Not Used.

L7 ITS 3.3.4 Required Actions and Completion Times C.1, C.2 and C.4 have been adopted as alternatives to the requirements presented in CTS Table 3.5.1-1 Note 23. Required Action C.1 specifically allows for a CONTROL ROD group with an inoperable ETA relay to be placed on a power supply which has OPERABLE or open ETA relays. Required Action C.2 allows for a safety rod group which is being powered from the auxiliary power supply to be returned to its DC hold power supply if one or both ETA relays associated with the auxiliary power supply become inoperable. Required Action C.4 provides an additional alternative for dealing with inoperable ETA relays. This Required Action allows the corresponding AC breaker to be opened to compensate for the loss of the ETA relay. These allowances all provide new flexibility which was not previously specified in CTS. Required Actions C.1 and C.2 are each acceptable alternatives to opening an inoperable ETA relay because they each place the affected CONTROL ROD group on a power supply which will ensure that the rods are de-energized upon a reactor trip. Required Action C.4 is an acceptable alternative because it interrupts power to the affected CONTROL ROD group in a manner equivalent to opening the ETA relay.

L7 (continued)

ITS 3.3.4 ACTION C provides similar requirements for each inoperable ETA relay regardless of the total number of inoperable relays. CTS Table 3.5.1-1 Note 23 requires that with two or more inoperable relays, all relays associated with the channel, whether OPERABLE or inoperable, be opened within one hour. This requirement places unnecessary restrictions on the operation of the unit, by requiring OPERABLE equipment to be removed from service. The adoption of ITS 3.3.4 ACTION C is less restrictive in that it will not require any action to be taken to open a specific ETA relay if that relay is not inoperable. Removal of the requirement to open all ETA relays in a channel, in the event two or more ETA relays are inoperable, is acceptable because the Required Actions and Completion Times of ITS 3.3.4 Condition C provide requirements to compensate for the loss of function for each ETA relay individually within the same relatively short 1 hour time frame.

L8 CTS Table 3.5.1-1 Note 25 requires that any Control Rod Drive Trip Breaker with an inoperable undervoltage or shunt trip function be restored within 48 hours or the trip breaker must be opened. Adoption of ITS 3.3.4 Required Action A.2 will allow power to be removed from the inoperable trip breaker as an alternative to opening the breaker. This additional allowance is less restrictive in that it provides additional flexibility in dealing with trip breakers with an inoperable diverse trip function. The allowance for removing power from a trip breaker as an alternative to opening the breaker is currently allowed by CTS Table 3.5.1-1 Note 24. However, this note is not specifically applicable to the inoperability of the diverse trip function for the trip breakers. The addition of ITS 3.3.4 Required Action A.2 provides consistent ACTION requirements to compensate for inoperable CRD trip breakers whether or not the inoperability is due to failure of a diverse trip function. The Completion Times in ITS will remain, as they are in CTS, significantly different for a CRD trip breaker with an inoperable diverse trip function, as opposed to one which is inoperable for any other reason.

L9 The CTS requirement to check and calibrate the power range instruments against the incore instruments monthly was found in CTS Table 4.1-1 Item 4. The calibration was also required to be performed within some unspecified period of time after each startup if not performed within the previous week. These CTS requirements have been replaced with ITS SR 3.3.1.3 and SR 3.3.1.5. The adoption of these ITS SRs and their specified 31 day Frequencies represents less restrictive requirements in that the calibration will no longer be required following each startup if not performed within the previous week. Removal of the required calibration, within a week prior to or following each startup, is acceptable because deviation between the AXIAL POWER IMBALANCE indicated by the power range instruments and that indicated by the incore instruments generally occurs slowly. Adoption of the 31 day Frequency maintains requirements consistent with NUREG-1430.

L10 CTS Table 3.5.1-1 Note 2 indirectly provided a statement of Applicability for the OPERABILITY requirements for the source range and intermediate range instrument channels. This note provides a relaxation of action requirements when "2 of 4 power range instrument channels are greater than 10% rated power." The Applicability of ITS 3.3.9 and 3.3.10 does not require either the source range instrument channel or the intermediate range instrument channel to be maintained OPERABLE above MODE 2. This represents a relaxation of requirements, by removing the requirement to take actions in the event that either the required source range instrument channel or the required intermediate range instrument channel is inoperable, when above 5% RTP (ITS) but less than or equal to 10% rated power, as indicated on the power range instruments (CTS). This change also allows for less than a full decade of overlap between the PRMs and the IRMs, but continues to require IRMs until the PRMs are on-scale. This change is being made to provide clear statements of Applicability for these specifications which are consistent with the requirements of NUREG-1430.

L11 Not used.

L12 The Note modifying ITS 3.3.1.2 has been adopted. No specific allowance is provided in the ANO-1 CTS which removes the requirement to perform this calibration while in MODE 1 at low power levels. Adoption of this Note provides an exception to the performance of this calibration which recognizes the difficulty in its performance and the limitations of the calorimetric while operating at very low power levels. Below 20% RTP ANO-1 calculates heat balance power level based totally upon the primary system parameters. Above 20% RTP, the secondary system parameters are also considered since they are generally more accurate at higher power levels. By allowing the delay in performance of this calibration until RTP is above 20%, a generally more accurate calorimetric (one including secondary system parameters) is available. This allowance is being adopted to provide requirements consistent with NUREG-1430.

L13 The Note modifying ITS SR 3.3.1.3 has been adopted. This Note allows a delay in performance of this SR until the unit is above 20% RTP. This allowance is appropriate due to the usable range of the incore nuclear instruments which are required for the performance of this SR. Below about 20% the incore nuclear instruments are not capable of providing reliable accurate indication of AXIAL POWER IMBALANCE. Adoption of this Note provides a specific relaxation of requirements where none existed in CTS. This adoption is being made to provide requirements consistent with NUREG-1430.

L14 The CTS Table 4.1-1 Item 5 requirement, to perform CHANNEL FUNCTIONAL TEST on the intermediate range instrument channel, prior to each startup, if not performed within the previous week, has not been retained in the ITS. The requirement to perform this CHANNEL FUNCTIONAL TEST on a 31 day Frequency, was, however, retained as ITS 3.3.10.2. With the deletion of the required testing within 7 days of start-up, this testing will simply be required each 31 days. This 31 day frequency will also require the performance of the CHANNEL FUNCTIONAL TEST within 31 days of a start-up. This extension of the Frequency of this test from 7 days to 31 days prior to a start-up is acceptable based on operating experience which indicates that the intermediate range instrumentation is highly reliable and is not likely to experience an undetected failure during the extended period between tests.

L15 The requirement to perform a CHANNEL FUNCTIONAL TEST on the source range instrument channel within 7 days prior to start-up has not been retained in the ITS. This requirement was located in CTS 4.1-1 Item 6. This deletion has been made to provide testing requirements, for the required source range instrument channel, consistent with NUREG-1430.

A new requirement, to perform a CHANNEL CALIBRATION on the required source range instrument channel, on an 18 month Frequency, has been adopted in the ITS. No similar CHANNEL CALIBRATION requirements, for the source range instruments, existed in CTS. Because this calibration, by definition, encompasses the CHANNEL FUNCTIONAL TEST, performance of this calibration will ensure that testing, consistent with CTS requirements, continues to be required. The Frequency of this testing will, however, now be based strictly on the time since its last performance and not dependent upon whether or not the unit is in start-up. This change is acceptable, based on operating experience, which indicates that the source range instrumentation is highly reliable, and is no more susceptible to undetected failures within 7 days of start-up, than at any other time that the instrumentation is required to be OPERABLE.

The addition of the requirement to perform the CHANNEL CALIBRATION is discussed elsewhere in these Discussions of Change.

LESS RESTRICTIVE – ADMINISTRATIVE DELETION OF REQUIREMENTS

LA1 This information has been moved to the Bases or 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 an acceptable method of compliance or detail of unit design. 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. The details of performance of the surveillances have generally been relocated to the TRM along with various other requirements not appropriate for inclusion in the ITS. Changes to the TRM will be controlled by 10 CFR 50.59. This change is consistent with NUREG-1430.

CTS Location	New Location
1.5.6	Bases 3.3.1, SR 3.3.1.2
Table 2.3-1 Note (a)	Bases 3.3.1, BACKGROUND
Table 2.3-1 Note (b)	Bases 3.3.1, ASA
Table 2.3-1 Note (b)	Bases 3.3.1, LCO
Table 2.3-1 Note (b)	Bases 3.3.1, APPLICABILITY
Table 2.3-1 Note (c)	Bases 3.3.1, ASA
Table 2.3-1 Note (c)	Bases 3.3.1, LCO
Table 2.3-1 Note (c)	Bases 3.3.1, APPLICABILITY
Table 2.3-1 Note (d)	Bases 3.3.1, ASA
Table 2.3-1 Note (d)	Bases 3.3.1, LCO
Table 2.3-1 Note (d)	Bases 3.3.1, APPLICABILITY
3.5.1.3	Bases 3.3.1, BACKGROUND
Table 3.5.1-1 Note 7	Bases 3.3.9, BACKGROUND
Table 3.5.1-1 Note 7	Bases 3.3.10, BACKGROUND
Table 3.5.1-1 Note 23	Bases 3.3.4, LCO
Table 3.5.1-1 Note 26	Bases 3.3.4, BACKGROUND
Table 3.5.1-1 Columns 1 and 2	Bases 3.3.1, 3.3.2, 3.3.3, 3.3.4, 3.3.9, and 3.3.10, BACKGROUND
Table 4.1-1 Item 2 Note (1)	Bases 3.3.4, SR 3.3.4.1

3.3

1.5.4 Instrument Channel Calibration

An instrument channel calibration is a test, and adjustment (if necessary), to establish that the channel output responds with acceptable range and accuracy to known values of the parameter which the channel measures or an accurate simulation of these values. Calibration shall encompass the entire channel, including equipment actuation, alarm or trip and shall be deemed to include the channel test.

<LATER> (1.0)

1.5.5 Heat Balance Check

A heat balance check is a comparison of the indicated neutron power and core thermal power.

LATER

1.5.6 Heat Balance Calibration

An adjustment of the power range channel amplifiers output to agree with the core thermal power as determined by a weighted primary and secondary heat balance considering all heat losses. Between 0 and 15% power, only the primary heat balance is considered. From 15 to 100% power the heat balance is weighted linearly with only the secondary heat balance being considered at 100% power.

LA1 Bases

1.6 POWER DISTRIBUTION

1.6.1 Quadrant Power Tilt

Quadrant power tilt shall be defined by the following equation and is expressed as a percentage

<LATER> (1.0)

$$100\left(\frac{\text{Power in any core quadrant}}{\text{Average power of all quadrants}} - 1\right)$$

LATER

1.6.2 Reactor Power Imbalance

Reactor power imbalance is the power in the top half of the core minus the power in the bottom half of the core expressed as a percentage of rated power. Imbalance is monitored continuously by the RPS using input from the power range channels. Imbalance limits are defined in Specification 2.1 and imbalance setpoints are defined in Specification 2.3.

2.3 LIMITING SAFETY SYSTEM SETTINGS, PROTECTIVE INSTRUMENTATION

A1

Applicability

Applies to instruments monitoring reactor power, reactor power imbalance, reactor coolant system pressure, reactor coolant outlet temperature, flow, number of pumps in operation, and high reactor building pressure.

Objective

To provide automatic protection action to prevent any combination of process variables from exceeding a safety limit.

Specification

LCO 3.3.1

2.3.1 The reactor protection system trip setting limits and the permissible bypasses for the instrument channels shall be as stated in Table 2.3-1 and the Protection System Maximum Allowable Setpoint for Axial Power Imbalance as given in the COLR.

LA1

Bases

Bases

A2

The reactor protection system consists of four instrument channels to monitor each of several selected plant conditions which will cause a reactor trip if any one of these conditions deviates from a pre-selected operating range to the degree that a safety limit may be reached.

The trip setting limits for protection system instrumentation are listed in Table 2.3-1. The safety analysis has been based upon these protection system instrumentation trip setpoints plus calibration and instrumentation errors.

Nuclear Overpower

A reactor trip at high power level (neutron flux) is provided to prevent damage to the fuel cladding from reactivity excursions too rapid to be detected by pressure and temperature measurements.

During normal plant operation with all reactor coolant pumps operating, reactor trip is initiated when the reactor power level reaches 104.9 percent of rated power. Adding to this the possible variation in trip setpoints due to calibration and instrument errors, the maximum actual power at which a trip would be actuated could be 112%, which is the value used in the safety analysis.

A. Overpower Trip Based on Flow and Imbalance

The power level trip set point produced by the reactor coolant system flow is based on a power-to-flow ratio which has been established to accommodate the most severe thermal transient considered in the design, the loss-of-coolant-flow accident from high power. Analysis has demonstrated that the specified power-to-flow ratio is adequate to prevent a DNBR of less than 1.30 (BAW-2) or 1.18 (BWC) should a low flow condition exist due to any electrical malfunction.

A2

The power level trip setpoint produced by the power-to-flow ratio provides both high power level and low flow protection in the event the reactor power level increases or the reactor coolant flow rate decreases. The power level trip setpoint produced by the power-to-flow ratio provides overpower DNB protection for all modes of pump operation. For every flow rate there is a maximum permissible power level, and for every power level there is a minimum permissible low flow rate.

The flux/flow ratios account for the maximum calibration and instrumentation errors and the maximum variation from the average value of the RC flow signal in such a manner that the reactor protective system receives a conservative indication of the RC flow.

No penalty in reactor coolant flow through the core was taken for an open core vent valve because of the core vent valve surveillance program during each refueling outage. For safety analysis calculations the maximum calibration and instrumentation errors for the power level were used.

The power-imbalance boundaries are established in order to prevent reactor thermal limits from being exceeded. These thermal limits are either power peaking kw/ft limits or DNBR limits. The reactor power imbalance (power in top half of core minus power in the bottom half of core) reduces the power level trip produced by the power-to-flow ratio so that the boundaries of the Protection System Maximum Allowable Setpoints for Axial Power Imbalance in the COLR are produced. The power-to-flow ratio reduces the power level trip associated reactor power-to-reactor power imbalance boundaries by the value specified in the COLR for a 1 percent flow reduction.

B. Pump Monitors

In conjunction with the power imbalance/flow trip, the pump monitors prevent the minimum core DNBR from decreasing below 1.30 (BAW-2) or 1.18 (BWC) by tripping the reactor due to the loss of reactor coolant

3.3.1

A2

pumps(s). The pump monitors also restrict the power level for the number of pumps in operation.

C. RCS Pressure

During a startup accident from low power or a slow rod withdrawal from high power, the system high pressure trip is reached before the nuclear overpower trip setpoint. The trip setting limit shown in Table 2.3-1 for high reactor coolant system pressure (2355 psig) has been established to maintain the system pressure below the safety limit (2750 psig) for any design transient.(2)

The low pressure (1800 psig) and variable low pressure (GOLR) trip setpoint shown in Table 2.3-1 have been established to maintain the DNB ratio greater than or equal to the minimum allowable DNB ratio for those design accidents that result in a pressure reduction.(2,3)

To account for the calibration and instrumentation errors, the accident analysis used the protective limit specified in the COLR.

D. Coolant Outlet Temperature

The high reactor coolant outlet temperature trip setting limit (618F) shown in Table 2.3-1 has been established to prevent excessive core coolant temperatures in the operating range. Due to calibration and instrumentation errors, the safety analysis used a trip setpoint of 620 F.

E. Reactor Building Pressure

The high reactor building pressure trip setting limit (4 psig) provides positive assurance that a reactor trip will occur in the unlikely event of a steam line failure in the reactor building or a loss-of-coolant accident, even in the absence of a low reactor coolant system pressure trip.

F. Shutdown Bypass

In order to provide for control rod drive tests, zero power physics testing, and startup procedures, there is provision for bypassing certain segments of the reactor protection system. The reactor protection system segments which can be bypassed are shown in Table 2.3-1. Two conditions are imposed when the bypass is used:

1. A nuclear overpower trip set point of ≤5.0 percent of rated power is automatically imposed during reactor shutdown.

2. A high reactor coolant system pressure trip setpoint of 1720 psig is automatically imposed.

A2

The purpose of the 1720 psig high-pressure trip setpoint is to prevent normal operation with part of the reactor protection system bypassed. This high-pressure trip setpoint is lower than the normal low-pressure trip setpoint so that the reactor must be tripped before the bypass is initiated. The overpower trip setpoint of ≤5.0 prevents any significant reactor power from being produced when performing the physics tests. Sufficient natural circulation (5) would be available to remove 5.0 percent of rated power if none of the reactor coolant pumps were operating.

REFERENCES

(1) FSAR, Section 14.1.2.3

(2) FSAR, Section 14.1.2.2

(3) FSAR, Section 14.1.2.7

(4) FSAR, Section 14.1.2.8

(5) FSAR, Section 14.1.2.6

A1

~~Table 2.3-1~~
~~Reactor Protection System Trip Setting Limits~~

Table 3.3.1-1 Allowable Values

Function #

<LATER (3.4A)>

LATER

Function #		~~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
1.a/1.b	Nuclear power, % of rated, max	104.9	104.9	104.9	5.0[a]
8	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
7	Nuclear Power based on pump monitors, % of rated, max[c]	NA	NA	55	Bypassed
3/11	High RC system pressure, psig, max	2355	2355	2355	1720[a]
4	Low RC system pressure, psig. min	1800	1800	1800	Bypassed
5	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
2	RC temp, F, max	618	618	618	618
6	High reactor building pressure, ~~psig~~, max	4 (18.7 psia)	4 (18.7 psia)	4 (18.7 psia)	4 (18.7 psia)

A1

A12

LA1 Bases

~~(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.~~

<LATER (3.4A)> ~~(d) Operation with one Reactor Coolant Pump operating in each loop is limited to 24 hrs. with the reactor critical.~~ LATER

<Add Table 3.3.1-1 Allowable Values for Functions 9 & 10> M19

<Add Table 3.3.1-1 Note (a)> A1

3.3.1

3.3.1
3.3.4
3.3.10

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

3.5.1.1 Startup and operation are not permitted unless the requirements 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 be the limit given under Table 3.5.1-1, Columns 3 and 4, operation shall be limited as specified in Column 5.

A3 & LATER

& (LATER) (3.3B, 3.3C, 3.3D, 3.4B)

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.

LA1 BASES

3.3.1 RA B.1 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.

A3

Only one channel bypass key shall be accessible for use in the control room.

LA1 BASES

3.3.1 RA A.1 & A.2
3.3.1 RA B.2.1 & B.2.2 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.

M13

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

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.

SR 3.3.10.4

A1

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.

3.3.10 RA A.1 & A.2

immediately suspend positive reactivity additions and open CRD trip breakers within 1 hr.

M10

3.5.1.6 In the event that one of the trip devices in either of the source 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.

3.3.4 RA B.2

1 hr

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.

L1

3.5.1.7 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. — LATER

<LATER> (3.4B)

3.5.1.8 The degraded voltage monitoring relay settings shall be as follows: — LATER

<LATER> (3.3D)

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.

3.5.1.9 The following Reactor Trip circuitry shall be operable as indicated: — A1

Table 3.3.1-1, Function 10, "Applicable MODES"

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.) — A1, L3

Table 3.3.1-1, Function 9 "Applicable MODES"

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.) — A1, L3

Table 3.3.1-1, Cond. ref. from RA C.1 Function 10 - 3.3.1 RA G.1 Function 9 - 3.3.1 RA F.1

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.~~ — Cannot be met, reduce power to ≤ 10% RTP, Reduce power to ≤ 45% RTP — L3

3.5.1.10 ~~Deleted~~ — A1

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> (3.3C)

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

<LATER> (3.3D)

3.3.1
3.3.4
3.3.10

Bases

Every reasonable effort will be made to maintain all safety instrumentation in operation. A startup is not permitted unless the requirements of Table 3.5.1-1, Columns 3 and 4, are met.

Operation at rated power is permitted as long as the systems have at least the redundancy requirements of Column 4 (Table 3.5.1-1). This is in agreement with redundancy and single failure criteria of IEEE 279 as described in FSAR, Section 7.

There are four reactor protection channels. Normal trip logic is two-out-of-four. Required trip logic for the power range instrumentation channels is two-out-of-three. Minimum trip logic on other instrumentation channels is one-out-of-two.

The four reactor protection channels were provided with key operated bypass switches to allow on-line testing or maintenance on only one channel at a time during power operation. Each channel is provided with alarm and lights to indicate when that channel is bypassed. There will be one reactor protection system channel bypass switch key permitted in the control room. Upon the discovery of inoperable functions in any one reactor protection channel, the effect of the failure on the reactor protection syst and other interconnected systems is evaluated. The affected reactor protect channel may be placed in channel bypass, remain in operation in a degraded condition, or placed in the tripped condition as determined by operating conditions and management judgment. This action allows placing the plant in the safest condition possible considering the extent of the failure, plant conditions, and guidance from plant management. Should the failure in the reactor protection channel prohibit the proper operation of another system, the appropriate actions for the affected system are implemented. Administra controls are established to preclude placing a reactor protection channel ir channel bypass when any other reactor protection channel contains an inopera function in the untripped state. A2

Each reactor protection channel key operated shutdown bypass switch is provided with alarm and lights to indicate when the shutdown bypass switch is being used. R TRM

The source range and intermediate range nuclear flux instrumentation scales overlap by one decade. This decade overlap will be achieved at 10^{-10} amps c the intermediate range scale.

The ESAS employs three independent and identical analog channels, which supply trip signals to two independent, identical digital subsystems. In order to actuate the safeguards systems, two out of three analog channels must trip. This will cause both digital subsystems to trip. Tripping of either digital subsystem will actuate all safeguards systems associated with that digital subsystem. A2

Because only one digital subsystem is necessary to actuate the safeguards systems and these systems are capable of tripping even when they are being tested, a single failure in a digital subsystem cannot prevent protective action.

3.3.1
3.3.4
3.3.10

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). A2

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.

3.3.1
3.3.4
3.3.10

Power is normally supplied to the control rod drive mechanisms from two separate parallel 480 volt sources. Redundant trip devices are employed in each of these sources. If any one of these trip devices fails in the untripped state, on-line repairs to the failed device, when practical, will be made and the remaining trip devices will be tested. Four hours is ample time to test the remaining trip devices and, in many cases, make on-line repairs.

The Degraded Voltage Monitoring relay settings are based on the short term starting voltage protection as well as long term running voltage protection. The 4.16 KV undervoltage relay setpoints are based on the allowable starting voltage plus maximum system voltage drops to the motor terminals, which allows approximately 78% of motor rated voltage at the motor terminals. The 460V undervoltage relay setpoint is based on long term motor voltage requirements plus the maximum feeder voltage drop allowance resulting in a 92% setting of motor rated voltage.

A2

The OPERABILITY of the accident monitoring instrumentation ensures that sufficient information is available on selected plant parameters to monitor and assess these variables during and following an accident. This capability is consistent with the recommendation of Regulatory Guide 1.97, "Instrumentation for Light-Water-Cooled Nuclear Power Plants to Assess Plant Conditions During and Following an Accident," December 1975 and NUREG-0578, "TMI-2 Lessons Learned Task Force Status Report and Short-Term Recommendations."

The subcooled margin monitors (SMM), and core-exit thermocouples (CET), Reactor Vessel Level Monitoring System (RVLMS) and Hot Leg Level Measurement System (HLLMS) are a result of the Inadequate Core Cooling (ICC) instrumentation required by Item II.F.2 NUREG-0737. The function of the ICC instrumentation is to increase the ability of the plant operators to diagnose the approach to and recovery from ICC. Additionally, they aid in tracking reactor coolant inventory. These instruments are included in the Technical Specifications at the request of NRC Generic Letter 83-37 and are not required by the accident analysis, nor to bring the plant to cold shutdown conditions. The Reactor Vessel Level Monitor is provided as a means of indicating level in the reactor vessel during accident conditions. The channel operability of the RVLMS is defined as a minimum of three sensors in the upper plenum region and two sensors in the dome region operable. When Reactor Coolant Pumps are running, all except the dome sensors are interlocked to read "invalid" due to flow induced variables that may offset the sensor outputs. The channel operability of the HLLMS is defined as a minimum of one wide range and any two of the narrow range transmitters in the same channel operable. If the equipment is inaccessible due to health and industrial safety concerns (for example, high radiation area, low oxygen content of the containment atmosphere) or due to physical location of the fault (for example, probe failure in the reactor vessel), then operation may continue until the next scheduled refueling outage and a report filed.

3.3.1
3.3.4
3.3.10

The principal function of the Control Room Isolation-High Radiation is to provide an enclosed environment from which the unit can be operated following an uncontrolled release of radioactivity. Due to the unique arrangement of the shared control room envelope, one control room isolation channel receives a high radiation signal from the ANO-1 control room ventilation intake duct monitor and the redundant channel receives a high radiation signal from the ANO-2 control room ventilation intake duct monitor. With no channel of the control room radiation monitoring system operable, the CREVS must be placed in a condition that does not require the isolation to occur (i.e., one operable train of CREVS is placed in the emergency recirculation mode of operation). Reactor operation may continue indefinitely in this state.

To support loss of main feedwater analyses, steam line/feedwater line break analyses, SBLOCA analyses, and NUREG-0737 requirements, the EFIC system is designed to automatically initiate EFW when:

1. all four RC pumps are tripped
2. both main feedwater pumps are tripped
3. the level of either steam generator is low
4. either steam generator pressure is low
5. ESAS ECCS actuation (high RB pressure or low RCS pressure)

The EFIC system is also designed to isolate the affected steam generator on a steam line/feedwater line break and supply EFW to the intact generator according to the following logic: A2

- If both SG's are above 600 psig, supply EFW to both SG's.
- If one SG is below 600 psig, supply EFW to the other SG.
- If both SG's are below 600 psig, but the pressure difference between the two SG's exceeds 100 psig, supply EFW only to the SG with the higher pressure.
- If both SG's are below 600 psig and the pressure difference is less than 100 psig, supply EFW to both SG's.

At cold shutdown conditions all EFIC initiate and isolate functions are bypassed except low steam generator level initiate. The bypassed functions will be automatically reset at the values or plant conditions identified in Specification 3.5.1.15. "Loss of 4 RC pumps" initiate and "low steam generator pressure" initiate are the only shutdown bypasses to be manually initiated during cooldown. If reset is not done manually, they will automatically reset. Main feedwater pump trip bypass is automatically removed above 10% power.

<u>REFERENCE</u>

FSAR, Section 7.1

<Add 3.3.1 Appl.> — A1

<Add 3.3.1 Condition C> — M18

<Add 3.3.1 Condition E> — M11

<Add 3.3.1 Surveillance Requirements – NOTE> — A1

LA1 BASES

3.3.1 LCO — Table 3.5.1-1 Instrumentation Limiting Conditions for Operation (Note 6)

A3

A7

REACTOR PROTECTION SYSTEM

A3

A6

	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
3.3.2 LCO	1. Manual pushbutton	1	1	1	0	Note 1 (M1)
T3.3.1-1, #1a	2. ~~Power range~~ nuclear overpower instrument channel	4	2	~~3 (Note 4)~~ 2	~~1 (Note 4)~~ 0	Note 1 (L5)
3.3.10 LCO	3. Intermediate range instrument channels	2	Note 7	1	0	Notes ~~1,~~ 2 (A1)
3.3.9 LCO	4. Source range instrument channels	2	Note 7	1	0	Notes ~~1,~~ 2, 3 (M8)
T3.3.1-1, #2	5. Reactor coolant temperature instrument channels	4	2	2	1	Note 1
T3.3.1-1, #5	6. Pressure-temperature instrument channels	4	2	2	1	Note 1
T3.3.1-1, #8	7. Flux/imbalance/flow instrument channels	4	2	2	1	Note 1
	8. Reactor coolant pressure					
T3.3.1-1, #3	a. High reactor coolant pressure instrument channels	4	2	2	1	Note 1 (A1)
T3.3.1-1, #4	b. Low reactor coolant pressure instrument channels	4	2	2	1	Note 1
T3.3.1-1, #7	9. Power/number of pumps instrument channels	4	2	2	1	Note 1
T3.3.1-1, #6	10. High reactor building pressure channels	4	2	2	1	Note 1

<Add Table 3.3.1-1 APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS for Functions 1 thru 8 & 11, including T3.3.1-1 Notes a, b, c & d.> — M12

3.3.1
3.3.2
3.3.9
3.3.10

<Add Table 3.3.1-1 Function 1b, Nuclear Overpower Low Setpoint, Condition & SRs> M11
<Add Table 3.3.1-1 Function 11, Shutdown Bypass RCS High Pressure, Condition & SRs> LA1 BASES

Amendment No. 6Z, 117, 44a

Table 3.5.1-1 (Cont'd) A3 A7

REACTOR PROTECTION 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
T3.3.1-1, function 10 — 11. Reactor trip upon loss of Main Feedwater	4	2	2	1	Notes 1, 15 (A1)
function 9 — 12. Reactor trip upon turbine trip	4	2	2	1	Notes 1, 16 (A1)
3.3.4 LCO c — 13. Electronic (SCR) Trip Relay	2	2	2	0	Note 23 (A8)
14. Control Rod Drive Trip Breakers					
3.3.4 LCO a — A. AC Breakers	2	2	2	0	Notes 24, 25 (A1)
3.3.4 LCO b — B. DC Breakers (Note 26)	2	2	2	0	Notes 24, 25 (A1)

LA1 BASES

<Add 3.3.2 Appl, Conditions A & C> M1
<Add 3.3.3 LCO, Appl, and ACTIONS> M14
<Add 3.3.4 ACTIONS - NOTE> A9
<Add 3.3.4 Appl, Conditions D & E> M15
<Add 3.3.9 Appl, Condition A> M16

3.3.1
3.3.2
3.3.3
3.3.4
3.3.9

A3

MODE 3 with the CRD breakers open within 6 hrs.

M3 & LATER

TABLE 3.5.1-1 (Cont'd)

3.3.1 Cond. D

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.

3.3.2 Cond. B & <LATER>(3.3B/C/D & 3.4B)

L10

2. When 2 of 4 power range instrument channels are greater than ~~10%~~ 5% rated power, hot shutdown is not required.

3.3.9 Appl. 3.3.10 Appl.

M7

3. When 1 of 2 intermediate range instrument channels is greater than 10-10 amps, ~~hot shutdown is not required.~~ initiate action to restore source range channel within 1 hr.

3.3.9 Cond. B

L5

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

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.~~

<LATER> (3.3B, 3.4B)

A1 & LATER

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.

3.3.1 RA B.1 & <LATER> (3.3B & 3.3C)

LA1 BASES

7. ~~These channels initiate control rod withdrawal inhibits not reactor trips at ~10% rated power. Above 10% rated power, those inhibits are bypassed.~~

LATER

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> (3.3B)

LATER

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 3 applies.~~

<LATER> (3.3D)

3.3.1 3.3.2 3.3.9 3.3.10

(A3)

TABLE 3.5.1-1 (Cont'd)

<LATER> (3.3D)

12. With the number of operable channels less than required, either return the indicator to operable status within 24 hours, or verify the block valve closed and power removed within an additional 24 hours. If the block valve cannot be verified closed within the additional 24 hours, de-energize the electromatic relief valve power supply within the following 12 hours.

13. Channels may be bypassed for not greater than 30 seconds during reactor coolant pump starts. If the automatic bypass circuit or its alarm circuit is inoperable, the undervoltage protection shall be restored within 1 hour, otherwise, Note 14 applies.

LATER

<LATER> (3.3D + 3.8)

14. With the number of channels less than required, restore the inoperable channels to operable status within 72 hours or be in HOT SHUTDOWN within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

LATER

Table 3.3.1-1 Function 10 Appl Function 9 Appl + <LATER> (3.3C)

15. This trip function may be bypassed at up to 10% reactor power.

16. This trip function may be bypassed at up to 45% reactor power.

(L3) + LATER

<LATER> (3.3D)

17. With no channel operable, within 1 hour initiate and maintain operation of the control room emergency ventilation system in the recirculation mode of operation.

18. With one channel inoperable, restore the inoperable channel to operable status within 7 days or within the next 6 hours initiate and maintain operation of the control room emergency ventilation system in the recirculation mode of operation.

LATER

<LATER> (3.3C)

19. This function may be bypassed below 750 psig OTSG pressure. Bypass is automatically removed when pressure exceeds 750 psig.

LATER

<LATER> (3.3D)

20. With one channel inoperable, (1) either restore the inoperable channel to operable status within 7 days, <u>or</u> (2) prepare and submit a Special Report to the Commission pursuant to Specification 6.12.5 within 30 days following the event, outlining the action taken, the cause of the inoperability, and the plans and schedule for restoring the system to operable status. With both channels inoperable, initiate alternate methods of monitoring the containment radiation level within 72 hours in addition to the actions described above.

21. With one channel inoperable, restore the inoperable channel to operable status within 30 days or be in hot shutdown within 72 hours unless containment entry is required. If containment entry is required, the inoperable channel must be restored by the next refueling outage. If both channels are inoperable, restore the inoperable channels within 30 days or be in HOT SHUTDOWN within 12 hours.

LATER

<Add 3.3.4 RA C.1, C.2, & C.4> A3 M5 L7

Table 3.5.1-1 (cont'd)

23. 3.3.4 RA C.3 ~~With the number of operable Electronic (SCR) Trip relays one less than the total number of Electronic (SCR) Trip relays in a channel, restore the inoperable Electronic (SCR) Trip relay to operable status in 48 hours or place the SCRs associated with the inoperable Electronic (SCR) Trip relay in trip in the next hour. With two or more Electronic (SCR) Trip relays inoperable, place all Electronic (SCR) Trip relays associated with that channel in trip in the next hour.~~ ~~This requirement does not apply to the Electronic Trip channels associated with Group 8 Regulating Power Supply.~~ LA1 Bases

24. With the number of OPERABLE channels one less than the Total Number of Channels, STARTUP and/or POWER OPERATION may proceed provided the following conditions are satisfied: A1

3.3.4 RA B.1 a. Within 1 hour:

3.3.4 RA B.2 1. Place the inoperable channel in the tripped condition, or

2. Remove power supplied to the control rod trip device associated with the inoperable channel.

~~b. One additional channel may be bypassed for up to 4 hours for surveillance testing and the in operable channel above may be bypassed for up to 30 minutes in any 24-hour period when necessary [sic] to test the trip breaker associated with the logic of the channel being tested. The inoperable channel above shall not be bypassed to test the logic of a channel of the trip system associated with the inoperable channel.~~ A13

3.3.4 Cond. A 25. With one of the Control Rod Drive Trip Breaker diverse trip features (undervoltage or shunt trip attachment) inoperable, restore it to OPERABLE status in 48 hours or place the breaker in trip ~~in the next hour.~~ Or remove power from the trip breaker M6 L8

26. ~~Interrupts motor power to the Safety Groups of control rods only.~~ LA1 Bases

27. ~~Deleted~~ A1

3.3.4

3.3.1
3.3.2
3.3.3
3.3.4
3.3.9
3.3.10

SURVEILLANCE REQUIREMENTS (Continued)

4.0.5 (Continued)

b. Surveillance intervals specified in Section XI of the ASME Boiler and Pressure Vessel Code and applicable Addenda for the inservice inspection and testing activities required by the ASME Boiler and Pressure Vessel Code and applicable Addenda shall be applicable as follows in these Technical Specifications:

ASME Boiler and Pressure Vessel Code and applicable Addenda terminology for inservice inspection and testing activities	Required frequencies for performing inservice inspection and testing activities
Weekly	At least once per 7 days
Monthly	At least once per 31 days
Quarterly or every 3 months	At least once per 92 days
Semiannually or every 6 months	At least once per 184 days
Yearly or annually	At least once per 366 days

LATER

<LATER> (5.0)

c. The provisions of Specification 4.0.2 are applicable to the above required frequencies for performing inservice inspection and testing activities.

d. Performance of the above inservice inspection and testing activities shall be in addition to other specified Surveillance Requirements.

e. Nothing in the ASME Boiler and Pressure Vessel Code shall be construed to supersede the requiremetns of any Technical Specification.

4.1 OPERATIONAL SAFETY ITEMS

Applicability

Applies to items directly related to safety limits and limiting conditions for operation.

A1

Objective

To specify the minimum frequency and type of surveillance to be applied to unit equipment and conditions

Specification

a. The minimum frequency and type of surveillance required for reactor protective system and engineered safeguards system instrumentation when the reactor is critical shall be as stated in Table 4.1-1. A3 + ® TRM

3.3.1
3.3.2
3.3.3
3.3.4
3.3.9
3.3.10

OPERATIONAL SAFETY ITEMS (continued) A1

4.1 (Continued)

& R TRM

A3

& <LATER> (3.3B, 3.3C, 3.3D)

b. Equipment and sampling test shall be performed as detailed in Table 4.1-2 and 4.1-3.

c. Discrepancies noted during surveillance testing will be corrected and recorded.

<LATER> (3.2)

d. A power distribution map shall be made to verify the expected power distribution at periodic intervals at least every 10 effective full power days using the incore instrumentation detector system.

LATER

A2

BASES

4.0.1 through 4.0.5 Establish the general requirements applicable to Surveillance Requirements. These requirements are based on the Surveillance Requirements stated in the Code of Federal Regulations, 10CFR 50.36(c)(3):

"Surveillance Requirements are requirements relating to test, calibration, or inspection to ensure that the necessary quality of systems and components is maintained, that facility operation will be within safety limits, and that the limiting conditions of operation will be met."

4.0.1 Establishes the requirement that surveillances must be performed during the operational modes or other conditions for which the requirements of the Limiting Conditions for Operation apply unless otherwise stated in an individual Surveillance Requirement. The purpose of this specification is to ensure that surveillances are performed to verify the operational status of systems and components and that parameters are within specified limits to ensure safe operation of the facility when the plant is in a mode or other specified condition for which the associated Limiting Conditions for Operation are applicable. Surveillance Requirements do not have to be performed when the facility is in an operational mode for which the requirements of the associated Limiting Condition for Operation do not apply unless otherwise specified.

3.3.1
3.3.2
3.3.3
3.3.4
3.3.9
3.3.10

BASES (continued)

Under the terms of this specification, the more restrictive requirements of the Technical Specifications take precedence over the ASME Boiler and Pressure Vessel Code and applicable Addenda. The requirements of Specification 4.0.4 to perform surveillance activities before entry into an operational mode or other specified condition takes precedence over the ASME Boiler and Pressure Vessel Code provision which allows pumps and valves to be tested up to one week after return to normal operation. The Technical Specification definition of OPERABLE does not allow a grace period before a component, that is not capable of performing its specified function, is declared inoperable and takes precedence over the ASME Boiler and Pressure Vessel Code provision which allows a valve to be incapable of performing its specified function for up to 24 hours before being declared inoperable. A2

4.1 Bases

Check

Failures such as blown instrument fuses, defective indicators, faulted amplifiers which result in "upscale" or "downscale" indication can be easily recognized by simple observation of the functioning of an instrument or system. Furthermore, such failures are, in many cases, revealed by alarm or annunciator Action. Comparison of output and/or state of independent channels measuring the same variable supplements this type of built-in surveillance. Based on experience in operation of both conventional and nuclear plant systems, when the plant is in operation, the minimum checking frequency stated is deemed adequate for reactor system instrumentation. A2 & R TRM

Calibration

Calibration shall be performed to assure the presentation and acquisition of accurate information. The nuclear flux (power range) channels shall be calibrated at least twice weekly (during steady state operating conditions) against a heat balance standard to compensate for instrumentation drift. During nonsteady state operation, the nuclear flux channels shall be calibrated daily to compensate for instrumentation drift and changing rod patterns and core physics parameters. A2

3.3.1
3.3.2
3.3.3
3.3.4
3.3.9
3.3.10

Other channels are subject only to "drift" errors induced within the instrumentation itself and, consequently, can tolerate longer intervals between calibrations. Process system instrumentation errors induced by drift can be expected to remain within acceptable tolerances if recalibration is performed once every 18 months.

Substantial calibration shifts within a channel (essentially a channel failure) will be revealed during routine checking and testing procedures.

Thus, minimum calibration frequencies for the nuclear flux (power range) channels, and once every 18 months for the process system channels is considered acceptable.

Testing

On-line testing of reactor protective channel and EFIC channels is required once every 4 weeks on a rotational or staggered basis. The rotation scheme is designed to reduce the probability of an undetected failure existing within the system and to minimize the likelihood of the same systematic test errors being introduced into each redundant channel.

All reactor protective channels will be tested before startup if the individual channel rotational frequency has been discontinued or if outage activities could potentially have affected the operability of one or more channels. A rotation will then be established to test the first Channel one week after startup, the second Channel two weeks after startup, the third Channel three weeks after startup, and the fourth Channel four weeks after startup. A2

The established reactor protective system instrumentation and EFIC test cycle is continued with one channel's instrumentation tested each week. Upon detection of a failure that prevents trip action, all instrumentation associated with the protective channels will be tested after which the rotational test cycle is started again. If actuation of a safety channel occurs, assurance will be required that actuation was within the limiting safety system setting.

The protective channels coincidence logic and control rod drive trip breakers are trip tested every quarter. The trip test checks all logic combinations and is to be performed on a rotational basis. The logic and breakers of the four protective channels shall be trip tested prior to startup and their individual channels trip tested on a cyclic basis. Discovery of a failure requires the testing of all channel logic and breakers, after which the trip test cycle is started again.

3.3.1
3.3.2
3.3.3
3.3.4
3.3.9
3.3.10

~~The equipment testing and system sampling frequencies specified in Table 4.1-2 and Table 4.1-3 are considered adequate to maintain the status of the equipment and systems to assure safe operation.~~ A2

& P
TRM

~~**REFERENCE**~~

~~FSAR Section 7.1.2.3.4~~

<Add SR 3.3.1.2 - NOTE > L12

<Add SR 3.3.1.3 - NOTE > L13

<Add SR 3.3.1.6 - NOTE > A10

A3

Table 4.1-1
Instrument Surveillance Requirements

	Channel Description	Check	Test	Calibrate	Remarks
SR 3.3.3.1	1. ~~Protective Channel Coincidence Logic~~ Reactor Trip Module	NA	Q	NA	
SR 3.3.4.1	2. Control Rod Drive Trip Breaker	NA	Q(1)	NA	~~(1) To include independent testing of the shunt and undervoltage trip attachments.~~
SR 3.3.1.2	3. Power Range Amplifier	NA	NA	~~T/W(1)~~ Adjust Power range Channel output if Calorimetric exceeds power range channel by ≥ 2% RTP.	(1) Heat balance calibration ~~twice weekly under steady state operating conditions, daily~~ every 96 hours 24 hours under non-steady state operating conditions.
SR 3.3.1.5, SR 3.3.1.1, SR 3.3.1.3, Table 3.3.1-1 Function 1a + Function 8	4. ~~Power Range Channel~~ Nuclear Overpower and Nuclear Overpower RCS Flow and measured AXIAL POWER IMBALANCE	S M(1)	~~M~~	M(1)(2)	(1) Using core instrumentation. (2) Axial offset upper and lower chambers ~~monthly and after each startup if not done previous week.~~ every 31 days
SR 3.3.10.1, SR 3.3.10.2	5. Intermediate Range Channel	S	~~P~~ M	~~NA~~ R	
SR 3.3.9.1	6. Source Range Channel	S(1)	~~P~~	~~NA~~ R	~~(1) When in service.~~
Table 3.3.1-1 Function 2	7. Reactor Coolant Temperature Channel	S	M	R	
Function 3	8. High Reactor Coolant Pressure Channel	S	M	R	
Function 4	9. Low Reactor Coolant Pressure Channel	S	M	R	
Function 8	10. Flux-Reactor Coolant Flow Comparator	S	~~M~~	R	
Function 5	11. Reactor Coolant Pressure Temperature Comparator	S	M	R	
Function 7	12. Pump Flux Comparator	S	M	R	
		SR 3.3.1.1	SR 3.3.1.4	SR 3.3.1.6	

Add SR 3.3.9.2 and SR 3.3.10.3 with NOTES

A1 LA1 Bases A1 M4 L2 A5 L9 A11 A6 A1 M9 L14 L15 A1 A11

3.3.1
3.3.3
3.3.4
3.3.9
3.3.10

A3

Table 4.1-1 (cont.)

	Channel Description	Check	Test	Calibrate	Remarks
Table 3.3.1-1 Function 6	13. High Reactor Building Pressure Channel	S SR 3.3.1.1	M SR 3.3.1.4	R SR 3.3.1.6	A1
	14. High Pressure Injection Logic Channel	NA	M	NA	
	15. High Pressure Injection Analog Channels				
<LATER> (3.3B, 3.3D)	a. Reactor Coolant Pressure Channel	S	M(1)	R	(1) Including test of shutdown bypass function (ECCS bypass function).
	b. Reactor Building 4 psig Channel	S	M	R	
<LATER> (3.3B)	16. Low Pressure Injection Logic Channel	NA	M	NA	
	17. Low Pressure Injection Analog Channels				
<LATER> (3.3B, 3.3D)	a. Reactor Coolant Pressure Channel	S	M(1)	R	(1) Including test of shutdown bypass function (ECCS bypass function).
	b. Reactor Building 4 psig Channel	S	M	R	
	18. Reactor Building Emergency Cooling and Isolation System Logic Channel	NA	M	NA	
	19. Reactor Building Emergency Cooling and Isolation System Analog Channels				
	a. Reactor Building 4 psig Channels	S	M	R	

LATER

70

A3

Table 4.1-1 (Cont.)

	Channel Description	Check	Test	Calibrate	Remarks	
<LATER> (3.3 D)	29. High and Low Pressure Injection Systems: Flow Channels	NA	NA	R		LATER
<LATER> (3.4 B)	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.	LATER
	31. Deleted					A1
<LATER> (3.8)	32. Diesel generator protective relaying starting interlocks and circuitry	M	Q	NA		LATER
<LATER> (3.8)	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.	LATER
<LATER> (3.3 D)	34. Borated water storage tank level indicator	W	NA	R		LATER
Table 3.3.1-1 Function 10	35. Reactor trip upon loss of main feedwater circuitry	M 12 hour SR 3.3.1.1	PC 31 days SR 3.3.1.4	R SR 3.3.1.6		M2

3.3.1

(A3)

Table 4.1-1 (Cont.)

	Channel Description	Check	Test	Calibrate	Remarks	
⟨LATER⟩ (3.5)	36. Boric Acid Addition Tank					
	a. Level Channel	NA	NA	R		LATER
	b. Temperature Channel	M	NA	R		LATER
⟨LATER⟩ (3.30)	37. Degraded Voltage Monitoring	W	R	R		(R) TRM
	38. Sodium Hydroxide Tank Level Indicator	NA	NA	R		
⟨LATER⟩ (3.2)	39. Incore Neutron Detectors	M(1)	NA	NA	(1) Check Functioning	LATER
⟨LATER⟩ (3.30)	40. Emergency Plant Radiation Instruments	M(1)	NA	R	(1) Battery Check	LATER
Table 3.3.1-1 Function 9	41. Reactor Trip Upon Turbine Trip Circuitry	M 12 hours SR 3.3.1.1	PC 31 days SR 3.3.1.4	R SR 3.3.1.6		M 2
	42. Deleted					A1

3.3.1

(A3)

Table 4.1-1 (Cont.)

	Channel Description	Check	Test	Calibrate	Remarks
<LATER> (3.3B)	43. ~~ESAS Manual Trip Functions~~				LATER
	a. ~~Switches & Logic~~	~~NA~~	~~R~~	~~NA~~	
	b. ~~Logic~~	~~NA~~	~~M~~	~~NA~~	
SR 3.3.2.1	44. Reactor Manual Trip	NA	P	NA	(A1)
<LATER> (3.4B)	45. ~~Reactor Building Sump Level~~	~~NA~~	~~NA~~	~~R~~	LATER
<LATER> (3.3D)	46. ~~EFW Flow Indication~~	~~M~~	~~NA~~	~~R~~	LATER

3.3.2

(A3)

Table 4.1-1 (Cont.)

Channel Description	Check	Test	Calibrate	Remarks
<LATER> (3.3c) d. SG A High Range Level High-high	S	M	R	LATER
e. SG B High Range Level High-high	S	M	R	
<LATER> (3.3D) 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	
<LATER> (3.4B) 60. Low Temperature Overpressure Protection Alarm Logic	NA	R	R	LATER
<LATER> (3.3D) 61. Core-exit Thermocouples	M	NA	R	LATER
SR 3.3.4.1 — 62 Electronic (SCR) Trip Relays	NA	Q	NA	(A1)
<LATER> (3.3D) 63 RVLMS	M	NA	R	LATER
64 HLLMS	M	NA	R	

<LATER> (3.3B, 3.3C, 3.3D, 3.4B)

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

(A4) + LATER + (R) TRM

3.3.4

NO SIGNIFICANT HAZARDS CONSIDERATIONS STATEMENTS

ITS Section 3.3A: Instrumentation - RPS

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:

<u>3.3A L1</u>

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

This change provides an extension of the Completion Time, for removing power from an inoperable control rod drive (CRD) trip breaker which is failed in the untripped position, from 30 minutes to 1 hour. This 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 does not change (and therefore any initiation scenarios are not changed) and the proposed Completion Time extension is short (and therefore limits the impact on probability). Also, an extension of the Completion Time provides additional opportunity to restore compliance with the requirements and avoid the increased potential for a transient during the shutdown process. Further, the Completion Time for performance of Required Actions does not significantly increase the consequences of an accident because a change in the Completion Time does not change the assumed response of the equipment in performing its specified mitigation functions from that considered during the original Completion Time.

This change will also allow operation to continue indefinitely, with one CRD trip breaker failed in the untripped state, provided power is removed from the breaker. Once power has been removed from the inoperable breaker, the position of the breaker, tripped or untripped, is irrelevant, because the safety function of the breaker, to interrupt power to the CRDs, has been accomplished. Therefore, as long as the Required Action (removing power from the untripped inoperable CRD trip breaker) is accomplished, this change does not involve any increase in the probability or consequences of any accident previously evaluated.

The final portion of this change is to remove the requirement to test each of the other CRD trip breakers, in the event one of them is determined to be inoperable, in the untripped state. This change removes an unnecessary, additional performance of a surveillance which has been performed within its normally required Frequency. Not performing the surveillance would not affect any equipment which is assumed to be an initiator of any analyzed event. Further, since the surveillance continues to be performed on its normal Frequency, there is no impact on the capability of the system to perform its required safety function. Therefore, the proposed change does not involve a significant increase in the probability or consequences of any accident previously evaluated.

<u>3.3A L1</u> (continued)

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. Additionally, the proposed change will still ensure proper surveillances are required for all equipment 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?

Prompt and appropriate Required Actions have been determined based on the safety analysis functions to be maintained. The proposed Completion Time has been determined appropriate based on a combination of the time required to perform the action, the relative importance of the function or parameter to be restored, and engineering judgment. Therefore, the short extension of the Completion Time interval does not involve a significant reduction in the margin of safety.

The ability to promptly initiate the insertion of the CONTROL RODS continues to be provided, while one CRD trip breaker is in the untripped state with its power removed. Therefore, this change does not involve a significant reduction in a margin to safety.

The normal surveillance Frequency, for CRD trip breaker testing, has been shown, based on operating experience, to be adequate for assuring the equipment is available and capable of performing its intended function. Further, any indication of a common mode failure results (through administrative controls) in timely evaluation of the condition and appropriate determination of the need for additional testing. Additionally, the requirements of SR 3.0.4 (CTS 4.0.4) provide assurance the equipment is OPERABLE prior to beginning the functions for which it is required. Therefore, the proposed change does not involve a significant reduction in the margin of safety.

NO SIGNIFICANT HAZARDS CONSIDERATIONS STATEMENTS

<u>3.3A L2</u>

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

This change replaces the twice per week Frequency, for performance of a heat balance calibration of the power range instruments, with a 96 hour Frequency. 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. The power range instrument channels are used to support mitigation of the consequences of an accident; however, operating experience has shown 96 hours is sufficient to detect deviations between heat balance power and indicated reactor power. Further, this information is readily available to the operator, i.e., heat balance power indication and power range neutron flux indicators, to identify abnormalities. 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 all equipment 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

3.3A L3

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

The Applicability for the Loss of Main Feedwater Pumps reactor trip Function and the Main Turbine Trip reactor trip Function has been changed from 5% reactor power to ≥10% RTP and ≥45% RTP respectively. Similarly, the Required Actions have been revised to require only that the MODE of Applicability be exited. This change in Applicability and Required Actions for these functions does not result in any hardware changes. This change 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). Also, the changes do not change the assumed response of the equipment in performing its specified mitigation functions from that considered during the original Applicability since these trip functions were allowed by CTS to be bypassed during the Conditions which will be omitted from the revised Applicability. Therefore, the changes do not significantly increase the consequences of an accident.

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

The proposed change does not 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 availability for the required anticipatory trip functions. 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 Loss of Main Feedwater Pumps reactor trip Function and the Main Turbine Trip reactor trip Function provide anticipatory trips under certain operating conditions. In the conditions to be excluded from the Applicability, the trip functions are bypassed and provide no input to the safety analysis. Therefore, the changes do not involve a significant reduction in the margin of safety.

3.3A L4 - Not used.

3.3 A L5

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

This change allows indefinite continued operation with two inoperable power range channels provided one of the inoperable channels is placed in a tripped condition. This change does not result in any hardware changes. This change also does not significantly increase the probability of occurrence for initiation of any analyzed event since the function of the RPS does not change (and therefore any initiation scenarios are not changed). Since the CTS allows indefinite continued operation in this MODE, i.e., not single failure proof for inducing a spurious trip, for all other functions, any increase in the probability of a spurious trip is not considered significant. Also, the change does not change the assumed response of the equipment in performing its specified mitigation functions from that considered during the original allowed four hour time period. Therefore, the changes do not significantly increase the consequences of an accident.

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

The proposed change does not 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 availability for the required trip functions. 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 to the RPS requirements does not involve a change in setpoints and cannot affect any margin of safety associated with the response to a design basis accident. The RPS is currently allowed to operate with any or all other functions in conditions which are not single failure proof to prevent a spurious trip. Therefore, this change to allow the power range trip function to operate under the same conditions as the rest of the RPS functions, is not considered to involve a significant reduction in the margin of safety.

3.3A L6 - Not used.

3.3A L7

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

This change provides Required Actions which allow OPERABLE equipment to remain in service to be available to perform its safety function rather than remove it from service due to inoperability of another portion of the channel. This change does not result in any hardware changes. This change also does not significantly increase the probability of occurrence for initiation of any analyzed event since the function of the RPS and CRD trip devices does not change (and therefore any initiation scenarios are not changed), and appropriate response of the RPS and CRD trip devices continues to be provided by the alternative Required Actions. Also, the change does not change the assumed response of the equipment in performing its specified mitigation functions from that considered in the safety analysis. Therefore, the changes do not significantly increase the consequences of an accident.

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

The proposed change does not 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 availability for the required trip devices. 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?**

Prompt and appropriate Required Actions have been determined based on the safety analysis functions to be maintained. The continued availability of OPERABLE trip devices is enhanced by the proposed change. Therefore, the change does not involve a significant reduction in the margin of safety.

NO SIGNIFICANT HAZARDS CONSIDERATIONS STATEMENTS

3.3A L8

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

This change provides Required Actions which allow alternative, equivalent compensatory activities for inoperable equipment to maintain the overall availability of the RPS safety function. This change does not result in any hardware changes. This change also does not significantly increase the probability of occurrence for initiation of any analyzed event since the function of the RPS does not change (and therefore any initiation scenarios are not changed), and appropriate response of the RPS continues to be provided by the alternative Required Actions. Also, the change does not change the assumed response of the equipment in performing its specified mitigation functions from that considered in the safety analysis. Therefore, the changes do not significantly increase the consequences of an accident.

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

The proposed change does not 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 availability for the required trip devices. 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?

Prompt and appropriate Required Actions have been determined based on the safety analysis functions to be maintained. The continued availability of the RPS trip devices is maintained by the proposed change. Therefore, the change does not involve a significant reduction in the margin of safety.

NO SIGNIFICANT HAZARDS CONSIDERATIONS STATEMENTS

3.3A L9

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

This change does not result in any hardware changes or changes in operating methods. The change removes an unnecessary additional performance of a surveillance which has been performed within its normal monthly Frequency. Not performing the surveillance at the startup would not affect any equipment which is assumed to be an initiator of any analyzed event. Further, since the surveillance continues to be performed on its normal Frequency, there is no impact on the capability of the system to perform its required safety function. Therefore, the proposed 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 adequate surveillance is performed to identify any degradation of the power range instrumentation channel. 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 a power range instrument channel is based on availability and capability of the instrument to perform its safety function. Since the monthly Frequency has been determined (through experience with CTS) to be adequate to confirm the availability and capability, the removal of an additional confirmatory check of the instrumentation does not impact that availability and capability. Therefore, this change does not involve a significant reduction in a margin of safety.

3.3A L10

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

The Applicability for the source range and intermediate range instrument channels of RPS is limited such that they are not required above MODE 2. Similarly, the Required Actions have been revised such that no actions are required if these channels are inoperable in MODE 1. This change in Applicability and Required Actions for these functions does not result in any hardware changes. This change 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). Also, the changes do not change the assumed response of the equipment in performing its specified mitigation functions from that considered in the safety analysis since the power range monitors provide on-scale indication prior to entering MODE 1. Therefore, the changes do not significantly increase the consequences of an accident.

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

The proposed change does not 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 appropriate availability for the instrument channels 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 instrument channels provide neutron power indication and control rod withdrawal inhibit interlocks (based on high startup rate), under low power operating conditions. In the conditions to be excluded from the Applicability, indication of neutron power is provided by the power range instrumentation channels. Additionally, the control rod withdrawal inhibit interlock functions are not credited by the safety analysis. Therefore, the changes do not involve a significant reduction in the margin of safety.

3.3A L11 - Not used.

3.3A L12

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

A Note is included which allows deferring the calorimetric heat balance for adjustment of the power range instrument channels of the RPS while at low power levels. This Note recognizes the limitations of a calorimetric at low power levels. This change in applicability for this Surveillance does not result in any hardware changes. The power range monitors are not considered as initiators for any previously analyzed accidents. As such, the change does not significantly increase the probability of occurrence of any analyzed event. Performance of this Surveillance at low power levels generally provides less accurate results than at higher power levels and therefore, may not enhance the response of the power range instrumentation channels. This same result, i.e., uncertain calibration, may be obtained without performing the surveillance. Since the results of the Surveillance are typically small adjustments, the change which allows nonperformance during low power does not significantly increase the consequences of an accident.

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

The proposed change does not 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 appropriate availability for the instrument channels 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 power range instrument channels provide no identifiable margin of safety since the difficulties of calibration to the calorimetric heat balance at low power are well known. In the conditions to be excluded from the Surveillance, the power range instrumentation is available, but calibration is recognized as generally less reliable than calibration at higher power levels. Therefore, the change does not involve a significant reduction in the margin of safety.

3.3A L13

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

A Note is included which allows deferring the comparison and adjustment of the power range instrument channels of the RPS against the incore detectors while at low power levels. This Note recognizes that such a comparison at these low power levels provides inaccurate results. This change in applicability for this Surveillance does not result in any hardware changes. The power range monitors are not considered as initiators for any previously analyzed accidents. As such, the change does not significantly increase the probability of occurrence of any analyzed event. Performance of this Surveillance at low power levels does not provide accurate results and therefore, may not enhance the response of the power range instrumentation channels. This same result, i.e., uncertain calibration, may be obtained without performing the surveillance. Since the results of the Surveillance are typically small adjustments, the change which allows nonperformance during low power does not significantly increase the consequences of an accident.

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

The proposed change does not 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 appropriate availability for the instrument channels 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 power range instrument channels provide no identifiable margin of safety at low power since their calibration to the incore instrumentation does not provide accurate results. In the conditions to be excluded from the Surveillance, the power range instrumentation is available, but calibration is recognized as uncertain. Therefore, the change does not involve a significant reduction in the margin of safety.

3.3A L14

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

This change does not result in any hardware changes or changes in operating methods. The change removes an unnecessary additional performance of a surveillance which has been performed within its normal Frequency. Not performing the surveillance prior to startup would not affect any equipment which is assumed to be an initiator of any analyzed event. Further, since the surveillance continues to be performed on its normal Frequency, there is no impact on the capability of the system to perform its required safety function. Therefore, the proposed 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 adequate surveillance is performed to identify any degradation of the intermediate range instrumentation channel. 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 intermediate range instrument channel is based on availability and capability of the instrument to perform its safety function. Since the normal periodic Frequency has been determined (through experience with CTS) to be adequate to confirm the availability and capability, the removal of an additional confirmatory check of the instrumentation does not impact that availability and capability. Therefore, this change does not involve a significant reduction in a margin of safety.

NO SIGNIFICANT HAZARDS CONSIDERATIONS STATEMENTS

3.3A L15

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

This change does not result in any hardware changes or changes in operating methods. The change replaces a function test which is required to be performed "prior to startup" with a CHANNEL CALIBRATION on an 18 month Frequency. This change does not result in any hardware changes, and the source range instrumentation is not considered as the initiator of any previously analyzed accident. Therefore, the proposed change does not involve a significant increase in the probability of any accident previously evaluated. Additionally, neither the test, nor the test Frequency impact the operation of equipment or its response to any event. Therefore, the proposed 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 adequate surveillance is performed to identify any degradation of the source range instrumentation channel. 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 source range instrument channel is based on availability and capability of the instrument to perform its safety function. If the unit operates with only one "startup" per cycle, the Frequency for these surveillances is the same, but the calibration would be an additional requirement because it includes testing activities in addition to the functional test. Industry performance history of this type of instrumentation has demonstrated reliability of the equipment over an operating cycle. Therefore, a periodic Frequency of 18 months has been determined to be adequate to confirm the availability and capability. Therefore, this change does not involve a significant reduction in a margin of safety.

ITS DISCUSSION OF DIFFERENCES
ITS Section 3.3A: Instrumentation - RPS

Note: The ITS Section 3.3A package addresses the following NUREG-1430 RSTS:

RSTS 3.3.1	RPS Instrumentation
RSTS 3.3.2	RPS Manual Reactor Trip
RSTS 3.3.3	RPS - Reactor Trip Module
RSTS 3.3.4	CRD Trip Devices
RSTS 3.3.9	Source Range Neutron Flux
RSTS 3.3.10	Intermediate Range Neutron Flux

1 NUREG 3.3.1 - The CTS flexibility to allow one Reactor Protection System (RPS) channel, containing inoperable functions, to be maintained in an untripped and unbypassed state is maintained. This flexibility is consistent with CTS 3.5.1.3. As modified, ITS 3.3.1 ACTION A will allow one channel containing inoperable function(s) to be maintained untripped and unbypassed. ITS 3.3.1 ACTION B will, in the event that two channels contain inoperable functions, require one of these channels to be tripped. Two options are available for dealing with the second inoperable channel. Either it will be bypassed or bypass of the two remaining OPERABLE channels will be prevented. This change is consistent with current license basis.

2 NUREG 3.3.1 - Condition C has been revised to specify that this Condition applies when three or more RPS channels are inoperable. This change was made to maintain requirements consistent with CTS Table 3.5.1-1 Column 5 and Note 1 which provide specific requirements for the inoperability of more than 2 channels.

Without this addition, entry into the ACTION requirements of ITS LCO 3.0.3 would be required if three channels of RPS contained inoperable functions. Entry into the Required Actions of ITS 3.3.1 Condition C rather than the ACTION requirements of LCO 3.0.3 is more appropriate because specific Required Actions which result in the unit exiting the unique Applicability for each RPS Function are provided in ITS 3.3.1. These Required Actions consistently result in the unit exiting the specific Applicability within a specific Completion Time. For example ITS LCO 3.0.3 ACTION requirements would not provide specific Completion Times for reducing THERMAL POWER to less than 45% RTP in the event three channel of the Main Turbine Trip function were inoperable.

Additionally, several of the RPS functions are required to be OPERABLE while in MODE 5. ITS LCO 3.0.3 is not applicable in MODE 5 and therefore would not require the unit to exit the MODES of Applicability for those Functions required OPERABLE in MODE 5. This change is consistent with TSTF-217, Rev 1.

3 NUREG 3.3.1 - Response time testing of the Reactor Protection System (RPS), i.e., NUREG SR 3.3.1.7, is not adopted in ITS. Testing of this type is not required by ANO-1 CTS. Deletion of these Surveillance Requirements maintains consistency with the current ANO-1 administrative control of these activities and neither removes any current requirement nor adds any additional requirement. This change is consistent with current license basis.

4 NUREG 3.3.1 - The Applicable MODES for Nuclear Overpower High Setpoint function and RCS High Pressure function have been expanded to include MODE 3 when not in shutdown bypass operation with any CRD trip breaker in the closed position and the CRD System capable of rod withdrawal. This change provides for requirements which are more restrictive than those provided by ANO-1 CTS. This additional Applicability is appropriate to ensure that the instrumentation required to initiate the insertion of any withdrawn CONTROL RODS is OPERABLE whenever CONTROL RODS are withdrawn or capable of withdrawal. The automatic insertion of any withdrawn CONTROL ROD is consistent with evaluations of abnormalities initiated from MODE 3 (although no specific analyses have been performed for MODE 3 events.)

This change in Applicability also requires the addition of Note (d) to NUREG-1430 Table 3.3.1-1 and appropriate changes to the Bases. This change is consistent with TSTF-218, as revised to reflect ANO-1 plant specific differences.

5 NUREG 3.3.1 - The Frequency of SR 3.3.1.2 has been changed from 24 hours to 96 hours and once within 24 hours after a THERMAL POWER change of ≥ 10% RTP in one direction. This Frequency replaces the CTS requirement to perform this calibration twice weekly under steady state conditions and daily under non-steady state operating conditions. The change from twice weekly to 96 hours has been made to provide for calibration at an interval essentially equivalent to CTS requirements while presenting the Frequency in a format consistent with that of the ITS. The requirement to perform this calibration daily under non-steady state conditions has been clarified with the site specific details of what constitutes non-steady state conditions. These changes provide requirements which maintain calibration on a Frequency similar to current requirements while presenting these requirements in a format consistent with NUREG-1430. Further, the adoption of 96 hour Frequency is acceptable based on operating experience which indicates that the deviation between the power range instruments and the calorimetric seldom exceeds 2% RTP even after a four day period with no adjustment to the power range instruments.

6 NUREG 3.3.1 - The specific details of performance of ITS SR 3.3.1.3 have been removed. These details provided methodology and acceptance criteria not contained in CTS. The removal of these details maintains requirements consistent with CTS. The details of this testing are currently contained in implementing procedures and will be retained there. This change neither adds any new requirements nor removes any existing requirement. This change is consistent with current license basis.

In addition, the requirement to perform a CHANNEL CALIBRATION if the absolute value of the imbalance error is ≥ [2]% RTP is not included since the Frequency for ITS 3.3.1.5 (NUREG SR 3.3.1.5) is the same as SR 3.3.1.3 and SR 3.3.1.5 also requires the power range CHANNEL CALIBRATION.

ITS DISCUSSION OF DIFFERENCES

7 NUREG 3.3.1 - Reactor Protection System (RPS) CHANNEL FUNCTIONAL TEST requirements are contained in CTS Table 4.1-1. The Frequency of this testing is specified as monthly with the Bases providing implementation details for performing this testing on a rotational or staggered basis. ITS SR 3.3.1.4 is adopted with a 31 day Frequency. The details of performance of this testing on a rotational basis is presented in the Bases of the SR. This change is made to provide requirements consistent with CTS for this testing. No new requirements are added by this change and no existing requirements are removed. This change is consistent with current license basis.

8 NUREG 3.3.4 - ACTION C has been revised and presented in INSERT 3.3-11A. These changes were made to provide requirements which are appropriate for all Control Rod Drive (CRD) power supplies which contain Electronic Trip Assembly (ETA) Relays, including the auxiliary power supply, and to maintain flexibility consistent with CTS requirements.

Required Action C.1 was revised to allow transfer of a CONTROL ROD group to a power supply with an inoperable ETA relay which is open. This change was made to allow a CONTROL ROD group to be transferred to a power supply, and powered indefinitely from it, even if that power supply has one inoperable, open ETA relay. As presented in NUREG 3.3.4 Required Action C.1 would not have allowed the affected CONTROL ROD group to be transferred to a power supply with an inoperable, though open ETA. This change was made to retain the CTS allowance to operate indefinitely with a CONTROL ROD group being powered from a power supply with an inoperable, but open ETA relay.

ITS Required Action C.2 was added to allow a safety rod group which is being powered from the auxiliary power supply to be transferred back to its normal DC hold power supply in the event one of the ETA relays associated with the auxiliary power supply is inoperable. By design, the safety rod groups' normal power supplies do not contain ETA relays. The requirement in NUREG 3.3.4 Required Action C.1 to transfer the affected CONTROL ROD group to a power supply with OPERABLE ETA relays would not have been fulfilled by transferring an affected safety rod group back to its fully OPERABLE normal power supply. This change is being made to ensure that the inoperability of an ETA relay associated with the auxiliary power supply does not result in unnecessarily restrictive action requirements.

ITS Required Action C.3 was added to provide the flexibility to place the SCRs associated with an inoperable ETA relay in trip rather than the associated AC CRD trip breaker. Placing the SCRs associated with the inoperable ETA relay in trip accomplishes the design function of the ETA relay, which is to interrupt power, while allowing the associated AC CRD trip breaker to remain closed. This change is consistent with the requirements of CTS Table 3.5.1-1, Note 23.

The word "Trip" in NUREG 3.3.4 Required Action C.2 was changed to "Open" in ITS 3.3.4 Required Action C.4. This change is considered editorial in nature and is made to prevent possible confusion about the acceptable methods of accomplishing this requirement.

ITS DISCUSSION OF DIFFERENCES

9 NUREG 3.3.9 - ITS LCO 3.3.9 presents the OPERABILITY requirements for the source range instruments. This LCO was modified to indicate that only one channel of instrumentation is required to be OPERABLE. This change was made to provide requirements consistent with those in CTS Table 3.5.1-1, RPS Functional Unit 4. Additionally, NUREG 3.3.9 ACTION A, which dealt with the inoperability of one of the two required instrument channels, was deleted to provide ACTIONS which are consistent with the requirements of the LCO. This change is consistent with current license basis.

10 NUREG 3.3.10 - ITS LCO 3.3.10 presents the OPERABILITY requirements for the intermediate range instruments. This LCO was modified to indicate that only one channel of instrumentation is required to be OPERABLE. This change was made to provide requirements consistent with those in CTS Table 3.5.1-1 RPS Functional Unit 3. Additionally, NUREG 3.3.10 ACTION A, which dealt with the inoperability of one of the two required instrument channels, was deleted to provide ACTIONS which are consistent with the requirements of the LCO. This change is consistent with current license basis.

11 NUREG 3.3.9 - The Note modifying NUREG LCO 3.3.9 was not retained in the ITS. The current source range instrument channels at ANO-1 use a fission chamber detector. This design does not require that the high voltage be removed from these instruments to protect them from operation above 1.0 E -10 amp indicated on the intermediate range channels. This change is being made to provide requirements consistent with unit design features. This change is consistent with current license basis.

12. NUREG 3.3.9 - Conditions B and C were revised to remove the term "THERMAL POWER level" from each. The difference between Condition B and Condition C is whether the intermediate range neutron flux instrumentation channel indicates greater than or less than 1.0 E-10 amp. These instruments provide a relative indication of neutron flux and are not calibrated against heat balance power. The correlation between THERMAL POWER level and intermediate range neutron flux indication, especially when low in the intermediate range, is not easily determined. Intermediate range instrument channel indications in the range of 1.0 E-10 amp are generally indicative of reactor power levels below "the point of adding heat" and therefore not detectable as THERMAL POWER. By specifying that the requirements in Condition B and C are based on THERMAL POWER level rather than indicated neutron flux, the requirements of Conditions B and C were unnecessarily confusing.

13 NUREG 3.3.1 - The Allowable Value for Table 3.3.1-1, Function 5, is revised to retain the CTS as recently approved for Amendment 186. The Function 8 wording is similarly revised for consistency. This change is consistent with current license basis.

14 NUREG 3.3.10 - The CTS requirement to perform a CHANNEL FUNCTIONAL TEST on the intermediate range instrument channels has been retained in ITS. CTS Table 4.1-1, Item 5, requires this testing monthly ITS SR 3.3.10.2 has been adopted to provide a requirement to perform this testing with a Frequency of 31 days. This requirement to perform a CHANNEL FUNCTIONAL TEST on the intermediate range instrument channels is retained in ITS to maintain testing requirements consistent with the ANO-1 CTS.

15 NUREG 3.3.9 - SR 3.3.9.3 has been moved to ITS 3.3.10 as SR 3.3.10.4. This SR provides verification that at least one decade of overlap exists between the source range and intermediate range instruments when the intermediate range instruments come on scale. By associating this SR with the LCO for the source range instruments, the NUREG establishes the successful performance of this SR as an OPERABILITY requirement for the source range instruments. By associating this SR with LCO 3.3.10, this requirement is established as an OPERABILITY requirement for the intermediate range instrument channels.

The requirement to verify one decade overlap between the source range and intermediate range instrument channels ensures a continuous source of power indication is maintained during the approach to criticality. Provided the source range instruments are maintained on scale, a continuous indication of power is maintained, even if the intermediate range instruments fail to come on scale with the required one decade overlap. This is supported by CTS 3.5.1.5. By associating this SR with the intermediate range instrument channels rather than the source range channels, successful performance of this SR will ensure that the intermediate range instrument channels are OPERABLE prior to relying upon them as the primary indication of core reactor power.

As presented in ITS, the Frequency of SR 3.3.10.4 is "Once each reactor startup prior to source range counts exceeding 10^5 cps." This Frequency was adopted over the NUREG Frequency of "Once each reactor startup prior to the source range counts exceeding 10^5 cps if not performed within the previous 7 days. The allowance to not perform this SR, if performed within the previous 7 days, is not consistent with CTS requirements and therefore is not retained.

16 NUREG 3.3.10 - SR 3.3.10.3 was not retained in the ITS because no similar requirement to perform this verification exists in the ANO-1 CTS. According to the BASES for SR 3.3.10.3, this SR is designed to ensure "a continuous source of power indication during the approach to criticality." The design of the ANO-1 intermediate range instruments is such that they provide indication from 1.0 E -11 to 1.0 E -3 amps. The eight decades of indication provided by the intermediate range instrument channels, in conjunction with the six decades of indication provided by the source range instrument channels, provides indication throughout the approach to criticality with no reliance upon the power range instruments for this function.

ITS DISCUSSION OF DIFFERENCES

17 NUREG 3.3.10 - The Applicability has been changed to specify that the intermediate range instrument channel is required in MODE 2 and in MODES 3, 4, and 5 with any CRD trip breaker in the closed position and the CRD System capable of rod withdrawal. The addition of "MODES 3, 4, and 5" to the second statement of the Applicability was made to maintain the CTS allowance provided by Table 3.5.1-1 Note 2. This Note, as applied at ANO-1, defined the upper limit of the applicable MODES for the required intermediate range instrument channel as being 10% indicated neutron power. Without the addition of the appropriate MODES, to the second statement of the Applicability for ITS 3.3.10, an intermediate range channel would be required at all times in MODE 1. This requirement is inconsistent with the design of the intermediate range instrument channels which is to provide indication of neutron power while operating at low power levels (MODE 2). The required indication of neutron power level is provided by the power range instruments while in MODE 1. This change is consistent with current license basis and generic traveler TSTF-291 .

18 NUREG 3.3.1 - The Allowable Value for the Reactor Coolant Pump to Power trip function in ITS Table 3.3.1-1 has been modified to indicate the ANO-1 unit specific value including the appropriate description of the reactor coolant pump combination. This change has been made to provide an accurate description of this Allowable Value which is consistent with the ANO-1 CTS and design function. This change is consistent with current license basis.

19 NUREG Bases 3.3.1 - Reactor Protection System (RPS) design at ANO-1 provides a separate bistable for the Nuclear Overpower Low Setpoint function and for the Nuclear Overpower High Setpoint function. The Low Setpoint bistable is inserted into the RPS channel trip string, along with the Shutdown Bypass RCS High Pressure bistable, when the channel is placed in shutdown bypass. The Bases of ITS 3.3.1 has been modified to describe this site specific design difference in the ANO-1 RPS. This change is consistent with current license basis.

20 NUREG 3.3.3 - Conditions B and C have been revised to specify that these Conditions also apply when two or more Reactor Protection System (RPS) Reactor Trip Modules (RTMs) are inoperable. This change was made to provide ACTION requirements which specifically remove the unit from the Applicability for this Specification.

Without this addition, entry into the ACTION requirements of ITS LCO 3.0.3 would be required if more than one RTM is inoperable. Entry into the Required Actions of ITS 3.3.3 Condition B or C (depending upon the current MODE), rather than the ACTION requirements of LCO 3.0.3, is more appropriate because specific Required Actions which result in the unit exiting the Applicability for LCO 3.3.3 are provided. These Required Actions result in the unit exiting the specific Applicability by either opening the Control Rod Drive (CRD) trip breakers or removing power from the CRD system within a specific Completion Time. ITS LCO 3.0.3 ACTION requirements would not require opening the CRD trip breakers or removing power from the CRD system, and therefore, would not result in exiting the Applicability of ITS 3.3.3.

ITS DISCUSSION OF DIFFERENCES

Additionally, because the Applicability of ITS 3.3.3 includes operation in MODE 5 with the any CRD trip breaker in the closed position and the CRD System capable of rod withdrawal, this change to Condition C will provide for appropriate ACTION requirements to exit the Applicability where ITS 3.0.3 would not. ITS LCO 3.0.3 is not applicable in MODE 5 and therefore would not require any ACTION while in MODE 5. This change is consistent with TSTF-217, Rev 1, as revised for consistency with the description of Condition B.

21 NUREG 3.3.1 Bases - The Bases have been generally revised to reflect unit specific design, analysis, and operating practices, or to provide descriptions consistent with other unit specific documents and terminology. These changes are consistent with current license basis or are editorial. Some specific revisions are as follows:

-Background-
The discussion of trip signals is revised for clarity. Two channels are required to initiate a reactor trip. However, two trip signals will not generate a reactor trip if they are in the same channel.

Discussion of CRD trip devices was revised to more clearly represent the actual configuration of the ANO-1 Control Rod Drive system power supplies. This clarification has been made to provide unit specific information to better describe the function and design of the electronic trip assembly (ETA) relays and other portions of the CRD system.

Discussion revised to omit unnecessary detail for "complete" testing of "all" RPS trip Functions since there may be minor portions of specific RPS functions which are not fully testable. For example, many surveillances exclude neutron detectors. Such exclusions would not constitute complete testing of all Functions.

Discussion of Main Turbine Automatic Stop Oil Pressure was revised to match ANO-1 unit specific design.

Discussion of an indicator light was added to the Channel Bypass discussion.

Potentially confusing discussion of methodology of determining Allowable Values was removed from Trip Setpoints/Allowable Value discussion.

-Applicable Safety Analyses, LCO, and Applicability-
Discussion revised to omit "at all time the reactor is critical" since this is not consistent with the Applicability of all RPS Functions. This change is consistent with TSTF-292.

Reference to the Main Steam Safety Valves (MSSVs) has been removed from the RCS High Pressure Function discussion. The NUREG Bases indicated that these valves functioned to prevent RCS overpressurization and thereby protect the RCS High Pressure Safety Limit (SL). Chapter 3 and 14 of the ANO-1 SAR do not credit the MSSVs as functioning to prevent exceeding RCS SL.

ITS DISCUSSION OF DIFFERENCES

Functions 2, 3, and 6 are revised to identify that the Allowable Value does not consider harsh environmental conditions because the trip is assumed to occur prior to degraded conditions being reached, not because the associated functions are not required to mitigate accidents that create harsh conditions. This is consistent with the ANO-1 Safety Analyses.

Specific Allowable Values for the reactor coolant pump power monitors were not included in the Bases of Function 7 Reactor Coolant Pump to Power. This change was made to maintain these values under current administrative controls.

The ANO-1 unit specific terminology of "electromatic relief valve (ERV)" has been substituted for the generic term "power operated relief valve (PORV)." This change has been made to maintain the specific terminology which is consistent with other ANO-1 licensing basis documentation.

ANO-1 unit specific design information for the RPS trip on Main Turbine Trip function has been added. The system at ANO-1 provides for four main turbine oil pressure switches each providing input to a single RPS channel though a buffer device.

The statement regarding the utilization of the bypass trips to prevent unit conditions from reaching a point where actuation is necessary is not true for all the Functions listed.

A heading of "General Discussion" was added near the bottom of NUREG page B 3.3-20. This was done to separate the general information, which follows, from the preceding discussion which is specific to the Shutdown Bypass RCS Pressure Function. Additionally, the heading of NUREG page B 3.3-21 was appropriately changed due to this new heading.

Applicability discussion on NUREG page B 3.3-20 and B 3.3-21 was replaced with inserted discussion. This change was made to more clearly express the individual Applicability of the Functions in Table 3.3.1-1.

-Actions-

Discussion of ACTIONS D.1 through G.1 were revised to more accurately express the Conditions. This change has been made to clarify possibly confusing descriptions of these Conditions.

-Surveillance Requirements-

Bases SR 3.3.1.2 has been revised by the addition of unit specific details of the determination of calorimetric (heat balance) power.

ITS DISCUSSION OF DIFFERENCES

22 NUREG 3.3.3 - The Note modifying NUREG SR 3.3.3.1 has been deleted. This Note allowed a delay of up to 8 hours for the entry into the Conditions and Required Actions for the performance of this surveillance. The performance of this SR, at ANO-1, does not render the reactor trip module (RTM) inoperable. During performance of the referenced CHANNEL FUNCTIONAL TEST, the coincidence logic network of the RTM is aligned for normal service and the RTM is capable of receiving trip commands from and issuing trip commands to the other three channels. Without the removal of the Note modifying SR 3.3.3.1, the possibility for confusion by the unit staff with regard to the OPERABILITY of the RTM during testing. Because no allowance, similar to this Note, exists in CTS, its removal from ITS maintains current requirements. Reference to the Note was additionally removed from the Bases of SR 3.3.3.1.

23 NUREG 3.3.9 Bases - The Bases have been generally revised to reflect unit specific design, analysis, and operating practices, or to provide descriptions consistent with other unit specific documents and terminology. Some specific revisions are as follows:

-Background- and -LCO-
Discussion has been revised to indicate ANO-1 unit specific design information. These changes have been made to provide accurate information appropriate to the ANO-1 design.

-Actions-
Discussion associated with ITS Required Actions A.1 through A.4 has been revised to more clearly describe the intent of these Required Actions. The NUREG Bases, as written, implied that these Required Actions would remove the unit from the Applicability of this LCO. These Required Actions do not remove the unit from the Applicability of the LCO, but rather provide actions to limit positive reactivity additions and to detect any changes in SDM. This change is consistent with TSTF-293.

Discussion was also added to provide guidance in the event that no indication of intermediate range flux is available, coincident with a loss of the required source range instrument channel. This discussion indicates that with no indication of intermediate range flux, ITS Required Actions A.1 through A.4 are applicable.

-Surveillance Requirements-
General discussion, in SR 3.3.9.1, was edited to match the specific design of the instrumentation to which this SR is applicable. Discussion of "transmitter...drift" is not appropriate due to the design of the source range instrument channels.

Discussion of performance of CHANNEL CHECKS, for off scale low current loop instrument channels, was removed from SR 3.3.9.1 discussion. This discussion was not appropriate based on the design of the source range instrument channels

24 NUREG 3.3.2 Bases - The Bases have been generally revised to reflect unit specific design, analysis, and operating practices, or to provide descriptions consistent with other unit specific documents and terminology. Some specific revisions are as follows:

-Applicability-
The word "only" was replaced with "primary" when describing the RPS safety function of tripping the CONTROL RODS. The ANO-1 unit specific design of the Reactor Protection System (RPS) is such that it provides inputs to other safety systems. Most notably, the Emergency Feedwater Initiation and Control (EFIC) system receives input from the RPS.

-Actions-
The specific Condition descriptions have been corrected for accuracy.

25 NUREG 3.3.3 Bases - The Bases have been generally revised to reflect unit specific design, analysis, and operating practices, or to provide descriptions consistent with other unit specific documents and terminology. Some specific revisions are as follows:

-Actions-
The specific Condition descriptions have been corrected for accuracy.

26 NUREG 3.3.4 Bases - The Bases have been generally revised to reflect unit specific design, analysis, and operating practices, or to provide descriptions consistent with other unit specific documents and terminology. Some specific revisions are as follows:

-Background -
Discussion of CRD trip devices was revised to more clearly represent the actual configuration of the ANO-1 Control Rod Drive system power supplies. This clarification has been made to provide unit specific information to better describe the function and design of the electronic trip assembly (ETA) relays and other portions of the CRD system.

-LCO-
Discussion of CRD trip devices was revised to more clearly represent the actual configuration of the ANO-1 Control Rod Drive system power supplies. This clarification has been made to provide unit specific information to better describe the function and design of the electronic trip assembly (ETA) relays and other portions of the CRD system.

-Actions-
The headings "Condition A" and "Condition B" were removed and the appropriate Required Action headings were moved to provide a format consistent with other sections.

The phrase "channel containing" was removed to ensure that this was not misinterpreted to mean that power was required to be removed from the RPS channel associated with the inoperable trip breaker.

Discussion previously under "Condition B" was revised to remove possibly confusing descriptions this Condition.

The specific Condition descriptions have been corrected for accuracy.

-Surveillance Requirements-
"AC" was replaced with "trip" in Bases SR 3.3.4.1 to ensure that this statement was not interpreted to mean that testing of the diverse trip features of the DC breakers was not required.

27 NUREG 3.3.10 Bases - The Bases have been generally revised to reflect unit specific design, analysis, and operating practices, or to provide descriptions consistent with other unit specific documents and terminology. Some specific revisions are as follows:

-Background-
Discussion was revised to indicate ANO-1 unit specific design information. These changes have been made to provide accurate information appropriate to the ANO-1 design.

-Surveillance Requirements-
General discussion, in SR 3.3.10.1, was edited to match the specific design of the instrumentation to which this SR is applicable. Discussion of "transmitter...drift" is not appropriate due to the design of the intermediate range instrument channels.

Discussion of performance of CHANNEL CHECKS, for off scale low current loop instrument channels, was added to SR 3.3.10.1 discussion. This discussion was appropriate based on the design of the intermediate range instrument channels

28 NUREG 3.3.9 - Incorporates TSTF-009, Rev. 1.

29 NUREG Bases 3.3.1 - Incorporates TSTF-019, Rev. 1.

30 NUREG SR 3.3.1.5 has been revised to present a requirement to "Calibrate the power range channels to the incore channels" in the ITS. The NUREG SR 3.3.1.5 requirement to perform CHANNEL CALIBRATION is not actually a calibration of the function but is, instead, only a calibration of the inputs to the function (i.e., calibrates the power range channels to the incore channels). SR 3.3.1.5 has been reworded to specifically require the power range channels to be calibrated rather than to incorrectly specify a CHANNEL CALIBRATION. As such, there would be no requirements to demonstrate the OPERABILITY of the entire instrument channel. Therefore, requirements are added in Table 3.3.1-1 to perform a CHANNEL CALIBRATION (3.3.1.6) every 18 months for Functions 1.a and 1.b and to perform a CHANNEL FUNCTIONAL TEST (SR 3.3.1.4) every 31 days for Functions 1.a, 1.b and 8.

A Note has been added to ITS 3.3.1.5 to allow a delay in performance until 24 hours after THERMAL POWER is ≥ 20% RTP since at low power levels calorimetric data are inaccurate and the incore nuclear instruments are not capable of providing reliable, accurate indication of AXIAL POWER IMBALANCE. SR 3.3.1.5 applicability to Function 1.b is deleted since the Nuclear Overpower Low Setpoint is only applicable at power levels < 5% RTP. This change incorporates TSTF-342, revised as discussed above.

31 NUREG 3.3.1, 3.3.4, and associated Bases - The requirement to remove all power to the CRD system could be interpreted to include all control power and logic cabinet power since they are a part of this system. It is more appropriate to remove power from all CRD trip breakers. This action places the unit in condition where the LCO no longer applies. This change incorporates TSTF-211, which has been revised for grammatical correctness in the discussion of Required Actions B.1, B.2.1, and B.2.2. The revision of this generic change is considered to be editorial in nature.

32 NUREG 3.3.1 -NUREG SR 3.3.1.2 and SR 3.3.1.3 are revised such that the Note delays applicability for these SRs until some period of time "after THERMAL POWER is ≥ 20% RTP" rather than "after THERMAL POWER is ≥ 15% RTP." This change is based on current unit specific application of validity of the calorimetric heat balance at low powers. This change is also consistent with the similar "standard" requirements for the other PWR vendors, i.e., NUREG-1431 and NUREG-1432.

33 NUREG 3.3.3 & 3.3.4 - Reactor Protection System (RPS) CHANNEL FUNCTIONAL TEST requirements are contained in CTS Table 4.1-1. The Frequency of this testing is specified as quarterly with the Bases providing implementation details for performing this testing on a rotational or staggered basis. ITS SR 3.3.3.1 and SR 3.3.4.1 are adopted with a 92 day Frequency. The details of performance of this testing on a rotational basis is presented in the Bases of each SR. This change is made to provide requirements consistent with CTS for this testing. No new requirements are added by this change and no existing requirements are removed. This change is consistent with current license basis.

CTS

3.3 INSTRUMENTATION

3.3.1 Reactor Protection System (RPS) Instrumentation

LCO 3.3.1 Four channels of RPS instrumentation for each Function in Table 3.3.1-1 shall be OPERABLE. Table 3.5.1-1

APPLICABILITY: According to Table 3.3.1-1. N/A

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME	
A. One channel inoperable.	A.1 Place channel in bypass or trip. <INSERT 3.3-1A>	1 hour	3.5.1.3 1
B. Two channels inoperable.	B.1 Place one channel in trip.	1 hour	Table 3.5.1-1 Note 6 3.5.1.3
	AND		
Three or more Channels inoperable OR	B.2.1 Place second channel in bypass. <INSERT 3.3-1B>	1 hour	3.5.1.3 1 2
C. Required Action and associated Completion Time of Condition A or B not met.	C.1 Enter the Condition referenced in Table 3.3.1-1 for the Function.	Immediately	Table 3.5.1-1 Column 5
D. As required by Required Action C.1 and referenced in Table 3.3.1-1.	D.1 Be in MODE 3. AND D.2 Open all ~~CONTROL ROD~~ control rod drive (CRD) trip breakers.	6 hours 6 hours	Table 3.5.1-1 Note 1 edit

(continued)

<INSERT 3.3-1A>

			CTS
<u>OR</u>			
A.2	Prevent bypass of remaining channels.	1 hour	3.5.1.3

<INSERT 3.3-1B>

<u>OR</u>			
B.2.2	Prevent bypass of remaining channels.	1 hour	3.5.1.3

 CTS

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME	
E. As required by Required Action C.1 and referenced in Table 3.3.1-1.	E.1 Open all CRD trip breakers.	6 hours	N/A
F. As required by Required Action C.1 and referenced in Table 3.3.1-1.	F.1 Reduce THERMAL POWER < ~~[45]~~ 45 % RTP.	6 hours	3.5.1.9.3
G. As required by Required Action C.1 and referenced in Table 3.3.1-1.	G.1 Reduce THERMAL POWER < ~~[15]~~ 10 % RTP.	6 hours	3.5.1.9.3

SURVEILLANCE REQUIREMENTS

-----NOTE-----
Refer to Table 3.3.1-1 to determine which SRs apply to each RPS Function.
---------- N/A

SURVEILLANCE	FREQUENCY	
SR 3.3.1.1 Perform CHANNEL CHECK.	12 hours	Table 4.1-1 'Check'

(continued)

CTS

SURVEILLANCE REQUIREMENTS (continued)

	SURVEILLANCE	FREQUENCY	
SR 3.3.1.2	------NOTE------ Not required to be performed until ~~[24]~~ hours after THERMAL POWER is ≥ [15%] RTP. 20% ------ Verify calorimetric heat balance is ≤ ~~[2]~~% RTP greater than power range channel output. Adjust power range channel output if calorimetric exceeds power range channel output by ≥ ~~[2]~~% RTP.	5 96 hours AND Once within 24 hours after a THERMAL POWER change of ≥10% RTP	N/A 32 Table 4.1-1 Item 3
SR 3.3.1.3	------NOTE------ Not required to be performed until ~~[24]~~ hours after THERMAL POWER is ≥ [5%] RTP. 20% ------ Compare out of core measured AXIAL POWER IMBALANCE ~~(API_0)~~ to incore measured AXIAL POWER IMBALANCE. ~~(API_I) as follows:~~ ~~$(RTP/TP)(API_0 - API_I)$ = imbalance error~~ ~~Perform CHANNEL CALIBRATION if the absolute value of the imbalance error is ≥ [2]% RTP.~~ 6	31 days	32 N/A EDIT Table 4.1-1 Item 4 "Check"
SR 3.3.1.4	Perform CHANNEL FUNCTIONAL TEST. Not required to be performed until 24 hours after THERMAL POWER is ≥ 20% RTP.	31 days ~~[45] days on a STAGGERED TEST BASIS~~ 7	Table 4.1-1 "Test"
SR 3.3.1.5	------NOTE------ ~~Neutron detectors are excluded from CHANNEL CALIBRATION.~~ ------ ~~Perform CHANNEL CALIBRATION.~~ Calibrate the power range channels to the incore channels.	31 ~~[92]~~ days	N/A 30 Table 4.1-1 Item 4 "Calibrate"

(continued)

CTS

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY	
SR 3.3.1.6	-----NOTE----- Neutron detectors are excluded from CHANNEL CALIBRATION. ----- Perform CHANNEL CALIBRATION.	[18] months	N/A Table 4.1-1 "Calibrate"
~~SR 3.3.1.7~~	~~-----NOTE----- Neutron detectors are excluded from RPS RESPONSE TIME testing. ----- Verify that RPS RESPONSE TIME is within limits.~~	~~[18] months on a STAGGERED TEST BASIS~~	3

Table 3.3.1-1 (page 1 of 1)
Reactor Protection System Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	CONDITIONS REFERENCED FROM REQUIRED ACTION C.1	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE
1. Nuclear Overpower —				
a. High Setpoint	1,2(a) 5(d)	D	SR 3.3.1.1 SR 3.3.1.2 SR 3.3.1.5 ~~SR 3.3.1.7~~ SR 3.3.1.6 SR 3.3.1.4	≤ ~~104.9~~ 104.9% RTP

Table 3.5.1-1 RPS Functional Unit 2
+ Table 2.3-1

N/A

4

Table 3.5.1-1 Note 1

Table 4.1-1 Item 4

Table 4.1-1 Item 3

3

30

30

Table 2.3-1

(a) When not in shutdown bypass operation.

(b) During shutdown bypass operation with any CRD trip breaker~~s~~ in the closed position and the CRD System capable of rod withdrawal. edit

(c) With any CRD trip breaker in the closed position and the CRD System capable of rod withdrawal.

(d) with any CRD trip breaker in the closed position, the CRD system capable of rod withdrawal, and not in shutdown bypass operation.

4

(1)

Table 3.3.1-1 (page 1 of 1)
Reactor Protection System Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	CONDITIONS REFERENCED FROM REQUIRED ACTION C.1	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE
b. Low Setpoint	2(b),3(b) 4(b),5(b)	E	SR 3.3.1.1; SR 3.3.1.~~6~~ ~~SR 3.3.1.7~~ SR 3.3.1.6	≤ 5% RTP
Table 2.3-1	N/A	N/A	N/A	Table 2.3-1

(a) When not in shutdown bypass operation.

(b) During shutdown bypass operation with any CRD trip breaker~~s~~ in the closed position and the CRD System capable of rod withdrawal. edit

(c) With any CRD trip breaker in the closed position and the CRD System capable of rod withdrawal.

(d) with any CRD trip breaker in the closed position, the CRD system capable of rod withdrawal, and not in shutdown bypass operation. 4

(2)

3 30 4

Table 3.3.1-1 (page 1 of 1)
Reactor Protection System Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	CONDITIONS REFERENCED FROM REQUIRED ACTION C.1	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE
2. RCS High Outlet Temperature	1,2	D	SR 3.3.1.1 SR 3.3.1.4 SR 3.3.1.6	≤ 618 °F

Table 3.5.1-1 RPS Functional Units
+ Table 2.3-1

N/A

Table 3.5.1-1 Note 1

Table 4.1-1 Item 7 "Check"

Table 4.1-1 Item 7 "Calibrate"
Table 4.1-1 Item 7 "Test"

Table 2.3-1

(a) When not in shutdown bypass operation.

(b) During shutdown bypass operation with any CRD trip breakers in the closed position and the CRD System capable of rod withdrawal. edit

(c) With any CRD trip breaker in the closed position and the CRD System capable of rod withdrawal.

(d) with any CRD trip breaker in the closed position, the CRD system capable of rod withdrawal, and not in shutdown bypass operation. 4

(3)

Table 3.3.1-1 (page 1 of 1)
Reactor Protection System Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	CONDITIONS REFERENCED FROM REQUIRED ACTION C.1	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE
3. RCS High Pressure	1,2(a), 3(d)	D	SR 3.3.1.1 SR 3.3.1.4 SR 3.3.1.6 SR 3.3.1.7	≤ 2355 psig

Table 3.5.1-1 RPS Functional Unit 8a + Table 2.3-1

N/A

Table 3.5.1-1 Note 1

3

Table 4.1-1 Item 8 "calibrate"
Table 4.1-1 Item 8 "Test"

2355

Table 2.3-1

4

(a) When not in shutdown bypass operation.

(b) During shutdown bypass operation with any CRD trip breakers in the closed position and the CRD System capable of rod withdrawal. edit

(c) With any CRD trip breaker in the closed position and the CRD System capable of rod withdrawal.

(d) with any CRD trip breaker in the closed position, the CRD system capable of rod withdrawal, and not in shutdown bypass operation.

4

(4)

Table 3.3.1-1 (page 1 of 1)
Reactor Protection System Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	CONDITIONS REFERENCED FROM REQUIRED ACTION C.1	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE
4. RCS Low Pressure	1,2(a)	D	SR 3.3.1.1 SR 3.3.1.4 SR 3.3.1.6 ~~SR 3.3.1.7~~ (3)	≥ ~~[1600]~~ 1800 psig
Table 3.5.1-1 RPS Functional Unit 8b + Table 2.3-1	N/A	Table 3.5.1-1 Note 1	Table 4.1-1 Item 9 "check" Table 4.1-1 Item 9 "Calibrate" Table 4.1-1 Item 9 "Test"	Table 2.3-1

(a) When not in shutdown bypass operation.

(b) During shutdown bypass operation with any CRD trip breaker~~s~~ in the closed position and the CRD System capable of rod withdrawal. edit

(c) With any CRD trip breaker in the closed position and the CRD System capable of rod withdrawal.

(d) with any CRD trip breaker in the closed position, the CRD system capable of rod withdrawal, and not in shutdown bypass operation.

(4)

(5)

Table 3.3.1-1 (page 1 of 1)
Reactor Protection System Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	CONDITIONS REFERENCED FROM REQUIRED ACTION C.1	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE
5. RCS Variable Low Pressure	1,2[a]	D	SR 3.3.1.1 SR 3.3.1.4 SR 3.3.1.6	~~≥ (11.59) * T_{out} (5037.8) psig~~
Table 3.5.1-1 RPS Functional Unit 6 + Table 2.3-1	N/A	Table 3.5.1-1 Note 1	Table 4.1-1 Item 11 "check" Table 4.1-1 Item 11 "calibrate" Table 4.1-1 Item 11 "test"	Table 2.3-1

13

As specified in the COLR

(a) When not in shutdown bypass operation.

(b) During shutdown bypass operation with any CRD trip breaker~~s~~ in the closed position and the CRD System capable of rod withdrawal. edit

(c) With any CRD trip breaker in the closed position and the CRD System capable of rod withdrawal.

(d) with any CRD trip breaker in the closed position, the CRD system capable of rod withdrawal, and not in shutdown bypass operation.

4

(6)

Table 3.3.1-1 (page 1 of 1)
Reactor Protection System Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	CONDITIONS REFERENCED FROM REQUIRED ACTION C.1	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE
Table 3.5.1-1 RPS Functional Unit 1D + Table 2.3-1	N/A	Table 3.5.1-1 Note 1	Table 4.1-1 Item 13 "check"; Table 4.1-1 Item 13 "test"; Table 4.1-1 Item 13 "Calibrate"	Table 2.3-1
6. Reactor Building High Pressure	1,2,3[(c)]	D	SR 3.3.1.1 SR 3.3.1.4 SR 3.3.1.6	~~≤ (4) psig~~ ≤ 18.7 psig

(a) When not in shutdown bypass operation.

(b) During shutdown bypass operation with any CRD trip ~~breakers~~ breaker in the closed position and the CRD System capable of rod withdrawal. edit

(c) With any CRD trip breaker in the closed position and the CRD System capable of rod withdrawal.

(d) with any CRD trip breaker in the closed position, the CRD system capable of rod withdrawal, and not in shutdown bypass operation.

4

(7)

Table 3.3.1-1 (page 1 of 1)
Reactor Protection System Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	CONDITIONS REFERENCED FROM REQUIRED ACTION C.1	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE
7. Reactor Coolant Pump to Power	1,2(a)	D	SR 3.3.1.1 SR 3.3.1.4 SR 3.3.1.6 ~~SR 3.3.1.7~~ 3	≤55% RTP with one pump operating in each loop 18

Table 3.5.1-1 RPS Functional Unit 9 & Table 2.3-1

N/A

Table 3.5.1-1 Note 1

Table 4.1-1 Item 12 "check"

Table 4.1-1 Item 12 "test"

Table 4.1-1 Item 12 "calibrate"

Table 2.3-1

(a) When not in shutdown bypass operation.

(b) During shutdown bypass operation with any CRD trip breaker~~s~~ in the closed position and the CRD System capable of rod withdrawal. edit

(c) With any CRD trip breaker in the closed position and the CRD System capable of rod withdrawal.

(d) with any CRD trip breaker in the closed position, the CRD system capable of rod withdrawal, and not in shutdown bypass operation. 4

(8)

Table 3.3.1-1 (page 1 of 1)
Reactor Protection System Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	CONDITIONS REFERENCED FROM REQUIRED ACTION C.1	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE
8. Nuclear Overpower RCS Flow and Measured AXIAL POWER IMBALANCE	1,2[(a)] N/A	D Table 3.5.1-1 Note 1	SR 3.3.1.1 SR 3.3.1.3 SR 3.3.1.5 SR 3.3.1.6 SR 3.3.1.7 SR 3.3.1.4	Nuclear Overpower RCS Flow and AXIAL POWER IMBALANCE setpoint envelope in COLR As specified the

Table 3.5.1-1 RPS Functional Unit 7
+ Table 2.3-1

Table 4.1-1 Item 4 "Calibrate"
Table 4.1-1 Item 4 "check"
Table 4.1-1 Item 10 "check"
Table 4.1-1 Item 10 "Calibrate"
Table 2.3-1

30

3

13

(a) When not in shutdown bypass operation.

(b) During shutdown bypass operation with any CRD trip breakers in the closed position and the CRD System capable of rod withdrawal. edit

(c) With any CRD trip breaker in the closed position and the CRD System capable of rod withdrawal.

(d) with any CRD trip breaker in the closed position, the CRD system capable of rod withdrawal, and not in shutdown bypass operation.

4

(9)

Table 3.3.1-1 (page 1 of 1)
Reactor Protection System Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	CONDITIONS REFERENCED FROM REQUIRED ACTION C.1	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE
9. Main Turbine Trip (Control Oil Pressure) edit	≥ (45)% RTP 45	F	SR 3.3.1.1 SR 3.3.1.4 SR 3.3.1.6	≥ (45) psig 40.5
Table 3.5.1-1 RPS Functional Unit 12	3.5.1.9.2 & Table 3.5.1-1 Note 10	3.5.1.9.3	Table 4.1-1 Item 41 Check Table 4.1-1 Item 41 Test Table 4.1-1 Item 41 Calibrate	N/A

(a) When not in shutdown bypass operation.

(b) During shutdown bypass operation with any CRD trip breakers in the closed position and the CRD System capable of rod withdrawal.

(c) With any CRD trip breaker in the closed position and the CRD System capable of rod withdrawal.

(d) with any CRD trip breaker in the closed position, the CRD system capable of rod withdrawal, and not in shutdown bypass operation.

4

(12)

Table 3.3.1-1 (page 1 of 1)
Reactor Protection System Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	CONDITIONS REFERENCED FROM REQUIRED ACTION C.1	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE
Table 3.5.1-1 RPS Functional Unit 11	3.5.1.9.1 & Table 3.5.1-1 Note 15	3.5.1.9.3	Table 3.5.1-1 Item 35 "check" Table 3.5.1-1 Item 35 "test" Table 3.5.1-1 Item 35 "calibrate"	N/A
10. Loss of Main Feedwater Pumps (Control Oil Pressure)	≥ (15)% RTP 10	G	SR 3.3.1.1 SR 3.3.1.4 SR 3.3.1.6	≥ (55) psig 55.5

(a) When not in shutdown bypass operation.

(b) During shutdown bypass operation with any CRD trip ~~breakers~~ breaker in the closed position and the CRD System capable of rod withdrawal. edit

(c) With any CRD trip breaker in the closed position and the CRD System capable of rod withdrawal.

(d) with any CRD trip breaker in the closed position, the CRD system capable of rod withdrawal, and not in shutdown bypass operation. 4

(11)

Table 3.3.1-1 (page 1 of 1)
Reactor Protection System Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	CONDITIONS REFERENCED FROM REQUIRED ACTION C.1	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE
Table 2.3-1 11. Shutdown Bypass RCS High Pressure	N/A $2^{(b)}, 3^{(b)}$ $4^{(b)}, 5^{(b)}$	N/A E	N/A SR 3.3.1.1 SR 3.3.1.4 SR 3.3.1.6	≤ ~~1720~~ 1720 psig Table 2.3-1

(a) When not in shutdown bypass operation.

(b) During shutdown bypass operation with any CRD trip breakers in the closed position and the CRD System capable of rod withdrawal.

(c) With any CRD trip breaker in the closed position and the CRD System capable of rod withdrawal.

(d) with any CRD trip breaker in the closed position, the CRD system capable of rod withdrawal, and not in shutdown bypass operation.

4

12

CTS

3.3 INSTRUMENTATION

3.3.2 Reactor Protection System (RPS) Manual Reactor Trip

LCO 3.3.2 The RPS Manual Reactor Trip Function shall be OPERABLE. Table 3.5.1-1 Item 1

APPLICABILITY: MODES 1 and 2,
MODES 3, 4, and 5 with any CONTROL ROD control rod drive (CRD) trip breaker in the closed position and the CRD System capable of rod withdrawal. edit N/A

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME	
A. Manual Reactor Trip Function inoperable.	A.1 Restore Function to OPERABLE status.	1 hour	N/A
B. Required Action and associated Completion Time not met in MODE 1, 2, or 3.	B.1 Be in MODE 3. AND B.2 Open all CRD trip breakers.	6 hours 6 hours	Table 3.5.1-1 Note 1
C. Required Action and associated Completion Time not met in MODE 4 or 5.	C.1 Open all CRD trip breakers.	6 hours	N/A

CTS

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
SR 3.3.2.1 Perform CHANNEL FUNCTIONAL TEST.	Once prior to each reactor startup if not performed within the previous 7 days

Table 4.1-1 Item 44 "Test"

CTS

3.3 INSTRUMENTATION

3.3.3 Reactor Protection System (RPS)—Reactor Trip Module (RTM)

N/A

LCO 3.3.3 Four RTMs shall be OPERABLE.

APPLICABILITY: MODES 1 and 2,
MODES 3, 4, and 5 with any CONTROL ROD control rod drive (CRD) trip breaker in the closed position and the CRD System capable of rod withdrawal.

edit N/A

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME		
A. One RTM inoperable.	A.1.1 ~~Trip~~ Open the associated CRD trip breaker.	1 hour	EDIT	N/A
	OR			
	A.1.2 Remove power from the associated CRD trip breaker.	1 hour		N/A
	AND			
Two or more RTMs inoperable in MODE 1, 2, or 3. OR	A.2 Physically remove the inoperable RTM.	1 hour		N/A
B. Required Action and associated Completion Time not met in MODE 1, 2, or 3.	B.1 Be in MODE 3.	6 hours	20	N/A
	AND			
	B.2.1 Open all CRD trip breakers.	6 hours		N/A
	OR			
	B.2.2 Remove ~~all~~ power ~~to~~ from all ~~the~~ CRD ~~System.~~ trip breakers.	6 hours	31	N/A

(continued)

CTS

Two or more RTMs inoperable in MODE 4 or 5.
OR

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
C. Required Action and associated Completion Time not met in MODE 4 or 5.	C.1 Open all CRD trip breakers. OR C.2 Remove ~~all~~ power ~~to~~ from all ~~the~~ CRD ~~system.~~ trip breakers.	6 hours 6 hours

20 N/A

31 N/A

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
SR 3.3.3.1 ~~-----NOTE----- When an RTM is placed in an inoperable status solely for performance of this Surveillance, entry into associated Conditions and Required Actions may be delayed for up to 8 hours, provided at least two RTM channels are OPERABLE.~~ Perform CHANNEL FUNCTIONAL TEST.	92 days ~~[45] days on a STAGGERED TEST BASIS~~

22

33 Table 4.1-1 Item 1 "Test"

CTS

3.3 INSTRUMENTATION

3.3.4 ~~CONTROL ROD~~ Drive (CRD) Trip Devices

edit

LCO 3.3.4 The following CRD trip devices shall be OPERABLE: Table 3.5.1-1

a. Two AC CRD trip breakers; item 14A

b. Two DC CRD trip breaker pairs; and item 14B

c. Eight electronic trip assembly (ETA) relays. item 13

APPLICABILITY: MODES 1 and 2,
MODES 3, 4, and 5 with ~~when~~ any CRD trip breaker ~~is~~ in the closed position and the CRD System ~~is~~ capable of rod withdrawal.

edit
N/A

ACTIONS

NOTE
Separate Condition entry is allowed for each CRD trip device.

N/A

CONDITION		REQUIRED ACTION	COMPLETION TIME	
A. One or more CRD trip breaker(s) ~~(~~or breaker pair~~)~~ undervoltage or shunt trip Functions inoperable.	A.1	Open ~~Trip~~ the CRD trip breaker.	48 hours	edit Table 3.5.1-1 Note 25
	OR			
	A.2	Remove power from the CRD trip breaker.	48 hours	
B. One or more CRD trip breaker(s) ~~(~~or breaker pair~~)~~ inoperable for reasons other than those in Condition A.	B.1	Open ~~Trip~~ the CRD trip breaker.	1 hour	edit Table 3.5.1-1 Note 24, a.1
	OR			
	B.2	Remove power from the CRD trip breaker.	1 hour	Table 3.5.1-1 Note 24, a.2 3.5.1.6

(continued)

CTS

ACTIONS (continued)

CONDITION	REQUIRED ACTION		COMPLETION TIME
C. One or more required ETA relays inoperable. <INSERT 3.3-11A>	C.1	Transfer affected CONTROL ROD group to power supply with OPERABLE or open ETA relays.	1 hour
	OR		
	C.2 (4)	Open ~~Trip~~ corresponding CRD trip breaker.	1 hour
D. Required Action and associated Completion Time not met in MODE 1, 2, or 3.	D.1	Be in MODE 3.	6 hours
	AND		
	D.2.1	Open all CRD trip breakers.	6 hours
	OR		
	D.2.2	Remove ~~all~~ power from ~~to~~ all ~~the~~ CRD ~~system.~~ trip breakers.	6 hours
E. Required Action and associated Completion Time not met in MODE 4 or 5.	E.1	Open all CRD trip breakers.	6 hours
	OR		
	E.2	Remove ~~all~~ power from ~~to~~ all ~~the~~ CRD ~~system.~~ trip breakers.	6 hours

EDIT N/A

(8)

N/A

N/A

(31)

N/A

(31)

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
SR 3.3.4.1 Perform CHANNEL FUNCTIONAL TEST.	92 ~~31~~ days

(33) Table 4.1-1 ITEMS 2 "Test" 62 "Test"

<INSERT 3.3-11A>

			CTS
OR			
C.2	Transfer affected CONTROL ROD group to a DC hold power supply.	1 hour	N/A
OR			
C.3	Place the SCRs associated with the inoperable ETA relay(s) in trip.	1 hour	Table 3.5.1-1 Note 23

CTS

3.3 INSTRUMENTATION

3.3.9 Source Range Neutron Flux

9 Table 3.5.1-1 Functional Unit 4

LCO 3.3.9 One ~~Two~~ source range neutron flux channel~~s~~ shall be OPERABLE.

~~-----NOTE-----~~
~~High voltage to detector may be de-energized above 1E-10 amp on intermediate range channels.~~ 11

Table 3.5.1-1 Note 2

APPLICABILITY: MODES 2, 3, 4, and 5. N/A

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
~~A. One source range neutron flux channel inoperable with THERMAL POWER level ≤ 1E-10 amp on the intermediate range neutron flux channels.~~	~~A.1 Restore channel to OPERABLE status.~~	~~Prior to increasing THERMAL POWER~~ 9
A. ~~B.~~ Required ~~Two~~ source range neutron flux channel~~s~~ inoperable ~~with THERMAL POWER level~~ ≤ 1E-10 amp on the intermediate range neutron flux channel~~s~~. 12	A ~~B~~.1 Suspend operations involving positive reactivity changes.	Immediately N/A
	AND	
	A ~~B~~.2 Initiate action to insert all CONTROL RODS.	Immediately
	AND	
	A ~~B~~.3 Open CONTROL ROD drive trip breakers.	1 hour edit
	AND	
		(continued)

CTS

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A ~~B~~. (continued)	A ~~B~~.4 Verify SDM 25 ~~≥ 1% Δk/k~~ to be within the limit provided in the COLR.	1 hour **AND** Once per 12 hours thereafter
B ~~C~~. Required ~~One or more~~ source range neutron flux channel~~(s)~~ inoperable with ~~THERMAL POWER level~~ > 1E-10 amp on the intermediate range neutron flux channel~~(s)~~.	B ~~C~~.1 Initiate action to restore ~~affected~~ channel~~(s)~~ to OPERABLE status.	1 hour required

28 N/A

Table 3.5.1-1 Note 3 9

12

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.3.9.1	Perform CHANNEL CHECK.	12 hours
SR 3.3.9.2	-------------------NOTE------------------- Neutron detectors are excluded from CHANNEL CALIBRATION. -- Perform CHANNEL CALIBRATION.	~~18~~ months

Table 4.1-1 Item 6 "Check"

N/A

N/A

(continued)

CTS

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
~~SR 3.3.9.3~~	~~Verify at least one decade overlap with intermediate range neutron flux channels.~~	~~Once each reactor startup prior to source range counts exceeding 10^5 cps if not performed within the previous 7 days~~

15

CTS

3.3 INSTRUMENTATION

3.3.10 Intermediate Range Neutron Flux

LCO 3.3.10 ~~Two~~ One intermediate range neutron flux channel~~s~~ shall be OPERABLE.

10 Table 3.5.1-1 RPS Functional Unit 3.

APPLICABILITY: MODE 2, ~~When~~ with any ~~CONTROL ROD~~ control rod drive (CRD) trip breaker ~~is~~ in the closed position and the CRD System ~~is~~ capable of rod withdrawal.

MODES 3, 4, and 5

17

Table 3.5.1-1 Note 2 — N/A

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
~~A. One channel inoperable.~~	~~A.1~~	~~Reduce THERMAL POWER to ≤ 1E-10 amp.~~	~~2 hours~~
~~B.~~ A. ~~Two~~ Required channel~~s~~ inoperable.	~~B~~ A.1	Suspend operations involving positive reactivity changes.	Immediately
	AND ~~B~~ A.2	Open CRD trip breakers.	1 hour

10

3.5.15

N/A

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
SR 3.3.10.1 Perform CHANNEL CHECK.	12 hours

Table 4.1-1 Item 5 "Check"

(continued)

<INSERT 3.3-25A> 14

<INSERT 3.3-25A>

			CTS
SR 3.3.10.2	Perform CHANNEL FUNCTIONAL TEST.	31 days	Table 4.1-1 Item 5 "Test"

CTS

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY	
SR 3.3.10.~~2~~ 3	-------NOTE------- Neutron detectors are excluded from CHANNEL CALIBRATION. ------------------ Perform CHANNEL CALIBRATION.	~~18~~ months	N/A N/A
~~SR 3.3.10.3~~	~~Verify at least one decade overlap with power range neutron flux channels.~~	~~Once each reactor startup prior to intermediate range indication exceeding 1E-6 amp if not performed within the previous 7 days~~	16

<INSERT 3.3-26A> 15

<INSERT 3.3-26A>

			CTS
SR 3.3.10.4	Verify at least one decade overlap between source range and intermediate range neutron flux channels.	Once each reactor startup prior to source range counts exceeding 10^5 cps.	3.5.1.5

B 3.3 INSTRUMENTATION

B 3.3.1 Reactor Protection System (RPS) Instrumentation

BASES

BACKGROUND

The RPS initiates a reactor trip , if necessary, to protect against violating the core fuel design limits and the Reactor Coolant System (RCS) pressure boundary during ~~anticipated operational occurrences (AOOs)~~ abnormalities. By tripping the reactor, the RPS also assists the Engineered Safety Feature (ESF) Systems in mitigating accidents. 21 edit

The protection and monitoring systems have been designed to assure safe operation of the reactor. This is achieved by ~~specifying~~ identifying limiting safety system settings (LSSS) in terms of parameters directly monitored by the RPS, as well as the LCOs and administrative controls on other ~~reactor system~~ parameters and equipment performance. 21

The LSSS, defined in this Specification as the Allowable Value, in conjunction with the LCOs and administrative controls, establishes the threshold for protective system action to prevent exceeding specified acceptable limits during Design Basis Accidents (DBAs). 21

INSERT B3.3-1A

~~During AOOs, which are those events expected to occur one or more times during the unit's life, the acceptable limit is:~~ edit

a. For accidents other than locked rotor, the departure from nucleate boiling ratio (DNBR) shall be maintained above the Safety Limit (SL) value. For the locked rotor accident, the minimum DNBR shall not be less than the applicable critical heat flux correlation limit, or fuel cladding shall be shown to experience no significant temperature excursions 6 21

b. Fuel centerline ~~melt shall not occur~~ temperature shall be maintained below the SL value; ~~and~~

c. The RCS pressure SL of 2750 psig shall not be exceeded; and 9

d. Reactor power shall not exceed 112% RTP.

Maintaining the parameters within the above values ensures that the offsite dose will be within the ~~10 CFR 20 and~~ 10 CFR 100 criteria during ~~AOOs~~ abnormalities. 21 edit

~~Accidents are events that are analyzed even though they are not expected to occur during the unit's life. The acceptable limit during accidents is that the offsite dose shall be maintained within 10 CFR 100 limits. Meeting the acceptable dose limit for an accident category is considered having acceptable consequences for that event.~~ 21

(continued)

<INSERT B3.3-1A>

Acceptable consequences for accidents are that the offsite dose shall be maintained within 10 CFR 100 limits or other limits approved by the NRC.

During abnormalities, one or more of the following limits is maintained:

BASES

BACKGROUND (continued)

RPS Overview

The RPS consists of four separate redundant protection channels that receive inputs of neutron flux, RCS pressure, RCS flow, ~~RCS~~ reactor outlet temperature, RCS pump status, reactor building (RB) pressure, main feedwater (MFW) pump status, and turbine status. turbine main

21

Figure ~~[ ]~~ 7.1, FSAR, Chapter ~~[7]~~ (Ref. 1), shows the arrangement of ~~a typical~~ the RPS protection channels. A protection channel is composed of measurement channels, a manual trip channel, a reactor trip module (RTM), and ~~CONTROL ROD~~ control rod drive (CRD) trip devices. LCO 3.3.1 provides requirements for the individual measurement channels. These channels encompass all equipment and electronics from the point at which the measured parameter is sensed through the bistable relay contacts in the trip string. LCO 3.3.2, "Reactor Protection System (RPS) Manual Reactor Trip," LCO 3.3.3, "Reactor Protection System (RPS)—Reactor Trip Module (RTM)," and LCO 3.3.4, "~~CONTROL ROD~~ Drive (CRD) Trip Devices," discuss the remaining RPS elements. 21

The RPS instrumentation measures critical unit parameters and compares these to predetermined setpoints. If the setpoint is exceeded, a channel trip signal is generated. The generation of ~~any~~ two trip signals in any of the four RPS channels will result in the trip of the reactor. 21

The Reactor Trip System (RTS) contains multiple CRD trip devices, two AC trip breakers, and two DC trip breaker pairs that provide a path for power to the CRD System. In addition to the safety rods, ~~Additionally,~~ the power for ~~most of the CRDs passes through~~ the regulating rods and APSRs may be interrupted by the electronic trip assembly (ETA) relays. The system has two separate paths (or channels), with each path having either two breakers in series or a breaker and an ETA relay in series. Each path provides independent power to the CRDs. Either path can provide sufficient power to operate all CRDs. Two separate power paths to the CRDs ensure that a single failure that opens one path will not cause an unwanted reactor trip. 21

controlled silicon controlled rectifier (SCR)

The RPS consists of four Each independent protection channels, each containing an RTM. ~~The~~ RTM receives signals from its own measurement channels that indicate a protection channel trip is required. The RTM transmits this signal to its own two-out-of-four trip logic and to the two-out-of-four logic edit

(continued)

BASES

BACKGROUND

RPS Overview (continued)

of the RTMs in the other three RPS channels. Whenever any two RPS channels transmit channel trip signals, the RTM logic in each channel actuates to remove 120 VAC power from its associated CRD trip breaker. and de-energizing ETA relays 21

The reactor is tripped by opening circuit breakers that interrupt the power supply to the CRDs. Six breakers are installed to increase reliability and allow testing of the trip system. A one-out-of-two taken twice logic is used to interrupt power to the rods.

control

of rapidly insertable

manual

The RPS has two bypasses: a shutdown bypass and a channel bypass. Shutdown bypass allows the withdrawal of safety rods for SDM availability and rapid negative reactivity insertion during unit cooldowns or heatups. Channel bypass is used for maintenance and testing. Test circuits in the trip strings allow complete testing of all RPS trip Functions. 21

to provide the

typically

receives input

The RPS operates from the instrumentation channels discussed next. The specific relationship between measurement channels and protection channels differs from parameter to parameter. Three basic configurations are used:

INSERT B3.3-3A

a. Four completely redundant measurements (e.g., reactor coolant flow) with one channel input to each protection channel;

b. Four channels that provide similar, but not identical, measurements (e.g., power range nuclear instrumentation where each RPS channel monitors a different quadrant), with one channel input to each protection channel; and

c. Redundant measurements with combinational trip logic outside of the protection channels and the combined output provided to each protection channel (e.g., main turbine trip instrumentation). 21

These arrangements and the relationship of instrumentation channels to trip Functions are discussed next to assist in understanding the overall effect of instrumentation channel failure. edit

below

(continued)

<INSERT B3.3-3A>

Also, an automatic bypass is provided at low power levels for the Main Turbine Trip and the Loss of Main Feedwater Pump Functions.

BASES

BACKGROUND (continued)

Power Range Nuclear Instrumentation

Power Range Nuclear Instrumentation channels provide inputs to the following trip Functions: RPS

edit

1. Nuclear Overpower

   a. Nuclear Overpower—High Setpoint;

   b. Nuclear Overpower—Low Setpoint;

7. Reactor Coolant Pump to Power;

8. Nuclear Overpower RCS Flow and Measured AXIAL POWER IMBALANCE (Power Imbalance Flow);

9. Main Turbine Trip (~~Control~~ Oil Pressure); and

10. Loss of Main Feedwater ~~(LOMFW)~~ Pumps (Control Oil Pressure).

21

INSERT B 3.3-4A

The power range instrumentation has four linear ~~level~~ channels, one for each core quadrant. Each channel feeds one RPS protection channel. Each channel originates in a detector assembly containing two uncompensated ion chambers. The ion chambers are positioned to represent the top half and bottom half of the core. The individual currents from the chambers are fed to individual linear amplifiers. The summation of the top and bottom is the total reactor power. The ~~difference of the~~ top minus the bottom neutron signal is the measured AXIAL POWER IMBALANCE of the reactor core.

edit

Reactor ~~Coolant System~~ Outlet Temperature

The Reactor ~~Coolant System~~ Outlet Temperature provides input to the following Functions:

21

Reactor

2. RCS High Outlet Temperature; and

5. RCS Variable Low Pressure.

The RCS Outlet Temperature is measured by two resistance elements in each hot leg, for a total of four. One temperature detector is associated with each protection channel.

(continued)

<INSERT B3.3-4A>

The Main Turbine Trip and Loss of Main Feedwater Pumps Functions utilize the Power Range Nuclear Instrumentation only for enabling/disabling the operating bypass at low power levels.

BASES

BACKGROUND (continued)

Reactor Coolant System Pressure

The Reactor Coolant System Pressure provides input to the following Functions:

3. RCS High Pressure;

4. RCS Low Pressure;

5. RCS Variable Low Pressure; and

11. Shutdown Bypass RCS High Pressure.

The RPS inputs of reactor coolant pressure are provided by two pressure transmitters in each hot leg, for a total of four. One sensor is associated with each protection channel.

Reactor Building Pressure

The Reactor Building Pressure measurements provide input only to the Reactor Building High Pressure trip, Function 6. There are four RB High Pressure sensors, one associated with each protection channel.

Reactor Coolant Pump Power Monitoring

Reactor coolant pump power monitors are inputs to the Reactor Coolant Pump to Power trip, Function 7. Each RCP's operating current ~~and voltage~~ is measured by ~~four current transformers and four potential transformers driving four overpower and four underpower relays. Each power monitoring channel consists of an overpower relay and an underpower relay. One channel for each pump is associated with each protection channel.~~ 21

INSERT B3.3-5A

Reactor Coolant System Flow

The Reactor Coolant System Flow measurements are an input to the Nuclear Overpower RCS Flow and Measured AXIAL POWER IMBALANCE trip, Function 8. The reactor coolant flow inputs to the RPS are provided by eight high accuracy differential pressure transmitters, four on each loop, which measure flow edit

(continued)

<INSERT B3.3-5A>

a current transformer providing the current input to the associated RCP underpower relay, and the bus voltage is measured by a potential transformer providing the voltage input to the associated RCP underpower relays. Each RCP underpower relay provides individual RCP status to each protection channel.

BASES

BACKGROUND

Reactor Coolant System Flow (continued)

through calibrated flow tubes. One flow input in each loop is associated with each protection channel.

Main Turbine Automatic Stop Oil Pressure

Main Turbine Automatic Stop Oil Pressure is an input to the Main Turbine Trip (Control Oil Pressure) reactor trip, Function 9. Each of the four protection channels receives turbine status information from the same one of four pressure switches monitoring main turbine automatic stop oil pressure. ~~An open indication will be provided to the RPS on a turbine trip.~~ Contact buffers in each protection channel continuously monitor the status of the contact inputs and initiate an RPS trip when a main turbine trip is indicated. 21

edit 21

Feedwater Pump Control Oil Pressure

Feedwater Pump Control Oil Pressure is an input to the Loss of Main Feedwater Pumps (Control Oil Pressure) trip, Function 10. Control oil pressure is measured by four switches on each feedwater pump. One switch on each pump is associated with each protection channel.

RPS Bypasses

The RPS is designed with two types of manual bypasses: channel bypass and shutdown bypass. 21

Channel bypass provides a method of placing all Functions in one RPS protection channel in a bypassed condition, and shutdown bypass provides a method of leaving the safety rods withdrawn during cooldown and depressurization of the RCS. Each bypass is discussed next.

Channel Bypass

A channel bypass provision is provided to allow for maintenance and testing of the RPS. The use of channel bypass keeps the protection channel trip relay energized regardless of the status of the instrumentation channel or 21

(continued)

BASES

BACKGROUND

Channel Bypass (continued)

the key switch must be operated, and

The bistable relay contacts. To place a protection channel in channel bypass, the other three channels must not be in channel bypass. This is ensured by contacts from the other channels being in series with the channel bypass relay. If any contact is open, the second channel cannot be bypassed. ~~The second condition is the closing of the key switch.~~ When the bypass relay is energized, the bypass contact closes, maintaining the channel trip relay in an energized condition. All RPS trips are reduced to a two-out-of-three logic in channel bypass. 21

An indicator light remains lit while the channel is in bypass.

Only one channel bypass key is accessible for use in the control room.

Shutdown Bypass

allowable Some 21

During unit cooldown, it is ~~desirable~~ to leave ~~the~~ safety rods withdrawn to provide shutdown capabilities in the event of unusual positive reactivity additions (moderator dilution, etc.).

However, the unit is also depressurized as coolant temperature is decreased. If the safety rods are withdrawn and coolant pressure is decreased, an RCS Low Pressure trip will occur at 1800 psig and the rods will fall into the core. To avoid this, the protection system allows the operator to bypass the low pressure trip and maintain shutdown capabilities. During the cooldown and depressurization, the safety rods are inserted prior to the low pressure trip of 1800 psig. The RCS pressure is decreased to less than 1720 psig, then each RPS channel is placed in shutdown bypass.

When an RPS channel is placed in shutdown bypass,

~~In shutdown bypass, a normally closed contact opens and the operator closes the shutdown bypass key switch. This action bypasses~~ the RCS Low Pressure trip, Nuclear Overpower RCS Flow and Measured AXIAL POWER IMBALANCE trip, Reactor Coolant Pump to Power trip, and the RCS Variable Low Pressure trip, and ~~inserts~~ a ~~new~~ RCS High Pressure, ≤1720 ~~1950~~ psig trip. The operator can now withdraw the safety rods for additional ~~SDM~~ rapidly insertable negative reactivity. 19

are bypassed

and a Nuclear Overpower Low setpoint trip, ≤ 5% RTP, are inserted.

The insertion of the ~~new~~ high pressure trip ~~performs two functions. First,~~ with a trip setpoint of ≤ 1720 psig, ~~the bistable~~ prevents operation at normal system pressure, approximately 2155 psig, with a portion of the RPS bypassed. ~~The second~~

(continued)

BASES

BACKGROUND Shutdown Bypass (continued)

and ~~function is to~~ ensures that the bypass is removed prior to normal operation. When the RCS pressure is increased during a unit heatup, the safety rods are inserted prior to reaching 1720 psig. The shutdown bypass is removed, which returns the RPS to normal, and system pressure is increased to greater than 1800 psig. ~~The~~ All or some of the safety rods are then withdrawn and normally remain at the full out condition for the rest of the heatup. (19) (21)

~~In addition to the Shutdown Bypass RCS High Pressure trip, the high flux trip setpoint is administratively reduced to 5% RTP while the RPS is in shutdown bypass. This~~ The insertion of the Nuclear Overpower Low Setpoint Trip provides a backup to the Shutdown Bypass RCS High Pressure trip ~~and allows low temperature physics testing~~ while preventing the generation of any significant amount of power. (19) (21)

Module Interlock and Test Trip Relay

Each channel and each trip module is capable of being individually tested. When a module is placed into the test mode, it causes the test trip relay to open and to indicate an RPS channel trip. Under normal conditions, the channel to be tested is placed in bypass before a module is tested.

Trip Setpoints/Allowable Value

The trip setpoints are the ~~normal~~ nominal values at which the bistables are set. Any bistable is considered to be properly adjusted when the "as left" value is within the band for CHANNEL CALIBRATION accuracy ~~(i.e., ± [rack calibration + comparator setting accuracy])~~. edit

The trip setpoints used in the bistables are based on the analytical limits ~~stated~~ used in the safety analysis described in ~~F~~SAR, Chapter ~~[14]~~ (Ref. 2) and Chapter 3A. The selection of these trip setpoints is such that adequate protection is provided when ~~all~~ appropriate sensor and processing time delays are taken into account. To allow for calibration tolerances, instrumentation uncertainties, instrument drift, and ~~severe~~ environment errors ~~for those RPS channels that must function in harsh environments as defined by 10 CFR 50.49 (Ref. 3)~~, the Allowable Values specified in Table 3.3.1-1 ~~in the accompanying LCO~~ are equal to or conservatively (21)

(continued)

The explicit uncertainties are addressed in the individual design calculations as required

BASES

BACKGROUND

Trip Setpoints/Allowable Value (continued) Guidance

the uncertainties associated with

adjusted with respect to the analytical limits. A detailed description of the methodology used to calculate the trip setpoints, ~~including their explicit uncertainties,~~ is provided in "[Unit Specific Setpoint Methodology]" (Ref. 3). The ~~actual~~ nominal trip setpoint entered into the bistable is more conservative than that specified by the Allowable Value to account for changes in random measurement errors detectable by a CHANNEL FUNCTIONAL TEST. One example of such a change in measurement error is drift during the Surveillance Frequency. A channel is inoperable if its ~~actual~~ trip setpoint is not within its required Allowable Value. as-found instrument

Instrument Loop Error Analysis and Setpoint Methodology Manual, Design Guide, IDG-001

may be

21

edit

edit

Setpoints in accordance with the Allowable Value ensure that the limits of Chapter 2.0, "Safety Limits," in the Technical Specifications are not violated during AOOs and that the consequences of DBAs will be acceptable, providing the unit is operated from within the LCOs at the onset of the AOO or DBA and the equipment functions as designed. Note that in LCO 3.3.1 the Allowable Values listed in Table 3.3.1-1 are the LSSS.

abnormalities

abnormality

analyzed

21

Trip setpoint

Each channel can be tested online to verify that the signal and ~~setpoint accuracy~~ are within the specified allowance requirements of ~~Reference 4~~. Once a designated channel is taken out of service for testing, a simulated signal ~~is~~ may be injected in place of the field instrument signal. The process equipment for the channel in test ~~is~~ may then be tested, verified, and calibrated. Surveillances for the channels are specified in the SR section.

Approved Calibration procedures

edit

21

~~The Allowable Values listed in Table 3.3.1-1 are based on the methodology described in "[Unit Specific Setpoint Methodology]" (Ref. 4), which incorporates all of the known uncertainties applicable for each channel. The magnitudes of those uncertainties are factored into the determination of each trip setpoint. All field sensors and signal processing equipment for these channels are assumed to operate within the allowances of these uncertainty magnitudes.~~

21

BASES (continued)

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

~~Each of the~~ A analyzed accidents and transients can be detected by one or more RPS Functions. The accident analysis contained in ~~Ref. [ ]~~ the SAR, Chapter 14 and Chapter 3A, (Ref. 2), takes credit for most RPS trip Functions. Functions not specifically credited in the ~~safety~~ accident analysis were qualitatively credited in the ~~safety analysis and the~~ NRC staff approved licensing basis for the unit. These Functions are high RB pressure, high RCS temperature, turbine trip, ~~and~~ loss of main feedwater, the shutdown bypass nuclear overpower low setpoint, and shutdown bypass high pressure. These Functions may provide protection for conditions that do not require dynamic transient analysis to demonstrate Function performance. These Functions also serve as backups to Functions that were credited in the safety analysis.

21

The LCO requires all instrumentation performing an RPS Function to be OPERABLE. Failure of any instrument renders the affected channel(s) inoperable~~, and reduces the reliability of the affected Functions~~. The four channels of each Function in Table 3.3.1-1 of the RPS instrumentation shall be OPERABLE ~~at all times the reactor is critical~~ during its specified Applicability to ensure that a reactor trip will be actuated if needed. Additionally, during shutdown bypass with any CRD trip breaker closed, the applicable RPS Functions must also be ~~available~~ OPERABLE. This ensures the capability to trip the withdrawn CONTROL RODS exists at all times that rod motion is possible. The trip Function channels specified in Table 3.3.1-1 are considered OPERABLE when all channel components necessary to provide a reactor trip are functional and in service for the required MODE or Other Specified Condition listed in Table 3.3.1-1.

21

Required Actions allow maintenance (protection channel) bypass of individual channels, but the bypass activates interlocks that prevent operation with a second channel bypass. Bypass effectively places the unit in a two-out-of-three logic configuration that can still initiate a reactor trip, even with a single failure within the system.

Only the Allowable Values are specified for each RPS trip Function in the LCO. ~~Nominal~~ trip setpoints are specified in the ~~unit specific~~ setpoint calculations or calibration procedures. The ~~nominal~~ setpoints are selected ~~to ensure~~ such that the setpoint measured by CHANNEL FUNCTIONAL TESTS ~~does not~~ is not expected to exceed the Allowable Value if the bistable is performing as required. ~~Operation with a trip setpoint less conservative than the nominal trip setpoint, but within its Allowable Value, is acceptable~~

21

edit

(continued)

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

~~provided that operation and testing are consistent with the assumptions of the unit specific setpoint calculations. Each Allowable Value specified is more conservative than instrument uncertainties appropriate to the trip Function. These uncertainties are defined in the "[Unit Specific Setpoint Methodology]" (Ref. 4).~~ edit

21

For most RPS Functions, the ~~trip setpoint~~ Allowable Value is to ensure that the departure from nucleate boiling (DNB) or RCS pressure SLs are not challenged. Cycle specific figures for use during operation are contained in the COLR.

Certain RPS trips function to indirectly protect the SLs by detecting specific conditions that do not immediately challenge SLs but will eventually lead to challenge if no action is taken. These trips function to minimize the unit consequences of transients caused by the specific conditions. The Allowable Value for these Functions is selected at the minimum specified deviation from normal values that will indicate the condition, without risking spurious trips due to normal fluctuations in the measured parameter. edit edit

The Allowable Values for bypass removal Functions are stated in the Applicable MODE or Other Specified Condition column of Table 3.3.1-1.

The safety analyses applicable to each RPS Function are discussed next.

1. Nuclear Overpower

a. Nuclear Overpower—High Setpoint

The Nuclear Overpower—High Setpoint trip provides protection for the design thermal overpower condition based on the measured out of core fast neutron leakage flux.

The Nuclear Overpower—High Setpoint trip initiates a reactor trip when the neutron power reaches a predefined setpoint at the design overpower limit. Because THERMAL POWER lags the neutron power, tripping when the neutron power reaches the design overpower will limit THERMAL POWER to a maximum value of the design overpower.

(continued)

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

a. Nuclear Overpower—High Setpoint (continued)

Thus, the Nuclear Overpower—High Setpoint trip protects against violation of the DNBR and fuel centerline melt SLs. However, the RCS Variable Low Pressure, and Nuclear Overpower RCS Flow and Measured AXIAL POWER IMBALANCE, also provide ~~more direct~~ protection. The role of the Nuclear Overpower—High Setpoint trip is to limit reactor THERMAL POWER below the highest power at which the other two trips are known to provide protection. 21

The Nuclear Overpower—High Setpoint trip also provides transient protection for rapid positive reactivity excursions during power operations. These events include the rod withdrawal accident, the rod ejection accident, and the steam line break accident. By providing a trip during these events, the Nuclear Overpower—High Setpoint trip protects the unit from excessive power levels and also serves to reduce reactor power to prevent violation of the RCS pressure SL.

Rod withdrawal accident analyses cover a large spectrum of reactivity insertion rates (rod worths), which exhibit slow and rapid rates of power increases. At high reactivity insertion rates, the Nuclear Overpower—High Setpoint trip provides the primary protection. At low reactivity insertion rates, the high pressure trip provides primary protection.

The specified at or Allowable Value is selected to initiate ~~ensure that~~ a trip ~~occurs~~ before reactor power exceeds the highest point at which the RCS Variable Low Pressure and the Nuclear Overpower RCS Flow and Measured AXIAL POWER IMBALANCE trips are analyzed to provide protection against DNB and fuel centerline melt. The Allowable Value does not account for harsh environment induced errors, because the trip will actuate prior to degraded environmental conditions being reached. 21

(continued)

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

b. Nuclear Overpower—Low Setpoint

While in shutdown bypass, with the Shutdown Bypass RCS High Pressure trip OPERABLE, the Nuclear Overpower—Low Setpoint trip must be reduced to ≤ 5% RTP. The low power setpoint, in conjunction with the lower Shutdown Bypass RCS High Pressure setpoint, ensure that the unit is protected from excessive power conditions when other RPS trips are bypassed.

19

21

is instated with a trip setpoint of

protect

The setpoint Allowable Value was chosen to be as low as practical and still lie within the range of the out of core instrumentation.

edit

Reactor

2. RCS High Outlet Temperature

Reactor

The RCS High Outlet Temperature trip, in conjunction with the RCS Low Pressure and RCS Variable Low Pressure trips, provides protection for the DNBR SL. A trip is initiated whenever the reactor vessel outlet temperature approaches the conditions necessary for DNB. Portions of each RCS High Outlet Temperature trip channel are common with the RCS Variable Low Pressure trip. The RCS High Outlet Temperature trip provides steady state protection for the DNBR SL.

The RCS High Outlet Temperature trip limits the maximum RCS temperature to below the highest value for which DNB protection by the Variable Low Pressure trip is ensured. The trip setpoint Allowable Value is selected to ensure that a trip occurs before hot leg temperatures reach the point beyond which the RCS Low Pressure and Variable Low Pressure trips are analyzed. Above the high temperature trip, the variable low pressure trip need not provide protection, because the unit would have tripped already. The setpoint Allowable Value does not reflect errors induced by harsh environmental conditions that the equipment is expected to experience because the trip ~~is not required to mitigate accidents that create harsh conditions in the RB~~.

21

initiate

will actuate prior to degraded environmental conditions being reached

21

(continued)

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

3. RCS High Pressure

The RCS High Pressure trip works in conjunction with the pressurizer ~~and main steam~~ safety valves to prevent RCS overpressurization, thereby protecting the RCS High Pressure SL. 21

The RCS High Pressure trip has been credited in the accident analysis calculations for slow positive reactivity insertion transients (rod withdrawal accidents and moderator dilution) ~~and loss of feedwater accidents~~. The rod withdrawal accidents cover a large spectrum of reactivity insertion rates and rod worths that exhibit slow and rapid rates of power increases. At high reactivity insertion rates, the Nuclear Overpower—High Setpoint trip provides the primary protection. At low reactivity insertion rates, the RCS High Pressure trip provides the primary protection. 21

exceeded such

The ~~setpoint~~ Allowable Value is selected ~~to ensure~~ that the RCS High Pressure SL is not ~~challenged~~ during steady state operation or slow power increasing transients. The ~~setpoint~~ Allowable Value does not reflect errors induced by harsh environmental conditions because the ~~equipment is not required to mitigate accidents that create harsh conditions in the RB~~. 21

trip will actuate prior to degraded environmental conditions being reached

4. RCS Low Pressure

Reactor

The RCS Low Pressure trip, in conjunction with the ~~RCS~~ High ~~Outlet~~ Temperature and Variable Low Pressure trips, provides protection for the DNBR SL. A trip is initiated ~~whenever the system pressure approaches~~ the conditions necessary for DNB. The RCS Low Pressure trip provides the DNB low pressure limit for the RCS Variable Low Pressure trip. 21

prior to reactor outlet temperature exceeding

the

edit

initiate

The RCS Low Pressure ~~setpoint~~ Allowable Value is selected to ~~ensure that~~ a reactor trip ~~occurs~~ before RCS pressure is reduced below the lowest point at which the RCS Variable Low Pressure trip is analyzed. The RCS Low Pressure trip provides protection for primary system depressurization events and has been 21

(continued)

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

4. RCS Low Pressure (continued)

credited in the accident analysis calculations for small break loss of coolant accidents (LOCAs). Consequently, harsh RB conditions created by small break LOCAs can affect performance of the RCS pressure sensors and transmitters. Therefore, degraded environmental conditions are considered in the Allowable Value determination.

5. RCS Variable Low Pressure

The RCS Variable Low Pressure trip, in conjunction with the ~~RCS~~ Reactor ~~High~~ Outlet Temperature and RCS Low Pressure trips, provides protection for the DNBR SL. A trip is initiated ~~whenever~~ prior to the system parameters of pressure and temperature ~~approach~~ exceeding the conditions necessary for DNB. The RCS Variable Low Pressure trip provides a floating low pressure trip based on the ~~RCS~~ ~~High~~ reactor Outlet Temperature expressed in degrees Fahrenheit within the range specified by the ~~RCS~~ Reactor ~~High~~ Outlet Temperature and RCS Low Pressure trips.

The RCS Variable Low Pressure ~~setpoint~~ Allowable Value is selected to ~~ensure that~~ initiate a trip ~~occurs when~~ prior to temperature and pressure ~~approach~~ exceeding the conditions necessary for DNB while operating in a temperature pressure region constrained by the low pressure and high temperature trips. The RCS Variable Low Pressure trip is not assumed for transient protection in the unit safety analysis; therefore, ~~determination of~~ the ~~setpoint~~ Allowable Value does not account for errors induced by a harsh RB environment.

21

edit

6. Reactor Building High Pressure

The Reactor Building High Pressure trip provides an early indication of a high energy line break (HELB) inside the RB. By detecting changes in the RB pressure, the RPS can provide a reactor trip before the other system parameters have varied significantly. Thus, this trip acts to minimize accident consequences. ~~It also provides a backup for RPS trip instruments exposed to an RB HELB environment.~~ 21

(continued)

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

6. Reactor Building High Pressure (continued)

The Allowable Value for RB High Pressure trip is set at the lowest value consistent with avoiding spurious trips during normal operation. ~~The electronic components of the RB High Pressure trip are located in an area that is not exposed to high temperature steam environments during HELB transients. The components are exposed to high radiation conditions. Therefore, the determination of the setpoint Allowable Value accounts for errors induced by the high radiation.~~

<INSERT B33-16A>

21

7. Reactor Coolant Pump to Power

The Reactor Coolant Pump to Power trip provides protection for changes in the reactor coolant flow due to the loss of multiple RCPs. Because the flow reduction lags loss of power indications due to the inertia of the RCPs, the trip initiates protective action earlier than a trip based on a measured flow signal.

The trip also prevents operation may be with both pumps in either coolant loop tripped. Under these conditions, core flow and core fluid mixing are insufficient for adequate heat transfer. Thus, the Reactor Coolant Pump to Power trip functions to protect the DNBR and fuel centerline ~~melt~~ temperature SLs.

21

The Reactor Coolant Pump to Power trip has been credited in the accident analysis calculations for the loss of four RCPs. The trip also provides ~~the primary~~ protection for the loss of a pump or pumps, which would result in both pumps in a single steam generator loop being tripped.

edit

The Allowable Value for the Reactor Coolant Pump to Power trip setpoint is selected to prevent normal power operation unless at least ~~three RCPs are~~ one RCP is in each loop operating. RCP status is monitored by power transducers ~~on~~ associated with each pump. These relays indicate a ~~loss of an RCP on overpower with an Allowable Value of ≥ 14,400 kW and on underpower with an Allowable Value of ≤ 1752 kW. The overpower Allowable Value is selected low enough to detect locked rotor conditions~~

18

edit

21

(continued)

<INSERT B 3.3-16A>

Even in the case where this trip is a backup for other RPS trips for LOCA or MSLB, it is assumed to occur before degraded building conditions have an appreciable effect on RB High Pressure trip components. Therefore, determination of the Allowable Value does not account for errors induced by a harsh environment.

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

7. Reactor Coolant Pump to Power (continued)

~~(although credit is not allowed for this capability) but high enough to avoid a spurious trip on the in rush current when the pumps start.~~ 21 The underpower Allowable Value is selected to reliably trip on loss of voltage to the RCPs. Neither the reactor power nor the pump power Allowable Value account for instrumentation errors caused by harsh environments because the trip Function is not required to respond to events that could create harsh environments around the equipment.

8. Nuclear Overpower RCS Flow and Measured AXIAL POWER IMBALANCE

The Nuclear Overpower RCS Flow and Measured AXIAL POWER IMBALANCE trip provides steady state protection for the ~~power imbalance~~ reactor core SLs. A reactor trip is initiated ~~when~~ prior to the core power, AXIAL POWER IMBALANCE, and reactor coolant flow conditions ~~indicate an approach to~~ exceeding the DNB or fuel centerline ~~melt~~ temperature limits. 21

This trip supplements the protection provided by the Reactor Coolant Pump to Power trip, through the power to flow ratio, for loss of reactor coolant flow events. The power to flow ratio provides direct protection for the ~~DNBR SL for~~ limiting loss of flow transient which is the loss of ~~a single RCP and for locked RCP rotor accidents.~~ two RCPs from four pump operation. The imbalance portion of the trip is credited for steady state protection only. 21

The power to flow ratio of the Nuclear Overpower RCS Flow and Measured AXIAL POWER IMBALANCE trip also provides steady state protection to prevent reactor power from exceeding the allowable power when the primary system ~~flow rate is less than full four~~ is operating with two or three pump flow. Thus, the power to flow ratio prevents overpower conditions similar to the Nuclear Overpower trip. This protection ensures that during reduced flow conditions the core power is maintained below that required to begin DNB. 21

The Allowable Value is selected to ensure that a trip occurs ~~when the~~ prior to core power, axial power peaking, and 21

(continued)

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

8. Nuclear Overpower RCS Flow and Measured AXIAL POWER IMBALANCE (continued)

temperature

reaching

reactor coolant flow conditions, indicates an approach to DNB or fuel centerline melt limits. By measuring reactor coolant flow and by tripping only when conditions approach an SL, the unit can operate with the loss of one pump from a four pump initial condition. The Allowable Value for this Function is given in the unit COLR because the cycle specific core peaking changes affect the Allowable Value.

21

9. Main Turbine Trip (Control Oil Pressure)

21

edit

tripped

The Main Turbine Trip Function trips the reactor when the main turbine is lost at high power levels. The Main Turbine Trip Function provides an early reactor trip in anticipation of the loss of heat sink associated with a turbine trip. The Main Turbine Trip Function was added to the B&W designed units in accordance with NUREG-0737 (Ref. 8) following the Three Mile Island Unit 2 accident. The trip lowers the probability of an RCS power operated relief valve (PORV) actuation for turbine trip cases. This trip is activated at higher power levels, thereby limiting the range through which the Integrated Control System must provide an automatic runback on a turbine trip.

electromatic

(ERV)

21

Each of the four turbine oil pressure switches feeds all four protection channels through buffers that continuously monitor the status of the contacts. Therefore, failure of any pressure switch affects all protection channels.

a

One of the

only one

21

For the Main Turbine Trip (Control Oil Pressure) bistable, the Allowable Value of 45 psig is selected to provide a trip whenever feedwater pump control oil pressure drops below the normal operating range. To ensure that the trip is enabled as required by the LCO, the reactor power bypass is set with an Allowable Value of 45% RTP. The turbine trip is not required to protect against events that can create a harsh environment in the turbine building. Therefore, errors induced by harsh environments are not included in the determination of the setpoint Allowable Value.

≥40.5

main turbine

21

≤

to a

(continued)

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

10. Loss of Main Feedwater Pumps (Control Oil Pressure)

edit-

The Loss of Main Feedwater Pumps (Control Oil Pressure) trip provides a reactor trip at high power levels when both MFW pumps are ~~lost~~ tripped. The trip provides an early reactor trip in anticipation of the ~~the LOMFW~~ a loss of main feedwater. This trip was added in accordance with NUREG-0737 (Ref. ~~5~~ 4) following the Three Mile Island Unit 2 accident. This trip provides a reactor trip ~~at high power levels~~ for a ~~LOMFW~~ loss of main feedwater to minimize challenges to the PORV. {E} (21)

For the feedwater ~~pump~~ control oil pressure bistable, the Allowable Value of ~~55~~ {255.5} psig is selected to provide a trip whenever feedwater pump control oil pressure drops below the normal operating range. To ensure that the trip is enabled as required by the LCO, the reactor power bypass is set ~~with an Allowable~~ to a Value of ~~15%~~ <10% RTP. The Loss of Main Feedwater Pumps (Control Oil Pressure) trip is not required to protect against events that can create a harsh environment in the turbine building. Therefore, errors caused by harsh environments are not included in the determination of the setpoint Allowable Value. (21)

11. Shutdown Bypass RCS High Pressure

The RPS Shutdown Bypass ~~RCS High Pressure~~ is provided to allow for withdrawing the CONTROL RODS ~~prior to reaching~~ while operating below the normal RCS Low Pressure trip setpoint. The shutdown bypass ~~provides trip protection during deboration and RCS heatup by allowing~~ allows the operator to withdraw the safety groups of CONTROL RODS. This makes their negative reactivity available to terminate inadvertent reactivity excursions. <INSERT B 3.3-19A> ~~Use of the shutdown bypass trip requires that the neutron power trip setpoint be reduced to 5% of full power or less. The Shutdown Bypass RCS High Pressure trip forces a reactor trip to occur whenever the unit switches from power operation to shutdown bypass or vice versa.~~ ~~This ensures that~~ RPS trips cannot be bypassed unless the CONTROL RODS are all inserted. ~~and the flux distribution is known before power operation can begin. The operator is required to remove the shutdown bypass, reset the Nuclear~~ (19)

(continued)

<INSERT B 3.3-19A>

Because the shutdown bypass high pressure trip setpoint is below the normal RCS low pressure trip setpoint, the reactor must be tripped while passing between these two setpoints.

BASES

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

11. Shutdown Bypass RCS High Pressure (continued)

~~Overpower—High Power trip setpoint and again withdraw the safety rod groups before proceeding with startup.~~ (19)

and Chapter 3A

include

Accidents analyzed in the ~~F~~SAR, Chapter ~~[14]~~ (Ref. 2), do not ~~describe~~ events that occur during shutdown bypass operation~~, because the consequences of these events are enveloped by the events presented in the FSAR~~. (21)

active with a setpoint of ≤

are

During shutdown bypass operation with the Shutdown Bypass RCS High Pressure trip active with a setpoint of ≤ ~~[~~1720~~]~~ psig and the Nuclear Overpower—Low Setpoint ~~set at or below~~ 5% RTP, the trips listed below ~~can be~~ bypassed. ~~Under these conditions, the Shutdown Bypass RCS High Pressure trip and the Nuclear Overpower—Low Setpoint trip act to prevent unit conditions from reaching a point where actuation of these Functions is necessary.~~ (19)

~~1.a Nuclear Overpower—High Setpoint;~~

4. RCS Low Pressure;

5. RCS Variable Low Pressure;

7. Reactor Coolant Pump to Power; and

8. Nuclear Overpower RCS Flow and Measured AXIAL POWER IMBALANCE.

Nuclear Overpower—Low Setpoint

initiate

The Shutdown Bypass ~~RCS High Pressure Function's~~ Allowable Value is selected to ~~ensure~~ a trip ~~occurs~~ before producing significant THERMAL POWER. (19)

General Discussion

The RPS satisfies Criterion 3 of ~~the NRC Policy Statement~~ 10CFR50.36 (Ref. 5). (21)

In MODES 1 and 2,

~~In MODES 1 and 2, the following trips shall be OPERABLE because the reactor is critical in these MODES. These trips are designed to take the reactor subcritical to maintain the SLs during AOOs and to assist the ESFAS in providing acceptable consequences during accidents.~~ (21)

In MODES 3, 4, and 5 with any CRD trip breaker in the closed position and the CRD System capable of rod withdrawal, the RPS satisfies Criterion 4 of 10CFR50.36.

(continued)

BASES

21

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY

11. Shutdown Bypass RCS High Pressure (continued)

1.a Nuclear Overpower—High Setpoint;

2. RCS High Outlet Temperature;

3. RCS High Pressure;

4. RCS Low Pressure;

5. RCS Variable Low Pressure;

6. Reactor Building High Pressure;

7. Reactor Coolant Pump to Power; and

8. Nuclear Overpower RCS Flow and Measured AXIAL POWER IMBALANCE.

Functions 1, 4, 5, 7, and 8 just listed may be bypassed in MODE 2 when RCS pressure is below [1720] psig, provided the Shutdown Bypass RCS High Pressure and the Nuclear Overpower—Low setpoint trip are placed in operation. Under these conditions, the Shutdown Bypass RCS High Pressure trip and the Nuclear Overpower—Low setpoint trip act to prevent unit conditions from reaching a point where actuation of these Functions is necessary.

INSERT B 3.3-21A

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only

Two other Functions are required to be OPERABLE during portions of MODE 1. These are the Main Turbine Trip (Control Oil Pressure) and the Loss of Main Feedwater Pumps (Control Oil Pressure) trip. These Functions are required to be OPERABLE above [45]% RTP and [7.5]% RTP, respectively. ≥10% Analyses presented in BAW-1893 (Ref. 6) have shown that for operation below these power levels, these trips are not necessary to minimize challenges to the PORVs as required by NUREG-0737 (Ref. 5). 4

at ≥

E

21

Because the only safety function of the RPS is to trip the CONTROL RODS, the RPS is not required to be OPERABLE in MODE 3, 4, or 5 if the reactor trip breakers are open, or the CRD System is incapable of rod withdrawal. Similarly, the RPS is not required to be OPERABLE in MODE 6 when the CONTROL RODS are decoupled from the CRDs.

either

2

because

normally

(continued)

<INSERT B3.3-21A>

In MODE 1; in MODE 2, when not operating in shutdown bypass; and in MODE 3, when not operating in shutdown bypass but with any CRD trip breaker in the closed position and the CRD system capable of rod withdrawal, the following trips are required to be OPERABLE. These trips function to ensure that any withdrawn CONTROL RODS can be automatically inserted to make or maintain the reactor subcritical. 4

1.a. Nuclear Overpower-High Setpoint; and

3. RCS High Pressure.

In MODES 1 and 2, the following trips are required to be OPERABLE. These trips function as primary or as back-up trips to ensure that any withdrawn CONTROL RODS can be automatically inserted to make or maintain the reactor subcritical.

2. Reactor Outlet High Temperature: and

6. Reactor Building High Pressure.

In addition, Function 6, Reactor Building High Pressure, is required to be OPERABLE in MODE 3, whenever any CRD trip breaker is closed and the CRD system is capable of rod withdrawal. In this MODE, this Function serves purely as a back-up to other required Functions.

In MODE 1 and in MODE 2, when not in shutdown bypass operation, the following trips are required to be OPERABLE. These Functions operate to ensure that any withdrawn CONTROL RODS can be automatically inserted to make or maintain the reactor subcritical. These Functions are all bypassed when the channel is placed in a shutdown bypass condition. Therefore, they are not required to be OPERABLE during shutdown bypass operation.

4. RCS Low Pressure;

5. RCS Variable Low Pressure;

7. Reactor Coolant Pump to Power; and

8. Nuclear Overpower RCS Flow and Measured AXIAL POWER IMBALANCE.

4

BASES

during shutdown bypass operation

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

However, ~~in MODE 2, 3, 4, or 5,~~ the Shutdown Bypass RCS High Pressure and Nuclear Overpower—Low setpoint trips are required to be OPERABLE if the CRD trip breakers are closed and the CRD System is capable of rod withdrawal. Under these conditions, the Shutdown Bypass RCS High Pressure and Nuclear Overpower—Low setpoint trips ~~are sufficient to prevent an approach to~~ sufficiently reduce the potential for conditions that could challenge SLs.

ACTIONS

Conditions A, B, and C are applicable to all RPS protection Functions. If a channel's trip setpoint is found nonconservative with respect to the required Allowable Value in Table 3.3.1-1, or the transmitter, instrument loop, signal processing electronics or bistable is found inoperable, the channel must be declared inoperable and all applicable Condition~~A or Conditions A and B~~ entered immediately.

5 2

~~When the number of inoperable channels in a trip Function exceed those specified in the related Conditions associated with a trip Function, then the unit is outside the safety analysis. Therefore, LCO 3.0.3 must be immediately entered if applicable in the current MODE of operation.~~

~~[Reviewer's Note: If a unit is to take credit for topical reports as the basis for justifying Completion Times, the reports must be supported by an NRC Staff Safety Evaluation Report (SER) that establishes the acceptability of each topical report for that unit.]~~ EDIT

1

A.1 and A.2

1

If one or more Functions in one protection channel become inoperable, the affected protection channel must be placed in bypass or trip, or the bypass of the remaining channels prevented. If the channel is bypassed, all RPS Functions are placed in a two-out-of-three logic configuration and the bypass of any other channel is prevented. In this configuration, the RPS can still perform its safety function in the presence of a random failure of any single channel. Alternatively, the inoperable channel can be placed in trip. Tripping the affected protection channel places all RPS Functions in a one-out-of-three configuration.

(continued)

BASES

ACTIONS

INSERT B3.3-23A

A.1 and A.2 (continued)

Operation in ~~the two-out-of-three configuration or in the one-out-of-three configuration~~ these configurations may continue indefinitely ~~based on the NRC SER for BAW-10167, Supplement 2 (Ref. 7). In this configuration,~~ because the RPS is capable of performing its trip Function in the presence of any single random failure. The 1 hour Completion Time is sufficient to perform Required Action A.1 or Required Action A.2.

B.1, B.2.1, and B.2.2

INSERT B3.3-23B

For Required Action B.1 and Required Action B.2, if one or more Functions in two protection channels become inoperable, one of two inoperable protection channels must be placed in trip ~~and the other in bypass~~. These Required Actions place all RPS Functions in either a one-out-of-two logic configuration or a one-out-of-three ~~and prevent bypass of a second channel~~. In either of these ~~this~~ configurations the RPS can still perform its safety functions in the presence of a random failure of any single channel. The 1 hour Completion Time is sufficient time to perform Required Action B.1, Required Action B.2.1, and Required Action B.2.2.

C.1

Required Action C.1 directs entry into the appropriate Condition referenced in Table 3.3.1-1. The applicable Condition referenced in the table is Function dependent. If the ~~Each time an inoperable channel has not met any~~ Required Actions of Condition A or B ~~, as applicable,~~ are not met or if more than two channels are inoperable, ~~and the~~ and associated Completion Time has expired, Condition C is entered ~~for that channel and~~ provides for transfer to the appropriate subsequent Condition.

D.1 and D.2

If ~~the~~ Required Action C.1 ~~and associated Completion Time of Condition A or B are not met~~ and Table 3.3.1-1 directs entry into Condition D, the unit must be brought to a MODE in which the specified RPS trip Functions are not required to be OPERABLE. The allowed Completion Time of 6 hours is reasonable, based on operating experience, to reach MODE 3

(1) (1) (2) (21)

(continued)

<INSERT B3.3-23A>

Another option is to maintain the channel, which contains one or more inoperable Functions, in an untripped and unbypassed state. In this case, bypass of the remaining three channels must be prevented. This is accomplished by tagging them, under administrative controls, to prevent their being bypassed. This option assumes that the inoperability of the Function(s) does not require the channel containing the inoperable Function(s) to remain in a tripped condition, and that the channel contains other Functions which remain OPERABLE.

By maintaining the channel in an untripped and unbypassed state, the inoperable Function(s) are in a two-out-of-three logic configuration. This configuration is equivalent to bypassing the channel. However, by maintaining the channel in an untripped and unbypassed condition, the OPERABLE Functions within that channel remain in service in a normal two-out-of-four logic configuration.

<INSERT B3.3-23B>

The second inoperable channel may be bypassed or may be maintained in an untripped and unbypassed condition. If the channel is not bypassed, bypass of the remaining channels must be prevented. This is accomplished by tagging them, under administrative controls, to prevent their being bypassed. This option assumes that the inoperability of the Function(s) in the second channel does not require that channel to remain in a tripped condition, and that the channel contains one or more Function which remains OPERABLE.

BASES

ACTIONS

D.1 and D.2 (continued)

from full power conditions in an orderly manner and to open all CRD trip breakers without challenging plant systems. edit

Unit

E.1

C.1

If the Required Action and associated Completion Time of Condition A or B are not met and Table 3.3.1-1 directs entry into Condition E, the unit must be brought to a MODE in which the specified RPS trip Functions are not required to be OPERABLE. To achieve this status, all CRD trip breakers must be opened. The allowed Completion Time of 6 hours is reasonable, based on operating experience, to open CRD trip breakers without challenging plant systems. 21 edit

Unit

F.1

C.1

If the Required Action and associated Completion Time of Condition A or B are not met and Table 3.3.1-1 directs entry into Condition F, the unit must be brought to a MODE in which the specified RPS trip Function is not required to be OPERABLE. To achieve this status, THERMAL POWER must be reduced to < [45]% RTP. The allowed Completion Time of 6 hours is reasonable, based on operating experience, to reach [45]% RTP from full power conditions in an orderly manner without challenging plant systems. 21 edit edit

Unit

G.1

C.1

If the Required Action and associated Completion Time of Condition A or B are not met and Table 3.3.1-1 directs entry into Condition G, the unit must be brought to a MODE in which the specified RPS trip Function is not required to be OPERABLE. To achieve this status, THERMAL POWER must be reduced to < [15] 10% RTP. The allowed Completion Time of 6 hours is reasonable, based on operating experience, to reach [15] 10% RTP from full power conditions in an orderly manner without challenging plant systems. 21 edit edit

Unit

(continued)

BASES (continued)

SURVEILLANCE REQUIREMENTS

The SRs for each RPS Function are identified by the SRs column of Table 3.3.1-1 for that Function. Most Functions are subject to CHANNEL CHECK, CHANNEL FUNCTIONAL TEST, and CHANNEL CALIBRATION~~, and RPS RESPONSE TIME~~ testing. 3 edit

The SRs are modified by a Note, which ~~The [first] Note~~ directs the reader to Table 3.3.1-1 to determine the correct SRs to perform for each RPS Function.

~~[Reviewer's Note: The CHANNEL FUNCTIONAL TEST Frequencies are based on approved topical reports. For a licensee to use these times, the licensee must justify the frequencies as required by the NRC Staff SER for the topical report.]~~ edit

SR 3.3.1.1

Performance of the CHANNEL CHECK once every 12 hours provides reasonable assurance of prompt identification of ~~ensures that~~ a gross failure of instrumentation ~~has not occurred~~. A CHANNEL CHECK is normally a comparison of the parameter indicated on one channel to the same ~~a similar~~ parameter on other channels. It is based on the assumption that instrument channels monitoring the same parameter should read approximately the same value. Significant deviations between two instrument channels could be an indication of excessive instrument drift in one of the channels or of something even more serious. CHANNEL CHECK will detect gross channel failure; therefore, it is key in verifying that the instrumentation continues to operate properly between each CHANNEL CALIBRATION. 21 edit

Agreement criteria factors including are determined by the unit staff based on a combination of ~~the~~ channel instrument uncertainties~~, including isolation, indication, and readability~~. If a channel is outside the criteria, it may be an indication that the transmitter or the signal processing equipment has drifted outside its limit. If the channels are within the criteria, it is an indication that the channels are OPERABLE. If the channels are normally off scale during times when surveillance is required, the CHANNEL CHECK will only verify that they are off scale in the same direction. Off scale low current loop channels are, where practical, verified to be reading at the bottom of the range and not failed downscale. 21 edit

The Frequency~~, about once every shift,~~ is based on operating experience that demonstrates channel failure is rare. Since edit

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.3.1.1 (continued)

the probability of two random failures in redundant channels in any 12 hour period is extremely low, the CHANNEL CHECK minimizes the chance of loss of protective function due to failure of redundant channels. The CHANNEL CHECK supplements less formal but more frequent checks of channel OPERABILITY during normal operational use of the displays associated with the LCO's required channels.

For Functions that trip on a combination of several measurements, such as the Nuclear Overpower RCS Flow and Measured AXIAL POWER IMBALANCE Function, the CHANNEL CHECK must be performed on each input.

≥ 20%

and once within 24 hours after a THERMAL POWER change of ≥ 10% RTP in one direction

5

32

SR 3.3.1.2

96

This SR is the performance of ~~a~~ heat balance calibration for the power range channels every ~~24~~ hours when reactor power ~~is ≥ 15%~~ RTP. The heat balance calibration consists of a comparison of the results of the calorimetric with the power range channel output. The outputs of the power range channels are ~~normalized~~ calibrated to the calorimetric. If the calorimetric exceeds the Nuclear Instrumentation System (NIS) channel channel output by ≥ ~~[~~2~~]~~% RTP, the NIS is not declared inoperable but must be adjusted. If the NIS channel cannot be properly adjusted, the channel is declared inoperable. A Note clarifies that this Surveillance is required only if reactor power is ≥ ~~15%~~ 20% RTP and that 24 hours is allowed for performing the first Surveillance after reaching ~~15%~~ 20% RTP. ~~At lower power levels, calorimetric data are inaccurate.~~

21

INSERT B 3.3-26A

32

The power range channel's output shall be adjusted consistent with the calorimetric results if the calorimetric exceeds the power range channel's output by ≥ ~~[~~2~~]~~% RTP. The value of ~~[~~2~~]~~% is adequate because this value is assumed in the safety analyses of ~~F~~SAR, Chapter ~~[~~14~~]~~ (Ref. 2). These checks and, if necessary, the adjustment of the power range channels ensure that channel accuracy is maintained within the analyzed error margins. The ~~24~~ hour Frequency is adequate, based on unit operating experience, which demonstrates the change in the difference between the power range indication and the calorimetric results rarely exceeds

96

5

(continued)

<INSERT B3.3-26A>

Two calorimetric calculations are routinely performed. One relies upon primary system parameters and the other relies upon secondary system parameters. The primary calorimetric is generally less accurate than the secondary calorimetric at higher power levels and more accurate at lower power levels. For comparison to the nuclear instrumentation, between 0 and 15% power, only the primary calorimetric (heat balance) is considered. From 15 to 100% power the calorimetric is weighted linearly with only the secondary heat balance being considered at 100% power.

BASES

SURVEILLANCE REQUIREMENTS

SR 3.3.1.2 (continued) 5

96

~~a small fraction of~~ [2]% in any ~~24~~ hour period. Furthermore, the control room operators monitor redundant indications and alarms to detect deviations in channel outputs.

SR 3.3.1.3

≥ 20%

A comparison of power range nuclear instrumentation channels against incore detectors shall be performed at a 31 day Frequency when reactor power is ~~≥ 15%~~ RTP. A Note clarifies that 24 hours is allowed for performing the first Surveillance after reaching ~~15%~~ RTP. If the absolute difference between the power range and incore measurements is ≥ [2]% RTP, the power range channel is not inoperable, but a CHANNEL CALIBRATION that adjusts the measured imbalance to agree with the incore measurements is necessary. If the power range channel cannot be properly recalibrated, the channel is declared inoperable. The calculation of the Allowable Value envelope assumes a difference in out of core to incore measurements of 2.5%. Additional inaccuracies beyond those that are measured are also included in the setpoint envelope calculation. The 31 day Frequency is adequate, considering that long term drift of the excore linear amplifiers is small and burnup of the detectors is slow. Also, the excore readings are a strong function of the power produced in the peripheral fuel bundles, and do not represent an integrated reading across the core. The slow changes in neutron flux during the fuel cycle can also be detected at this interval.

32

20%

greater than the procedural limit

6

AXIAL POWER IMBALANCE

21

SR 3.3.1.4

A CHANNEL FUNCTIONAL TEST is performed ~~on each required RPS channel~~ to ensure that the entire channel will perform the intended function. Setpoints must be found within the Allowable Values specified in Table 3.3.1-1. Any setpoint adjustment shall be consistent with the assumptions of the current ~~unit specific~~ setpoint analysis. 21

edit

~~The as found and as left values must also be recorded and reviewed for consistency with the assumptions of the~~ 7

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.3.1.4 (continued)

surveillance interval extension analysis. The requirements for this review are outlined in BAW-10167 (Ref. 8). 7

The Frequency of [45] days on a STAGGERED TEST BASIS is consistent with the calculations of Reference 7 that indicate the RPS retains a high level of reliability for this test interval.

INSERT B3.3-28A

SR 3.3.1.5

INSERT B3.3-28B

This SR is the performance of a CHANNEL CALIBRATION every [92] days. This CHANNEL CALIBRATION normalizes the power range channel output to the calorimetric coincident with the imbalance output being normalized to the imbalance condition predicted by the incore neutron detector system. 30

The calibration for both imbalance and total power is integrated in the power imbalance detector calibration procedure. 5 The ~~[92]~~ day Frequency specified for the Nuclear Overpower trip string is consistent with the drift assumptions made in the calculation of the setpoint ~~"[Unit Specific Setpoint Methodology]" (Ref. 4)~~. 31 Furthermore, operating experience shows the reliability of the trip string is acceptable when calibrated on this interval. A Note clarifies that the neutron detectors are not required to be tested as part of the CHANNEL CALIBRATION. There is no adjustment that can be made to the detectors. Furthermore, adjustment of the detectors is unnecessary because they are passive devices with minimal drift. Slow changes in detector sensitivity are compensated for by performing the daily calorimetric calibration and the monthly axial channel calibration. 30 5

INSERT B3.3-28C

SR 3.3.1.6

A Note to the Surveillance indicates that neutron detectors are excluded from CHANNEL CALIBRATION. This Note is necessary because of the difficulty in generating an appropriate detector input signal. Excluding the detectors is acceptable because the principles of detector operation ensure a virtually instantaneous response.

(continued)

<INSERT B3.3-28A>

The Frequency of 31 days is based on operating experience, which has demonstrated through high reliability of the instrumentation, that failure of more than one channel, of a given Function, in any 31 day interval is rare.

Testing in accordance with this SR is normally performed on a rotational basis, with one channel being tested each week. Testing one channel each week reduces the probability of an undetected failure existing within the system and minimizes the likelihood of the same systematic test errors being introduced into each redundant channel.

<INSERT B3.3-28B>

This SR calibrates the power range (excore) channels to the incore channels every 31 days. This calibration adjusts the power range channel output to the calorimetric heat balance coincident with the imbalance output being calibrated to the imbalance condition predicted by the incore neutron detector system.

<INSERT B3.3-28C>

24 hours is allowed for performing the first Surveillance after reaching 20% RTP.

BASES

SURVEILLANCE REQUIREMENTS

SR 3.3.1.6 (continued)

A CHANNEL CALIBRATION is a complete check of the instrument channel, including the sensor. The test verifies that the channel responds to the measured parameter within the necessary range and accuracy. CHANNEL CALIBRATION leaves the channel adjusted to account for instrument drift to ensure that the instrument channel remains operational between successive tests. ~~CHANNEL CALIBRATION shall find that measurement errors and bistable setpoint errors are within the assumptions of the unit specific setpoint analysis. CHANNEL CALIBRATIONS must be performed consistent with the assumptions of the unit specific setpoint analysis.~~

instrument

INSERT B3.3-29A

21 edit

29

The Frequency is justified by the assumption of ~~an~~ at least ~~[18]~~ month calibration interval in the determination of the allowable magnitude of equipment drift in the setpoint analysis.

21

~~SR 3.3.1.7~~

~~This SR verifies individual channel actuation response times are less than or equal to the maximum values assumed in the accident analysis. Individual component response times are not modeled in the analyses. The analyses model the overall, or total, elapsed time from the point at which the parameter exceeds the analytical limit at the sensor to the point of rod insertion. Response time testing acceptance criteria for this unit are included in Reference 1.~~

~~A Note to the Surveillance indicates that neutron detectors are excluded from RPS RESPONSE TIME testing. This Note is necessary because of the difficulty in generating an appropriate detector input signal. Excluding the detectors is acceptable because the principles of detector operation ensure a virtually instantaneous response.~~

~~Response time tests are conducted on an [18] month STAGGERED TEST BASIS. Testing of the final actuation devices, which make up the bulk of the response time, is included in the testing of each channel. Therefore, staggered testing results in response time verification of these devices every [18] months. The [18] month Frequency is based on unit operating experience, which shows that random failures of~~

3

(continued)

<INSERT B3.3-29A>

Whenever a sensing element is replaced, the next required CHANNEL CALIBRATION of the resistance temperature (RTD) sensors is accomplished by an inplace cross calibration that compares the other sensing elements with the recently installed sensing element.

BASES

SURVEILLANCE REQUIREMENTS

~~SR 3.3.1.7 (continued)~~

~~instrumentation components causing serious response time degradation, but not channel failure, are infrequent occurrences.~~

3

REFERENCES

1. ~~F~~SAR, Chapter ~~[~~7~~]~~.
2. ~~F~~SAR, Chapter ~~[~~14~~]~~, and Chapter 3A.
3. ~~10 CFR 50.49.~~
3 ~~4~~. ~~"[Unit Specific Setpoint Methodology]."~~ Instrument Loop Error Analysis and Setpoint Methodology Manual, Design Guide, IDG-001.
4 ~~5~~. NUREG-0737, "Clarification of TMI Action Plan Requirements," November ~~1979~~ 1980.
5. 10CFR 50.36
6. BAW-1893, "Basis for Raising Arming Threshold for Anticipatory Reactor Trip on Turbine Trip," October 1985.
7. ~~NRC SER for BAW-10167, Supplement 2, July 8, 1992.~~ 1
8. ~~BAW-10167, May 1986.~~ 7

21

B 3.3 INSTRUMENTATION

B 3.3.2 Reactor Protection System (RPS) Manual Reactor Trip

BASES

BACKGROUND

The RPS Manual Reactor Trip provides the operator with the capability to trip the reactor from the control room in the absence of , or coincident with, any other trip condition. Manual trip is provided by a trip push button on the main control board. This push button operates four electrically independent switches~~, one for each train~~. This trip is independent of the automatic trip system. ~~As shown in Figure [...], FSAR, Chapter [7] (Ref. 1), power for the CONTROL ROD drive (CRD) breaker undervoltage coils and contactor coils comes from the reactor trip modules (RTMs). The manual trip switches are located between the RTM output and the breaker undervoltage coils. Opening of the switches opens the lines to the breakers, tripping them. The switches also energize the breaker shunt trip mechanisms. There is a separate switch in series, with the output of each of the four RTMs. All switches are actuated through a mechanical linkage from a single push button.~~

INSERT B3.3-31A

24

APPLICABLE SAFETY ANALYSES

The Manual Reactor Trip ensures that the control room operator can initiate a reactor trip at any time. The Manual Reactor Trip Function is required as a backup to the automatic trip functions~~, and allows operators to shut down the reactor whenever any parameter is rapidly trending toward its trip setpoint~~.

INSERT B3.3-31B

~~The Manual Reactor Trip Function satisfies Criterion 3 of the NRC Policy Statement.~~

24

LCO

The LCO on the RPS Manual Reactor Trip requires that the trip shall be OPERABLE whenever the reactor is critical or any time any ~~control rod~~ CRD breaker is closed and rods are capable of being withdrawn, including shutdown bypass. This enables the operator to terminate any reactivity excursion that in the operator's judgment requires protective action, even if no automatic trip condition exists.

24

(continued)

<INSERT B3.3-31A>

As shown in Figure 7.1, SAR, Chapter 7 (Ref. 1), control power for the control rod drive (CRD) breakers and electronic trip assembly (ETA) relays comes from the reactor trip modules (RTMs). The manual trip switches are located between the RTM output and the breaker undervoltage coils, breaker undervoltage relays, and ETA relays. The switches also initiate actuation of the breaker shunt trip mechanisms. These are separate switches which are actuated through a mechanical linkage from a single push button. Opening of the switches opens the circuits to the breakers, tripping them.

<INSERT B3.3-31B>

Operating experience has shown the Manual Reactor Trip Function to be significant to public health and safety, and therefore satisfy Criterion 4 of 10 CFR 50.36 (Ref. 2).

BASES

LCO (continued)	The Manual Reactor Trip Function is composed of four electrically independent trip switches sharing a common mechanical push button.
APPLICABILITY	The Manual Reactor Trip Function is required to be OPERABLE in MODES 1 and 2. It is also required to be OPERABLE in MODES 3, 4, and 5 if any CRD trip breaker is in the closed position and if the CRD System is capable of rod withdrawal. The ~~only~~ primary safety function of the RPS is to trip the CONTROL RODS; therefore, the Manual Reactor Trip Function is not needed in MODE 3, 4, or 5 if the reactor trip breakers are open or if the CRD System is incapable of rod withdrawal. Similarly, the RPS Manual Reactor Trip is not needed in MODE 6 ~~when~~ because the CONTROL RODS are normally decoupled from the CRDs. 24 24

ACTIONS

A.1

Condition A applies when the Manual Reactor Trip Function is found inoperable. One hour is allowed to restore the Function to OPERABLE status. The automatic functions and various alternative manual trip methods, such as removing power to the RTMs, are still available. The 1 hour Completion Time is sufficient time to correct minor problems. edit

B.1 and B.2

If the Required Action and associated Completion Time are not met ~~With the Manual Reactor Trip Function inoperable and unable to be returned to OPERABLE status within 1 hour~~ in MODE 1, 2, or 3, the unit must be placed in a MODE in which manual trip is not required. Required Action B.1 and Required Action B.2 place the unit in at least MODE 3 with all CRD trip breakers open within 6 hours. The allowed 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. 24

C.1

If the Required Action and associated Completion Time are not met ~~With the Manual Reactor Trip Function inoperable and unable to be returned to OPERABLE status within 1 hour~~ in MODE 4

(continued)

BASES

ACTIONS

C.1 (continued)

or 5, the unit must be placed in a MODE in which manual trip is not required. To achieve this status, all CRD trip breakers must be opened. The allowed Completion Time of 6 hours is reasonable, based on operating experience, to open all CRD trip breakers without challenging unit systems.

SURVEILLANCE REQUIREMENTS

SR 3.3.2.1

This SR requires the performance of a CHANNEL FUNCTIONAL TEST of the Manual Reactor Trip Function. This test verifies the OPERABILITY of the Manual Reactor Trip by actuation of the CRD trip breakers. The Frequency shall be once prior to each reactor startup if not performed within the preceding 7 days to ensure the OPERABILITY of the Manual Reactor Trip Function prior to achieving criticality. The Frequency was developed in consideration that ~~these~~ this Surveillance~~s~~ ~~are~~ is only performed during a unit outage. edit

---

REFERENCES

1. ~~F~~SAR, Chapter [7]. 24
2. 10 CFR 50.36.

B 3.3 INSTRUMENTATION

B 3.3.3 Reactor Protection System (RPS)—Reactor Trip Module (RTM)

BASES

BACKGROUND

The RPS consists of four independent protection channels, each containing an RTM. Figure ~~[1]~~, FSAR, Chapter ~~[7]~~ 7.1 edit (Ref. 1), shows a typical RPS protection channel and the relationship of the RTM to the RPS instrumentation, manual trip, and CONTROL ROD drive (CRD) trip devices. The RTM receives bistable trip signals from the functions in its own channel and channel trip signals from the other three RPS—RTMs. The RTM provides these signals to its own two-out-of-four trip logic and transmits its own channel trip signal to the two-out-of-four logic of the RTMs in the other three RPS channels. Whenever any two RPS channels transmit channel trip signals, the RTM logic in each channel actuates to remove 120 VAC power from its associated CRD trip device. edit

The RPS trip scheme consists of series contacts that are operated by bistables. During normal unit operations, all contacts are closed and the RTM channel trip relay remains energized. However, if any trip parameter exceeds its setpoint, its associated contact opens, which de-energizes the channel trip relay.

When an RTM channel trip relay de-energizes, several things occur:

a. Each of the four (4) output logic relays "informs" its associated RPS channel that a reactor trip signal has occurred in the tripped RPS channel;

b. The contacts in the trip device circuitry, powered by the tripped channel, open, but the trip device remains energized through the closed contacts from the other RTMs. (This condition exists in each RPS—RTM. Each RPS—RTM controls power to a trip device.); and

c. The contact in parallel with the channel reset switch opens and the trip is sealed in. To re-energize the channel trip relay, the channel reset switch must be depressed after the trip condition has cleared.

(continued)

BASES

BACKGROUND (continued)

When the second RPS channel senses a reactor trip condition, the output logic relays for the second channel de-energize and open contacts that supply power to the trip devices. With contacts opened by two separate RPS channels, power to the trip devices is interrupted and the CONTROL RODS fall into the core.

A minimum of two out of four RTMs must sense a trip condition to cause a reactor trip. Also, because the bistable relay contacts for each function are in series with the channel trip relays, two channel trips caused by different trip functions can result in a reactor trip. edit

APPLICABLE SAFETY ANALYSES

Accident analyses rely on a reactor trip for protection of reactor core integrity, reactor coolant pressure boundary integrity, and reactor building OPERABILITY. A reactor trip must occur when needed to prevent accident conditions from exceeding those calculated in the accident analyses. More detailed descriptions of the applicable accident analyses are found in the bases for each of the RPS trip Functions in LCO 3.3.1, "Reactor Protection System (RPS) Instrumentation."

~~RTM response time is included in the overall required response time for each RPS trip and is not specified separately.~~ 3

In MODES 1 and 2, The RTMs satisfy Criterion 3 ~~of the NRC Policy Statement~~ 10CFR 50.36 (Ref. 2). 25

In MODES 3, 4, and 5 with any CRD trip breaker in the closed position and the CRD System capable of rod withdrawal, the RTMs satisfy Criterion 4 of 10 CFR 50.36.

LCO

The RTM LCO requires all four RTMs to be OPERABLE. Failure of any RTM renders a portion of the RPS inoperable ~~and reduces the reliability of the affected Functions~~.

~~Four RTMs must be OPERABLE to ensure that a reactor trip would occur if needed any time the reactor is critical. OPERABILITY is defined as the RTM being~~ To be considered OPERABLE, an RTM must be able to receive and interpret trip signals from its own and other RPS channels and to open its associated trip device. OPERABLE 25

The requirement ~~of~~ for four channels to be OPERABLE ensures that a minimum of two RPS channels will remain OPERABLE if a single failure has occurred in one channel and if a second

(continued)

BASES

LCO (continued)

channel has been bypassed ~~for surveillance or maintenance~~. 25 This two-out-of-four trip logic also ensures that a single RPS channel failure will not cause an unwanted reactor trip. Violation of this LCO could result in a trip signal not causing a reactor trip when needed.

APPLICABILITY

The RTMs are required to be OPERABLE in MODES 1 and 2. They are also required to be OPERABLE in MODES 3, 4, and 5 if any CRD trip breakers are in the closed position and the CRD System is capable of rod withdrawal. The RTMs are designed to ensure a reactor trip would occur, if needed ~~any time the reactor is critical~~. 25 This ~~condition can~~ need may exist in ~~all~~ any of these MODES; therefore, the RTMs must be OPERABLE.

ACTIONS

A.1.1, A.1.2, and A.2

When an RTM is inoperable, the associated CRD trip breaker must then be placed in a condition that is equivalent to a tripped condition for the RTM. Required Action A.1.1 or Required Action A.1.2 requires this either by ~~tripping~~ opening the CRD trip breaker or by removing power to the CRD trip device. Tripping one RTM or removing power opens one set of CRD trip devices. Power to hold up CONTROL RODS is still provided via the parallel CRD trip device(s). Therefore, a reactor trip will not occur until a second protection channel trips. edit

To ensure the trip signal is registered in the other channels, Required Action A.2 requires that the inoperable RTM be removed from the cabinet. This action causes the electrical interlocks to indicate a tripped channel in the remaining three RTMs. Operation in this condition is allowed indefinitely because the actions put the RPS into a one-out-of-three configuration. The 1 hour Completion Time is sufficient time to perform the Required Actions.

B.1, B.2.1, and B.2.2

Condition B applies if two or more RTMs are inoperable in MODE 1, 2, or 3, or if the Required Actions of Condition A are not met ~~within the required Completion Time~~ associated Completion Time 20 25 in MODE 1, 2, or 3. In this case, the unit must be placed in a MODE in

(continued)

BASES

ACTIONS

B.1, B.2.1, and B.2.2 (continued)

trip breakers

which the LCO does not apply. This is done by placing the unit in at least MODE 3 with all CRD trip breakers open or with all power to the CRD System removed within 6 hours. The allowed 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. 31

C.1 and C.2 20

two or more RTMs are inoperable in MODE 4 or 5, or if

and associated Completion Times

Condition C applies if the Required Actions ~~of Condition A~~ are not met ~~within the required Completion Time~~ in MODE 4 or 5. In this case, the unit must be placed in a MODE in which the LCO does not apply. This is done by opening all CRD trip breakers or removing all power to the CRD System. The allowed Completion Time of 6 hours is reasonable, based on operating experience, to open all CRD trip breakers or remove all power to the CRD System without challenging unit systems.

trip breakers

from

25

31

SURVEILLANCE REQUIREMENTS

SR 3.3.3.1

~~The Note defines a channel as being OPERABLE for up to 8 hours while bypassed for Surveillance testing. The Note allows channel bypass for testing without defining it as inoperable although during this time period it cannot actuate a reactor trip. This allowance is based on the assumption of the RPS reliability analysis in BAW-10167 (Ref. 2) that 8 hours is the average time required to perform channel Surveillance. The analysis demonstrated that the 8 hour testing allowance does not significantly reduce the probability that the RPS will trip when necessary. It is not acceptable to routinely remove channels from service for more than 8 hours to perform required Surveillance testing. Such a practice would be contrary to the assumptions of the reliability analysis that justified the LCO's Completion Times.~~ 22

~~[Reviewer's Note: The CHANNEL FUNCTIONAL TEST frequency is based on an approved topical report. For a licensee to use~~ ~~]~~ EDIT

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.3.3.1 (continued)

~~this Frequency, the licensee must justify the Frequency as required by the NRC Staff SER for the topical report.~~ edit

The SRs include performance of a CHANNEL FUNCTIONAL TEST every {92} ~~[45]~~ days ~~on a STAGGERED TEST BASIS~~. This test shall verify the OPERABILITY of the RTM and its ability to receive and properly respond to channel trip and reactor trip signals. ~~Calculations have shown that the Frequency (45 days) maintains a high level of reliability of the Reactor Trip System in BAW-10167 (Ref. 2).~~ 33

<INSERT B 3.3-38A>

REFERENCES

1. ~~F~~SAR, Chapter ~~[7]~~.

~~2. BAW-10167, May 1986.~~ 2. 10CFR 50.36 25

3. BAW-10167A, "Justification for Increasing the Reactor Trip System On-line Test Intervals," Supplement 3, "Justification for increasing the Trip Device Test Interval," February 1998 33

<INSERT B3.3-38A>

The Frequency of 92 days is based on operating experience, which has demonstrated through high reliability of the instrumentation, that failure of more than one RTM in any 92 day interval is rare (Ref. 3).

Testing in accordance with this SR is normally performed on a rotational basis, with one RTM being tested each 23 days. Testing one RTM each 23 days reduces the probability of an undetected failure existing within the system and minimizes the likelihood of the same systematic test errors being introduced into each redundant RTM.

edit

B 3.3 INSTRUMENTATION

B 3.3.4 ~~CONTROL ROD~~ Drive (CRD) Trip Devices

BASES

BACKGROUND

The Reactor Protection System (RPS) contains multiple CRD trip devices: two AC trip breakers, two DC trip breaker pairs, and ~~eight~~ ten electronic trip assembly (ETA) relays. The system has two separate paths (or channels), with each path having one AC breaker ~~in series with either~~ functionally in series with a pair of DC breakers or ~~four~~ five ETA relays in parallel. Each path provides independent power to the CRDs. Either path can provide sufficient power to operate the entire CRD System. 26

Figure ~~7-4~~ 7-10, FSAR, Chapter ~~7~~ (Ref. 1), illustrates the configuration of CRD trip devices. To trip the reactor, power to the CRDs must be removed. Loss of power causes the CRD's mechanisms to release the CONTROL RODS, which then fall by gravity into the core.

Power to CRDs is supplied from two separate unit sources through the AC trip circuit breakers. These breakers are designated A and B, and their undervoltage and shunt trip coils are ~~powered~~ controlled by RPS channels A and B, respectively. From the circuit breakers, the CRD power travels through voltage regulators and stepdown transformers. These devices in turn supply redundant buses that feed the DC holding power supplies and the regulating rod ~~power supplies~~ APSR and auxiliary.

The DC holding power supplies rectify the AC input and supply power to hold the safety rods in their fully withdrawn position. One of the redundant power sources supplies phase A; the other, phase ~~B~~ CC. Either phase being energized is sufficient to hold the rod. Two breakers are located on the output of each power supply. Each breaker controls power to ~~one~~ half of the two of the four safety rod groups. The undervoltage and shunt trip coils on the two circuit breakers on the output of one of the power supplies is controlled by RPS channel C. The other two breakers are controlled by RPS channel D. 26

In addition to the DC holding power supplies, the redundant buses also supply power to the regulating rod, APSR and auxiliary power supplies. ~~These power supplies consist of ETAs that are gated on by programming lamps. Programming lamp power is controlled by contactors (E and F), which are controlled by~~ <INSERT B3.3-39A>

(continued)

<INSERT B3.3-39A>

These power supplies contain silicon controlled rectifiers (SCRs), which are gated on and off to provide power to, and remove power from, the phases of the CRD mechanisms. The gating control signal for these SCRs is supplied through the closed contacts of the ETA relays. These contacts are referred to as E and F contactors, and are controlled by the C and D RPS channels, respectively.

BASES

BACKGROUND (continued)

RPS ~~power~~. One of ~~the~~ redundant ~~programming~~ lamp supplies is ~~controlled~~ by ~~RPS~~ channel C; the ~~other~~, by RPS channel ~~D~~. (26)

The AC breaker and DC breakers , or gated SCRs, are in series in one of the power supplies; whereas, the redundant AC breaker and DC breakers or gated SCRs are in series in the other power supply to the CONTROL RODS. The logic required to cause a reactor trip is the opening of a circuit breaker or ETA relay in each of the redundant power supplies. (The pair of DC circuit breakers on the output of the power supply are treated as one breaker.) This is known as a one-out-of-two taken twice logic. The following examples illustrate the operation of the reactor trip circuit breakers.

a. If the A AC circuit breaker opens:

1. the input power to associated DC power supply is lost, and

2. the SCR supply from the associated power source is lost.

b. If the D DC circuit breaker(s) and F contactors open:

1. the output of the redundant DC power supply is lost and the safety rods de-energize, and (26)

2. when the F contactor opens, SCR gating ~~programming lamp~~ power is lost and the regulating rods will be de-energized.

c. The combination of (a) and (b) causes a reactor trip.

Any other combination of at least one circuit breaker opening in each power supply will cause a reactor trip.

In summary, two tripped RPS channels will cause a reactor trip. For example, a reactor trip occurs if RPS channel B senses a low Reactor Coolant System (RCS) pressure condition and if RPS channel C senses a variable low RCS pressure condition. When the channel B bistable relay de-energizes, the channel trip relay de-energizes and opens its associated contacts. The same thing occurs in channel C, except the variable lower pressure bistable relay de-energizes the channel C trip relay. When the output logic relays in channels B and C de-energize, the B and C contacts in the ~~undervoltage and~~ trip logic of each channel's reactor trip module (RTM) open causing an undervoltage to each trip breaker. (26)

(continued)

and the ETA relay contactors

BASES

All trip

From all CRD mechanisms 26

BACKGROUND (continued)

~~and / contacts de-energize. All circuit~~ breakers open, and ~~programming lamp~~ power is removed. All rods fall into the core, resulting in a reactor trip.

APPLICABLE SAFETY ANALYSES

Accident analyses rely on a reactor trip for protection of reactor core integrity, reactor coolant pressure boundary integrity, and reactor building OPERABILITY. A reactor trip must occur when needed to prevent accident consequences from exceeding those calculated in the accident analyses. The control rod insertion limits ensure that adequate rod worth is available upon reactor trip to shut down the reactor to the required SDM. Further, OPERABILITY of the CRD trip devices ensures that all CONTROL RODS ~~(except Group 8)~~ will trip when required. More detailed descriptions of the applicable accident analyses are found in the Bases for each of the individual RPS trip Functions in LCO 3.3.1, "Reactor Protection System (RPS) Instrumentation." 26

In MODES 1 and 2,

In MODES 3, 4 and 5 with any CRD trip breaker in the closed position and the CRD System capable of rod withdrawal, the CRD trip devices satisfy Criterion 4 of 10CFR 50.36.

the CRD trip devices satisfy Criterion 3 of ~~the NRC Policy Statement.~~ 10CFR 50.36 (Ref. 2). 26

LCO

specified

The LCO requires all of the CRD trip devices to be OPERABLE. Failure of any required CRD trip device renders a portion of the RPS inoperable. ~~and reduces the reliability of the affected Functions.~~ Without reliable CRD reactor trip circuit breakers and associated support circuitry, a reactor trip may not ~~cannot~~ occur when initiated either automatically or manually.

All required CRD trip devices shall be OPERABLE to ensure that the reactor remains capable of being tripped any time it is critical. OPERABILITY is defined as the CRD trip device being able to receive a reactor trip signal and to respond to this trip signal by interrupting ~~AC~~ power to the CRDs. Both of ~~the AC~~ a CRD trip breaker's diverse trip devices and the breaker itself must be functioning properly for the ~~AC~~ breaker to be OPERABLE.

<INSERT 3.3-41A>

Requiring all breakers and ETA relays to be OPERABLE ensures that at least one device in each of the two power paths to the CRDs will remain OPERABLE even with a single failure in MODES 1 and 2, and in MODES 3, 4 and 5 when any CRD trip breaker is in the closed position and the CRD System is capable of rod withdrawal 26

(continued)

<INSERT B3.3-41A>

Both ETA relays associated with each of the three regulating rod groups and the two ETA relays associated with the auxiliary power supply must be OPERABLE to satisfy the LCO. The ETA relays associated with the APSR power supply are not required to be OPERABLE because the APSRs are not designed to fall into the core upon initiation of a reactor trip.

BASES

LCO (continued)

~~Requiring all devices OPERABLE also ensures that a single failure will not cause an unwanted reactor trip.~~ 26

APPLICABILITY

are required to

The CRD trip devices ~~shall~~ be OPERABLE in MODES 1 and 2, and in MODES 3, 4, and 5 when any CRD trip breaker is in the closed position and the CRD System is capable of rod withdrawal. edit

The CRD trip devices are designed to ensure that a reactor trip would occur if needed ~~any time the reactor is critical~~. Since ~~this condition can exist~~ a trip may be required in all of these MODES, the CRD trip devices ~~shall~~ must be OPERABLE. 26 edit

ACTIONS

A Note has been added to the ACTIONS indicating separate Condition entry is allowed for each CRD trip device.

26

~~Condition A~~

Condition A represents reduced redundancy in the CRD trip Function. Condition A applies when:

- One diverse trip Function (undervoltage or shunt trip device) is inoperable in one or more CRD trip breaker(s) ~~(or breaker pair)~~; or
- One diverse trip Function is inoperable in both DC trip breakers associated with one protection channel. In this case, the inoperable trip Function does not need to be the same for both breakers.

A.1 and A.2

If one of the diverse trip Functions on a CRD trip breaker ~~(or breaker pair)~~ becomes inoperable, actions must be taken to preclude the inoperable CRD trip device from preventing a reactor trip when needed. This is done by manually ~~tripping~~ opening the inoperable CRD trip breaker or by removing power from the ~~channel containing the~~ inoperable CRD trip breaker. 26 Either of these actions places the affected CRDs in a one-out-of-two trip configuration, which precludes a single

(continued)

BASES

ACTIONS

A.1 and A.2 (continued)

failure, which in turn could prevent tripping of the reactor. The 48 hour Completion Time has been shown to be acceptable through operating experience.

~~Condition B~~

Condition B represents a loss of redundancy for the CRD trip Function. Condition B applies when

- ~~One or more CRD trip breaker(s) [or breaker pair] will not function on either undervoltage or shunt trip Functions; or~~
- ~~Both diverse trip Functions are inoperable in one or both DC trip breakers associated with one protection channel.~~

more trip breaker(s) or breaker pairs.

26

B.1 and B.2

Required Action B.1 and Required Action B.2 are the same as Required Action A.1 and Required Action A.2, but the Completion Time is shortened. The 1 hour Completion Time allowed to ~~trip~~ open or remove power from the CRD trip breaker allows the operator to take all the appropriate actions for the inoperable breaker and still ensures that the risk involved is acceptable.

edit

C.1 ~~and~~, C.2, C.3 and C.4

8

Condition C represents a loss of redundancy for the CRD trip Function. Condition C applies when one or more ETA relays are inoperable. The preferred action is to restore the ETA relay to OPERABLE status. If this cannot be done, the operator can perform one of ~~two~~ four ~~actions to eliminate reliance on the failed ETA relay~~. The first option is to switch the affected ~~control rod~~ CONTROL ROD group to an alternate power supply which has two OPERABLE or One OPERABLE and one open ETA relay. This removes the failed ETA relay from the trip sequence, and the unit can operate indefinitely. {INSERT B 3.3-43A} The ~~second~~ fourth option is to ~~trip~~ open the corresponding AC CRD trip breaker. This also results in the safety function being performed, thereby eliminating the failed ETA relay from the trip sequence.

8

(continued)

<INSERT B3.3-43A>

The second option is to transfer the affected CONTROL ROD group to a DC holding power supply. This option is only available if the affected group is a safety rod group and the affected power supply is the auxiliary power supply. The third option is to open the inoperable ETA contacts. This option results in the safety function being performed.

BASES

edit

ACTIONS

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

The 1 hour Completion Time is sufficient to perform the Required Action.

D.1, D.2.1, and D.2.2

and associated Completion Times 26

If the Required Actions of Condition A, B, or C are not met within the required Completion Time in MODE 1, 2, or 3, 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, with all CRD trip breakers open or with all power to the CRD System removed within 6 hours. The allowed 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. 31

all

trip breakers

E.1 and E.2

and associated Completion Times 26

If the Required Actions of Condition A, B, or C are not met within the required Completion Time in MODE 4 or 5, the unit must be brought to a MODE in which the LCO does not apply. To achieve this status, all CRD trip breakers must be opened or all power to the CRD System removed within 6 hours. The allowed Completion Time of 6 hours is reasonable, based on operating experience, to open all CRD trip breakers or remove all power to the CRD System without challenging unit systems. 31

trip breakers

from all

SURVEILLANCE REQUIREMENTS

SR 3.3.4.1 33

SR 3.3.4.1 is to perform a CHANNEL FUNCTIONAL TEST every 31 days. This test verifies the OPERABILITY of the trip devices by actuation of the end devices. Also, this test independently verifies the undervoltage and shunt trip mechanisms of the AC breakers. The Frequency of 31 days is based on operating experience, which has demonstrated that failure of more than one channel of a given function in any 31 day interval is a rare event. (Ref. 3) 26

92

trip

<INSERT B 3.3-44A>

REFERENCES

1. FSAR, Chapter [7].

2. 10CFR 50-36

EDIT

3. BAW-10167A, "Justification for Increasing the Reactor Trip System On-line Test Intervals," Supplement 3, "Justification for Increasing the Trip Device Test Interval," February 1998. 33

<INSERT B3.3-44A>

Testing in accordance with this SR is normally performed on a rotational basis, with one channel being tested each 23 days. Testing one channel each 23 days reduces the probability of an undetected failure existing within the system and minimizes the likelihood of the same systematic test errors being introduced into each redundant trip device.

B 3.3 INSTRUMENTATION

B 3.3.9 Source Range Neutron Flux

BASES

BACKGROUND

The source range neutron flux channels provide the operator with an indication of the approach to criticality ~~at lower power levels than can be seen on the intermediate range neutron flux instrumentation~~. These channels also provide the operator with a flux indication that reveals changes in reactivity ~~and helps to verify that SDM is being maintained~~. 23 23

The source range instrumentation has two redundant count rate channels originating in two high sensitivity fission chambers ~~proportional counters~~. Two source range detectors are externally located on opposite sides of the core ~~(180°)~~. These channels are used over a counting range of 0.1 cps to 1E5 ~~1E6~~ cps and are displayed on the operator's control console in terms of log count rate. The channels also measure the rate of change of the neutron flux level, which is displayed for the operator in terms of startup rate from -1 ~~-0.5~~ decades to +7 ~~+5~~ decades per minute. An interlock provides a control rod withdraw "inhibit" on a high startup rate of +2 decades per minute in either channel. This interlock is bypassed when the intermediate range neutron flux channels reach 1E-9 amps or power range neutron flux channels reach 10% RTP. 23

~~The proportional counters of the source range channels are BF_3 chambers. The detector high voltage is automatically turned off when the flux level is approximately one decade above the useful operating range. Conversely, the high voltage is turned on automatically when the flux level returns to within approximately one decade of the detectors' maximum useful range. High voltage will be turned off automatically when the flux level is above 1E-10 amp in both intermediate range channels, or 10% power in power range channels.~~ 23

APPLICABLE SAFETY ANALYSES

The source range neutron flux channels They are ~~are~~ necessary to monitor core reactivity changes. ~~It is~~ the primary means for detecting reactivity changes and triggering operator actions to respond to reactivity transients initiated from conditions in which the Reactor Protection System (RPS) is not required to be OPERABLE. They ~~It~~ also trigger ~~s~~ operator actions to anticipate RPS actuation in the event of reactivity transients starting from shutdown or low power conditions. 23

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

The source range neutron flux channels satisfy Criterion ~~2~~ 4 of ~~the NRC Policy Statement.~~ 10 CFR 50.36 (Ref. 1). 23

9

LCO

One ~~Two~~ source range neutron flux channel~~s~~ shall be OPERABLE ~~whenever the control rods are capable of being withdrawn~~ to provide the operator with ~~redundant~~ source range neutron instrumentation. The source range instrumentation ~~is the~~ provides primary power indication at ~~low power levels~~ ≤ 1E-10 amp on intermediate range instrumentation and must remain OPERABLE for the operator to continue increasing power. 23 12

~~A Note has been added allowing detector high voltage to be de-energized above 1E-10 amp on the intermediate range channels. Above this point, the source range instrumentation is no longer the primary power indicator. As such, the high voltage to the source range detectors may be de-energized.~~ 11

APPLICABILITY

One ~~Two~~ source range neutron flux channel~~s~~ shall be OPERABLE in MODE 2 to provide ~~redundant~~ indication during an approach to criticality. Neutron flux level is sufficient for monitoring on the intermediate range and on the power range instrumentation prior to entering MODE 1; therefore, source range instrumentation is not required in MODE 1. 9

In MODES 3, 4, and 5, source range neutron flux instrumentation shall be OPERABLE to provide the operator with a means of monitoring ~~changes in SDM~~ neutron flux and to provide an early indication of reactivity changes. 23

The requirements for source range neutron flux instrumentation during MODE 6 refueling operations are addressed in LCO 3.9.~~2~~ 2, "Nuclear Instrumentation." edit

ACTIONS

~~A.1~~

~~The Required Action for one channel of the source range neutron flux indication inoperable with THERMAL POWER ≤ 1E-10 amp on the intermediate range neutron flux~~ 9

(continued)

BASES

ACTIONS

~~A.1 (continued)~~

~~instrumentation is to delay increasing reactor power until the channel is repaired and restored to OPERABLE status. This limits power increases in the range where the operators rely solely on the source range instrumentation for power indication. The Completion Time ensures the source range is available prior to further power increases. Furthermore, it ensures that power remains below the point where the intermediate range channels provide primary protection until both source range channels are available to support the overlap verification required by SR 3.3.9.4.~~ (9)

A A A A
B.1, B.2, B.3, and B.4 (9)

the required
With both source range neutron flux channels inoperable with THERMAL POWER ≤ 1E-10 amp on the intermediate range neutron flux instrumentation, the operators must place the reactor in the next lowest condition for which source range instrumentation is not required. This is done by immediately suspending positive reactivity additions, initiating action to insert all CONTROL RODS, and opening the CONTROL ROD drive trip breakers within 1 hour. Periodic SDM verification of ≥ 1% Δk/k is then required to provide a means for detecting the slow reactivity changes that could be caused by mechanisms other than control rod withdrawal or operations involving positive reactivity changes. Since the source range instrumentation provides the only reliable direct indication of power in this condition, the operators must continue to verify the SDM every 12 hours until at least one channel of the source range instrumentation is returned to OPERABLE status. Required Action A B.1, Required Action B.2, and Required Action B.3 preclude rapid positive reactivity additions. The 1 hour Completion Time for Required Action B.3 and Required Action B.4 provides sufficient time for operators to accomplish the actions. The 12 hour Frequency for performing the SDM verification ensures that the reactivity changes possible with CONTROL RODS inserted are detected before SDM limits are challenged. (12) (23) (28) (9) (23)

take actions to limit the possibilities for adding positive reactivity.

edit

CONTROL ROD

these MODES

edit

provides reasonable assurance

<INSERT B 3.3-82A> (23)

B
C.1 (9)

With reactor power > 1E-10 amp in MODE 2, 3, 4, or 5 on the intermediate range neutron flux instrumentation, continued (12)

(continued)

<INSERT B3.3-82A>

If no indication of intermediate range flux is available, these Required Actions are also appropriate.

BASES

ACTIONS

3 ~~C~~.1 (continued)

operation is allowed with ~~one or more~~ the required source range neutron flux channel~~(s)~~ inoperable. The ability to continue operation is justified because the instrumentation does not provide a safety function during high power operation. However, actions are initiated within 1 hour to restore the required channel~~(s)~~ to OPERABLE status for future availability. The Completion Time of 1 hour is sufficient to initiate the action. The action must continue until the channel~~(s)~~ ~~are~~ is restored to OPERABLE status. 9 9

SURVEILLANCE REQUIREMENTS

SR 3.3.9.1

Performance of the CHANNEL CHECK once every 12 hours ~~ensures that~~ provides reasonable assurance of prompt identification of a gross failure of instrumentation ~~has not occurred~~. A CHANNEL CHECK is normally a comparison of the parameter indicated on one channel to ~~a similar~~ the same parameter on other channels. It is based on the assumption that instrument channels monitoring the same parameter should read approximately the same value. Significant deviations between the two instrument channels could be an indication of excessive instrument drift in one of the channels or of something even more serious. CHANNEL CHECK will detect gross channel failure; therefore, it is key in verifying that the instrumentation continues to operate properly between each CHANNEL CALIBRATION. 23 edit

Agreement criteria are determined by the unit staff, based on a combination of ~~the~~ factors including channel instrument uncertainties~~, including isolation, indication, and readability~~. If a channel is outside the criteria, it may be an indication that the ~~transmitter or the~~ signal processing equipment has drifted outside its limit. If the channels are within the criteria, it is an indication that the channels are OPERABLE. If the channels are normally off scale during times when surveillance is required, the CHANNEL CHECK will only verify that they are off scale in the same direction. ~~Off scale low current loop channels are verified to be reading at the bottom of the range and not failed downscale.~~ 23 23 23 edit

The Frequency~~, about once every shift,~~ is based on operating experience that demonstrates channel failure is rare. Since

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.3.9.1 (continued)

the probability of two random failures in redundant channels in any 12 hour period is extremely low. The CHANNEL CHECK minimizes the chance of loss of protective function due to failure of redundant channels. The CHANNEL CHECK supplements less formal, but more frequent, checks of channel OPERABILITY during normal operational use of the displays associated with the LCO's required channels. When operating ~~in Required Action A.1~~, CHANNEL CHECK is still required. However, in this condition, a redundant source range ~~is not~~ available for comparison. CHANNEL CHECK may still be performed via comparison with intermediate range detectors, if available, and verification that the OPERABLE source range channel is energized and indicating a value consistent with current unit status.

with only one channel OPERABLE

may not be

9

SR 3.3.9.2

a

For source range neutron flux channels, CHANNEL CALIBRATION is a complete check and readjustment of the channels from the preamplifier input to the indicators. This test verifies the channel responds to measured parameters within the necessary range and accuracy. CHANNEL CALIBRATION leaves the channel ~~adjusted to account~~ for instrument drift to ensure that the instrument channel remains operational between successive tests.

at a setpoint which accounts

The SR is modified by a Note excluding neutron detectors from CHANNEL CALIBRATION. It is not necessary to test the detectors because generating a meaningful test signal is difficult. ~~The~~ detectors are of simple construction, and any failures in the detectors will be apparent as change in channel output.

, and there is no adjustment that can be made to the detectors. Furthermore, adjustment of the detectors is unnecessary because they are passive devices with minimal drift. Finally, the

edit

The Frequency of ~~{18}~~ months is based on demonstrated instrument CHANNEL CALIBRATION reliability over an ~~{18}~~ month interval, such that the instrument is not adversely affected by drift.

~~SR 3.3.9.3~~

~~SR 3.3.9.3 is the verification of one decade of overlap with the intermediate range neutron flux instrumentation prior to~~

15

(continued)

BASES

SURVEILLANCE REQUIREMENTS

~~SR 3.3.9.3 (continued)~~

~~source range count rate exceeding 10^5 cps if not performed within 7 days prior to reactor startup. This ensures a continuous source of power indication during the approach to criticality. Failure to perform this Surveillance leaves the unit in a safe, subcritical condition until the verification can be made. The test may be omitted if performed within the previous 7 days based on operating experience, which shows that source range and intermediate range instrument overlap does not change appreciably within this test interval.~~

15

REFERENCES

1. 10CFR 50.36

None.

23

B 3.3 INSTRUMENTATION

B 3.3.10 Intermediate Range Neutron Flux

BASES

BACKGROUND

The intermediate range neutron flux channels provide the operator with an indication of reactor power at higher power levels than the source range instrumentation and lower power levels than the power range instrumentation.

The intermediate range instrumentation has two Log (2) channels originating in two ~~electrically identical~~ gamma compensated ion chambers. Each channel provides eight decades of flux level information in terms of the log of ion chamber current from ~~1E-10~~ 1E-11 amp to ~~1E-2~~ 1E-3 amp. The channels also measure the rate of change of the neutron flux level, which is displayed for the operator in terms of startup rate from -0.5 decades to +5 decades per minute. A high startup rate of +3 decades per minute in either channel will initiate a control rod withdrawal inhibit while below 10% RTP. 27 27

The intermediate range compensated ion chambers are of the electrically adjustable gamma compensating type. Each detector has a separate adjustable high voltage power supply and an adjustable compensating voltage supply.

APPLICABLE SAFETY ANALYSES

Intermediate range neutron flux channels provide ~~are~~ necessary to monitor core reactivity changes and ~~are~~ the primary indication to trigger operator actions to anticipate Reactor Protection System actuation in the event of reactivity transients starting from low power conditions. edit

The intermediate range neutron flux channels satisfy Criterion ~~2~~ 4 of ~~the NRC Policy Statement~~ 10CFR 50.36 (Ref. 1). 27

LCO

~~Two~~ One intermediate range neutron flux instrumentation channel~~s~~ shall be OPERABLE to provide the operator with ~~redundant~~ neutron flux indication. ~~These enable~~ This enables operators to control the increase in power and to detect neutron flux transients. This indication is used until the power range instrumentation is on scale. Violation of this requirement could prevent the operator from detecting and controlling 10

(continued)

BASES

LCO (continued)

neutron flux transients that could result in reactor trip during power escalation.

(10)

APPLICABILITY

required

with

The intermediate range neutron flux channels shall be OPERABLE in MODE 2 and when any CONTROL ROD drive (CRD) trip breaker (13) in the closed position and the CRD System (13) capable of rod withdrawal.

in MODES 3, 4, and 5

(17)

The intermediate range instrumentation is designed to detect power changes during initial criticality and power escalation , e.g., when the power range and source range instrumentation cannot provide reliable indications. Since those conditions can exist in all of these MODES, the intermediate range instrumentation must be OPERABLE.

(27)

, or propagate from,

ACTIONS

~~A.1~~

~~If one intermediate range channel becomes inoperable when the channels indicate 1E-10 amp, the unit is exposed to the possibility that a single failure will disable all neutron monitoring instrumentation. To avoid this, the inoperable channel must be repaired or power must be reduced to the point where source range channels can provide neutron flux indication. Completion of Required Action A.1 places the unit in this state, and LCO 3.3.9, "Source Range Neutron Flux," requires OPERABILITY of two source range detectors once this state is reached. If the one channel failure occurs when indicated power is < 1E-10 amp, the Required Action prohibits increases in power above the source range capability.~~

~~The 2 hour Completion Time allows controlled reduction of power into the source range and is based on unit operating experience that demonstrates the improbability of the second intermediate range channel failing during the allowed interval.~~

A A

B.1 and B.2

(10)

the required

With two intermediate range neutron flux channels inoperable when THERMAL POWER is ≤ 5% RTP, the operators must place the

(continued)

BASES

ACTIONS

A.1 and A.2 (continued)

reactor in the next lowest condition for which the intermediate range instrumentation is not required. This involves providing power level indication on the source range instrumentation by immediately suspending operations involving positive reactivity changes and, within 1 hour, placing the reactor in the tripped condition with the CRD trip breakers open. The Completion Times are based on unit operating experience and allow the operators sufficient time to manually insert the CONTROL RODS prior to opening the CRD breakers.

SURVEILLANCE REQUIREMENTS

SR 3.3.10.1

edit

Performance of the CHANNEL CHECK once every 12 hours ~~ensures~~ provides reasonable assurance of prompt identification that a gross failure of instrumentation has not occurred. A CHANNEL CHECK is normally a comparison of the parameter indicated on one channel to ~~a similar~~ the same parameter on other channels. It is based on the assumption that instrument channels monitoring the same parameter should read approximately the same value. Significant deviations between the two instrument channels could be an indication of excessive instrument drift in one of the channels or of something even more serious. CHANNEL CHECK will detect gross channel failure; therefore, it is key in verifying that the instrumentation continues to operate properly between each CHANNEL CALIBRATION.

27

edit

Agreement criteria are determined by the unit staff based on a combination of factors including ~~the~~ channel instrument uncertainties. ~~including isolation, indication, and readability.~~ If a channel is outside the criteria, it may be an indication that ~~the transmitter or~~ the signal processing equipment has drifted outside its limit. If the channels are within the criteria, it is an indication that the channels are OPERABLE. Off scale low current loop channels are verified, where practical, to be reading at the bottom of the range and not failed low. The Frequency, about once every shift, is based on operating experience that demonstrates channel failure is rare. Since the probability of two random failures in redundant channels in any 12 hour period is extremely low, the CHANNEL CHECK minimizes the chance of loss of protective function due to

27

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.3.10.1 (continued)

failure of redundant channels. The CHANNEL CHECK supplements less formal, but more frequent, checks of channel OPERABILITY during normal operational use of the displays associated with the LCO's required channels. 10

with only one channel OPERABLE

When operating ~~in Required Action A.1~~, CHANNEL CHECK is still required. However, in this condition, a redundant intermediate range is not available for comparison. CHANNEL CHECK may still be performed via comparison with power or source range detectors, if available, and verification that the OPERABLE intermediate range channel is energized and indicates a value consistent with current unit status.

<INSERT B3.3-89A> 14

SR 3.3.10.~~2~~ 3

For intermediate range neutron flux channels, CHANNEL CALIBRATION is a complete check and readjustment of the channels, from the preamplifier input to the indicators. This test verifies the channel responds to a measured parameter within the necessary range and accuracy. CHANNEL CALIBRATION leaves the channel ~~adjusted to account~~ at a setpoint which accounts for instrument drift to ensure that the instrument channel remains operational between successive tests. 27

The SR is modified by a Note excluding neutron detectors from CHANNEL CALIBRATION. It is not necessary to test the detectors because generating a meaningful test signal is difficult. In addition, the detectors are of simple construction, and any failures in the detectors will be apparent as a change in channel output. The Frequency is based on operating experience and consistency with the typical industry refueling cycle and is justified by demonstrated instrument reliability over an ~~[18]~~ month interval such that the instrument is not adversely affected by drift.

SR 3.3.10.~~3~~ 4 15

SR 3.3.10.~~3~~ 4 is the verification ~~within 7 days prior to~~ performed each reactor startup of one decade of overlap with the ~~power~~ source range neutron flux instrumentation ~~prior to intermediate range indication exceeding 1E-6 amp~~. This ensures a

(continued)

<INSERT B3.3-89A>

SR 3.3.10.2

A CHANNEL FUNCTIONAL TEST, of the required intermediate range instrument channel, verifies proper operation of the channel each 31 days. Monthly testing provides reasonable assurance that the instrument channel will function, if required, to provide indication during MODE 2 and during unanticipated reactivity excursions from MODES 3, 4, or 5.

BASES

SURVEILLANCE REQUIREMENTS

SR 3.3.10.~~7~~ 4 (continued)

continuous source of power indication during the approach to criticality. Failure to perform this Surveillance leaves the unit in a condition where the intermediate range source channels provide adequate protection until the verification can be made. 15

The test may be omitted if performed within the previous 7 days based on operating experience, which shows that intermediate range instrument overlap does not change appreciably within this test interval. 15

REFERENCES

1. 10CFR 50.36

~~None.~~

EDIT

This Section Addresses the Following Specifications:

NUREG-1430	ANO-1 ITS	Title
3.3.5	3.3.5	Engineered Safety Feature Actuation System (ESFAS) Instrumentation
3.3.6	3.3.6	Engineered Safety Feature Actuation System (ESFAS) Manual Initiation
3.3.7	3.3.7	Engineered Safety Feature Actuation System (ESFAS) Automatic Actuation Logic

3.3 INSTRUMENTATION

3.3.5 Engineered Safeguards Actuation System (ESAS) Instrumentation

LCO 3.3.5 Three ESAS analog instrument channels for each Parameter in Table 3.3.5-1 shall be OPERABLE.

APPLICABILITY: According to Table 3.3.5-1.

ACTIONS

-----NOTE-----

Separate Condition entry is allowed for each Parameter.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more Parameters with one analog instrument channel inoperable.	A.1 Place analog instrument channel in trip.	1 hour
B. One or more Parameters with more than one analog instrument channel inoperable. <u>OR</u> Required Action and associated Completion Time not met.	B.1 Be in MODE 3. <u>AND</u> B.2 -----NOTE----- Only required for RCS Pressure - Low setpoint. ----- Reduce RCS pressure < 1750 psig. <u>AND</u>	6 hours 36 hours (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
B. (continued)	B.3 ------------NOTE------------ Only required for Reactor Building Pressure High setpoint and High High setpoint. -------------------------------- Be in MODE 5.	36 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.3.5.1	Perform CHANNEL CHECK.	12 hours
SR 3.3.5.2	Perform CHANNEL FUNCTIONAL TEST.	31 days
SR 3.3.5.3	Perform CHANNEL CALIBRATION.	18 months

Table 3.3.5-1 (page 1 of 1)
Engineered Safeguards Actuation System Instrumentation

	PARAMETER	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	ALLOWABLE VALUE
1.	Reactor Coolant System Pressure - Low Setpoint	≥ 1750 psig	≥ 1585 psig
2.	Reactor Building (RB) Pressure - High Setpoint	1,2,3,4	≤ 18.7 psia
3.	RB Pressure - High High Setpoint	1,2,3,4	≤ 44.7 psia

3.3 INSTRUMENTATION

3.3.6 Engineered Safeguards Actuation System (ESAS) Manual Initiation

LCO 3.3.6 Two manual initiation channels of each one of the ESAS Functions below shall be OPERABLE:

a. High Pressure Injection (channels 1 and 2);

b. Low Pressure Injection (channels 3 and 4);

c. Reactor Building (RB) Cooling (channels 5 and 6);

d. RB Spray (channels 7 and 8); and

e. Spray Additive (channels 9 and 10).

APPLICABILITY: MODES 1 and 2,
MODES 3 and 4 when associated engineered safeguards equipment is required to be OPERABLE.

ACTIONS

-----NOTE-----

Separate Condition entry is allowed for each Function.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more ESAS Functions with one channel inoperable.	A.1 Restore channel to OPERABLE status.	72 hours
B. Required Action and associated Completion Time not met.	B.1 Be in MODE 3. <u>AND</u> B.2 Be in MODE 5.	6 hours 36 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.3.6.1	Perform CHANNEL FUNCTIONAL TEST.	18 months

3.3 INSTRUMENTATION

3.3.7 Engineered Safeguards Actuation System (ESAS) Actuation Logic

LCO 3.3.7 The ESAS digital actuation logic channels shall be OPERABLE.

APPLICABILITY: MODES 1 and 2,
MODES 3 and 4 when associated engineered safeguards equipment is required to be OPERABLE.

ACTIONS

-----NOTE-----

Separate Condition entry is allowed for each digital actuation logic channel.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more digital actuation logic channels inoperable.	A.1 Place associated component(s) in engineered safeguards configuration.	1 hour
	OR	
	A.2 Declare the associated component(s) inoperable.	1 hour

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.3.7.1	Perform digital actuation logic CHANNEL FUNCTIONAL TEST.	31 days

B 3.3 INSTRUMENTATION

B 3.3.5 Engineered Safeguards Actuation System (ESAS) Instrumentation

BASES

BACKGROUND

The ESAS initiates necessary safety systems, based on the values of selected unit Parameters, to protect against violating core design limits and to mitigate accidents.

The ESAS operates in a distributed manner to initiate the appropriate systems. The ESAS does this by determining the need for actuation in each of three analog instrument channels monitoring each actuation Parameter. Once the need for actuation is determined, the condition is transmitted to digital actuation logic channels, which perform the two-out-of-three logic to determine the actuation of each end device.

Three Parameters are used for actuation:

- Low Reactor Coolant System (RCS) Pressure;
- High Reactor Building (RB) Pressure; and
- High High RB Pressure.

LCO 3.3.5 covers only the analog instrument channels that measure these Parameters. These channels include the equipment necessary to produce an actuation signal input to the digital actuation logic channels. This includes sensors, bistable devices, operational bypass circuitry, and logic buffer modules. LCO 3.3.6, "Engineered Safeguards Actuation System (ESAS) Manual Initiation," and LCO 3.3.7, "Engineered Safeguards Actuation System (ESAS) Actuation Logic," provide requirements on the manual initiation and digital actuation logic Functions.

The ESAS monitors three parameters via analog instrument channels. Each analog instrument channel provides input to the appropriate digital actuation logic channels that initiate equipment with a two-out-of-three coincidence logic on each digital channel. Each digital actuation logic channel includes bistable inputs from all three analog instrument channels of one parameter, i.e., either Low RCS Pressure, High RB Pressure, or High High RB Pressure. The digital actuation logic combines the analog instrument channel trips to actuate the individual Engineered Safeguards (ES) components needed to initiate each ES System. Figure 7.6, SAR, Chapter 7 (Ref. 1), also illustrates how analog instrument channel trips combine to cause digital actuation logic channel trips.

The ESAS is divided into five Functions actuated by ten digital actuation logic channels.

The ESAS High Pressure Injection (HPI) Function is actuated by ESAS digital actuation logic channels 1 and 2 and includes the following system actuations: HPI, a subset of RB isolation valves, diesel generators (DGs), and ES electrical alignment. Digital actuation logic channels 1 and 2 are actuated by two-out-of-three RCS Pressure—Low analog instrument channels, or two-out-of-three RB Pressure—High analog instrument channels.

The ESAS Low Pressure Injection (LPI) Function is actuated by ESAS digital actuation logic channels 3 and 4 and includes the following system actuations: LPI, a subset of RB isolation valves, and emergency feedwater (EFW) through an ESAS signal provided to the Emergency Feedwater Initiation and Control (EFIC) Instrumentation System. Digital actuation logic channels 3 and 4 are actuated by two-out-of-three RCS Pressure—Low analog instrument channels, or two-out-of-three RB Pressure—High analog instrument channels.

The ESAS RB Cooling Function is actuated by ESAS digital actuation logic channels 5 and 6 and includes the following system actuations: RB cooling, a subset of RB isolation valves, and RB penetration room ventilation system. Digital actuation logic channels 5 and 6 are actuated by two-out-of-three RB Pressure—High analog instrument channels.

The ESAS RB Spray Function is actuated by ESAS digital actuation logic channels 7 and 8 and includes the following system actuations: RB spray. Digital actuation logic channels 7 and 8 are actuated by two-out-of-three RB Pressure—High High analog instrument channels.

The ESAS Spray Additive Function is actuated by ESAS digital actuation logic channels 9 and 10 and includes the following system actuations: spray additive. Digital actuation logic channels 9 and 10 are actuated by two-out-of-three RB Pressure—High High analog instrument channels.

The following matrix identifies the ESAS digital actuation logic channels and the systems actuated by each Parameter.

ESAS Digital Actuation Logic Channels	Actuated Systems	Parameter		
		RCS Press. Low	RB Press. High	RB Press. High High
1 and 2	Subset of RB Isolation, ES Electrical Alignment, HPI, and DG Start	X	X	
3 and 4	Subset of RB Isolation, LPI, and EFIC EFW	X	X	
5 and 6	Subset of RB Isolation, RB Cooling, and Penetration Room Ventilation		X	
7 and 8	RB Spray			X
9 and 10	Spray Additive			X

The ES equipment is divided between the two redundant actuation trains. The division of the equipment between the two actuation trains is based on the equipment redundancy and function and is accomplished in such a manner that the failure of one of the digital actuation logic channels and the related safeguards equipment will not inhibit the overall ES Functions. Redundant ES pumps are controlled from separate and independent digital actuation logic channels.

The actuation of ES equipment is also available by manual actuation switches located on the control room console.

The ESAS, in conjunction with the actuated equipment, provides protective functions necessary to mitigate Design Basis Accidents (DBAs), specifically the loss of coolant accident (LOCA), RB DBA, and as a backup to mitigate the steam line break (SLB) event (Ref. 2). The ESAS relies on the OPERABILITY of the digital actuation logic channels to perform the actuation of the selected systems.

Engineered Safeguards Actuation System Bypasses

No provisions are made for maintenance bypass of ESAS instrumentation channels. Operational bypass of certain channels is necessary to allow accident recovery actions to continue and, for some channels, to allow reactor shutdown without ESAS actuation.

The ESAS RCS pressure analog instrument channels include permissive bistables that allow manual bypass when reactor pressure is below the point at which the low pressure trip is required to be OPERABLE. Once permissive conditions are sensed, the RCS pressure trips may be manually bypassed. Bypasses are automatically removed when bypass permissive conditions are exceeded.

This bypass provides an operational provision only outside the Applicability for this Parameter, and provides no safety function.

Reactor Coolant System Pressure

The RCS pressure is monitored by three independent pressure transmitters located in the RB. These transmitters are separate from the transmitters that feed the Reactor Protection System (RPS). Each of the pressure signals generated by these transmitters is monitored by two bistables to provide a trip signal at ≥ 1585 psig and a bypass permissive signal at ≤ 1750 psig.

The outputs of the three low RCS pressure trip bistables drive relays in two sets of identical and independent digital instrument channels. These two sets of channels each use two-out-of-three coincidence digital logic for actuation.

Each analog channel can be tested online to verify that the signal and trip setpoint are within the specified allowance requirements of approved calibration procedures. The built-in test facilities permit an electrical trip test of each analog instrument string by the substitution of signals at the buffer amplifiers. When an analog instrument string is placed in test, all associated analog subsystem outputs go to the trip state. This assures that all protective action cannot be defeated by placing analog instrument strings in test.

Reactor Building Pressure

The RB pressure is monitored by three independent pressure transmitters located inside the RB. These transmitters are separate from the transmitters that feed the Reactor Protection System (RPS). Each of the pressure signals generated by these transmitters is monitored by two bistables to provide trip signals. The outputs of the bistables, associated with the RB Pressure—High and RB Pressure—High High trips, drive relays in two sets of identical and independent digital instrument channels. These two sets of channels each use two-out-of-three coincidence digital logic for automatic actuation.

Each channel can be tested online to verify that the signal and trip setpoint are within the specified allowance requirements of approved calibration procedures. The built-in test facilities permit an electrical trip test of each analog instrument string by the substitution of signals at the buffer amplifiers. When an analog instrument string is placed in test, all associated analog subsystem outputs go to the trip state. This assures that all protective action cannot be defeated by placing analog instrument strings in test.

APPLICABLE SAFETY ANALYSES

The following ESAS Functions have been assumed within the accident analyses.

High Pressure Injection

The ESAS actuation of HPI has been assumed for core cooling in the LOCA analysis and is available for boron addition in the SLB analysis.

Low Pressure Injection

The ESAS actuation of LPI has been assumed for large break LOCAs.

Reactor Building Spray, Reactor Building Cooling, and Reactor Building Isolation

The ESAS actuation of the RB coolers and RB Spray have been credited in the RB analysis for LOCAs. Accident dose calculations have credited RB Penetration Room Ventilation, RB Isolation, and RB Spray. The MSLB analysis also credits ESAS actuation of the RB Cooling and RB Spray.

Emergency Power

The ESAS initiated DG Start and ES electrical equipment alignment have been included in the design to ensure that emergency power is available throughout the limiting LOCA scenarios.

The small and large break LOCA analyses (Ref. 2) assume a conservative delay time for the actuation of HPI and LPI. This delay time includes allowances for DG starting, DG loading, Emergency Core Cooling Systems (ECCS) pump starts, and valve alignment. Similarly, the RB Cooling, RB Isolation, and RB Spray have been analyzed with delays appropriate for the entire system analyzed.

Accident analyses rely on automatic ESAS actuation for protection of the core temperature and containment pressure limits and for limiting off site dose levels following an accident. These include LOCA, SLB, and other events that result in RCS inventory reduction or severe loss of RCS cooling.

The ESAS instrumentation satisfies Criterion 3 of 10 CFR 50.36 (Ref. 3) for operation in MODE 1. There are no specific safety analyses for operation in MODES 2, 3 and 4. However, industry operating experience has identified the ESAS instrumentation as significant to public health and safety during these operating conditions. Therefore, the ESAS instrumentation satisfies Criterion 4 of 10 CFR 50.36 for operation in MODES 2, 3 and 4.

LCO

The LCO requires three ESAS analog instrument channels for each Parameter in Table 3.3.5-1 to be OPERABLE. Failure of any instrument renders the affected analog instrument channel(s) inoperable and reduces the reliability of the affected Functions.

Only the Allowable Value is specified for each ESAS Function in the LCO. Trip setpoints are provided in the calibration procedures. The trip setpoints are selected to ensure the setpoints measured by CHANNEL FUNCTIONAL TESTS do not exceed the Allowable Value if the bistable is performing as required. Each Allowable Value specified is equal to or more conservative than any analytical limit assumed in the safety analysis to account for instrument uncertainties appropriate to the trip Parameter. Guidance used to calculate the uncertainties associated with the trip setpoints is contained in Instrument Loop Error Analysis Setpoint Methodology, Design Guide, IDG-001 (Ref. 3).

The values for bypass removal functions are stated in the Applicable MODES or Other Specified Condition column of Table 3.3.5-1.

Three ESAS analog instrument channels shall be OPERABLE to ensure that a single failure in one channel will not result in loss of the ability to automatically actuate the required safety systems.

Reactor Coolant System Pressure

Three channels of RCS Pressure - Low are required OPERABLE. Each channel includes a sensor, trip bistable, bypass bistable, bypass relays, and output relays. Failure of a bypass bistable or bypass circuitry, such that an analog instrument channel cannot be bypassed, does not render the channel inoperable since the channel is still capable of performing its safety function, i.e., this is not a safety related bypass function.

The trip setpoints are the nominal values at which the bistables are set. For the RCS Pressure—Low, the limiting safety analysis assumes the HPI, LPI, EFIC EFW, ES electrical alignment, and two subsets of RB isolation actuate at ≥ 1520 psig (≥ 1535 psia). The Allowable Value of ≥ 1585 psig includes considerations for instrumentation error and an allowance for margin. Allowances for instrument drift and additional margin are included in the trip setpoint.

Guidance used to calculate the uncertainties associated with the trip setpoints is provided in Instrument Loop Error Analysis and Setpoint Methodology Manual, Design Guide, IDG-001 (Ref. 4). The explicit uncertainties associated with each setpoint are addressed in the individual design calculations or calibration procedures. Setpoints in accordance with the Allowable Value in conjunction with the LCOs and administrative controls ensure that the consequences of DBAs will be acceptable, providing the unit is operated from within the LCOs at the onset of the DBA and the equipment functions as analyzed. An analog instrument channel is inoperable if its actual trip setpoint is not within its required Allowable Value.

Reactor Building Pressure

Three channels of RB Pressure—High and RB Pressure—High High are required to be OPERABLE. Each channel includes a pressure switch, bypass relays, and output relays.

The trip setpoints are the nominal values at which the bistables are set. Credit is taken in the safety analyses for RB Pressure—High trip for the actuation of selected systems. The safety analyses for reactor building performance and equipment environmental qualification (pressure and temperature envelope definition) conservatively assume the RB cooling is not initiated until well beyond the expected actual automatic actuation time frame. Therefore, no additional consideration of the instrumentation uncertainties is warranted.

Credit is taken in the safety analyses for RB Pressure—High High trip for the actuation of selected systems. The safety analyses for reactor building performance and equipment environmental qualification (pressure and temperature envelope definition) conservatively assumes the RB spray is not initiated until well beyond the expected actual automatic actuation time frame. Therefore, no additional consideration of the instrumentation uncertainties is warranted.

Therefore, the bistable is considered to be properly adjusted when the "as left" value is consistent with the identified Allowable Value, i.e., for this parameter the trip setpoint and the Allowable Value are the same. Guidance used to calculate the uncertainties associated with the trip setpoints is provided in Instrument Loop Error Analysis and Setpoint Methodology Manual, Design Guide, IDG-001 (Ref. 4). Setpoints in accordance with the Allowable Value ensure that the consequences of DBAs will be acceptable, providing the unit is operated from within the LCOs at the onset of the DBA and the equipment functions as analyzed. An analog instrument channel is inoperable if its actual trip setpoint is not within its required Allowable Value.

APPLICABILITY

Three ESAS analog instrument channels for each of the following Parameters shall be OPERABLE.

1. Reactor Coolant System Pressure - Low Setpoint

The RCS Pressure - Low Setpoint actuation Parameter shall be OPERABLE during operation at or above 1750 psig. This requirement ensures the capability to automatically actuate safety systems and components during conditions indicative of a LOCA or secondary unit overcooling. Below 1750 psig, the low RCS Pressure actuation Parameter can be bypassed to avoid actuation during normal unit cooldowns when safety systems actuations are not required.

The allowance for the bypass is consistent with the transition of the unit to a lower energy state, where there is more margin to safety limits. The unit response to any event, given that the reactor is already tripped, will be less severe and allows more time for operator action to provide manual safety system actuations than in higher energy states. This is even more appropriate during unit heatups when the primary system and core energy content is low, prior to power operation.

In MODES 5 and 6, there is more time for the operator to evaluate unit conditions and respond by manually starting individual systems, pumps, and other equipment to mitigate the consequences of an abnormal condition or accident than in higher MODES. RCS pressure and temperature are very low, and many ES components are administratively controlled or otherwise prevented from actuating to prevent inadvertent overpressurization of unit systems.

2., 3. Reactor Building Pressure - High Setpoint and Reactor Building Pressure - High High Setpoint

The RB Pressure - High and RB Pressure - High High actuation Functions of ESAS shall be OPERABLE in MODES 1, 2, 3, and 4 when the potential for a HELB exists. In MODES 5 and 6, there is more time for the operator to evaluate unit conditions and respond by manually starting individual systems, pumps, and other equipment to mitigate the consequences of an abnormal condition or accident than in higher MODES. Plant pressure and temperature are very low and many ES components are administratively controlled or otherwise prevented from actuating to prevent inadvertent overpressurization of unit systems.

ACTIONS

Required Actions A and B apply to the ESAS instrumentation Parameters listed in Table 3.3.5-1.

A Note has been added to the ACTIONS indicating separate Condition entry is allowed for each Parameter.

If an analog instrument channel's trip setpoint is found nonconservative with respect to the Allowable Value, or the transmitter, instrument loop, signal processing electronics, or ESAS bistable is found inoperable, then all affected functions provided by that channel should be declared inoperable and the unit must enter the Conditions for the particular protection Parameter affected.

A.1

Condition A applies when one analog instrument channel becomes inoperable in one or more Parameters. If one ESAS analog instrument channel is inoperable, placing it in a tripped condition leaves the system in a one-out-of-two condition for actuation. Thus, if another analog instrument channel were to fail, the ESAS instrumentation could still perform its actuation functions. This action is completed when all of the affected output relays are tripped. This can normally be accomplished by tripping the affected bistables.

The 1 hour Completion Time is sufficient time to perform the Required Action.

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

Condition B applies when Required Action A.1 and its associated Completion Time are not met, or when one or more parameters have more than one analog instrument channel inoperable. If Condition B applies, the unit must be brought to a condition in which the LCO does not apply. To achieve this status, the unit must be brought to at least MODE 3 within 6 hours. Additionally, for the RCS Pressure—Low parameter, the unit must be brought to < 1750 psig within 36 hours, and for the RB Pressure—High and High High parameters, the unit must be brought 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.

SURVEILLANCE REQUIREMENTS

The ESAS Parameters listed in Table 3.3.5-1 are subject to CHANNEL CHECK, CHANNEL FUNCTIONAL TEST, and CHANNEL CALIBRATION.

<u>SR 3.3.5.1</u>

Performance of the CHANNEL CHECK every 12 hours provides reasonable assurance for prompt identification of a gross failure of instrumentation. A CHANNEL CHECK is normally a comparison of the parameter indicated on one channel to a similar parameter on other channels. It is based on the assumption that instrument channels monitoring the same parameter should read approximately the same value. Significant deviations between the two instrument channels could be an indication of excessive instrument drift in one of the channels or of something even more serious. CHANNEL CHECK will detect gross channel failure; therefore, it is key in verifying that the instrumentation continues to operate properly between CHANNEL CALIBRATIONs.

Agreement criteria are determined by the unit staff, based on a combination of factors including channel instrument uncertainties. If a channel is outside the criteria, it may be an indication that the transmitter or the signal processing equipment has drifted outside its limit.

The Frequency is based on operating experience that demonstrates channel failure is rare. Since the probability of two random failures in redundant channels in any 12 hour period is extremely low, the CHANNEL CHECK minimizes the chance of loss of protective function due to failure of redundant channels. The CHANNEL CHECK supplements less formal, but potentially more frequent, checks of channel OPERABILITY during normal operational use of the displays associated with the LCO's required channels.

SR 3.3.5.2

A CHANNEL FUNCTIONAL TEST is performed on each required ESAS analog instrument channel to ensure the entire channel will perform the intended functions. Any setpoint adjustment shall be consistent with the assumptions of the setpoint calculations.

The Frequency of 31 days is based on unit operating experience, with regard to channel OPERABILITY and drift, which demonstrates that failure of more than one channel of a given function in any 31 day interval is a rare event.

SR 3.3.5.3

CHANNEL CALIBRATION is a complete check of the analog instrument channel, including the sensor. The test verifies that the channel responds to a measured parameter within the necessary range and accuracy. CHANNEL CALIBRATION leaves the channel adjusted to account for instrument drift to ensure that the analog instrument channel remains OPERABLE between successive tests. CHANNEL CALIBRATION shall find that measurement errors and bistable setpoint errors are within the assumptions of the setpoint calculations. CHANNEL CALIBRATIONS must be performed consistent with the assumptions of the setpoint calculations.

This Frequency is justified by the assumption of at least an 18 month calibration interval to determine the magnitude of equipment drift in the setpoint calculations.

REFERENCES

1. SAR, Chapter 7.
2. SAR, Chapter 14 and Chapter 3A.
3. 10 CFR 50.36.
4. Instrument Loop Error Analysis and Setpoint Methodology Manual, Design Guide, IDG-001.

B 3.3 INSTRUMENTATION

B 3.3.6 Engineered Safeguards Actuation System (ESAS) Manual Initiation

BASES

BACKGROUND

The ESAS manual initiation capability allows the operator to actuate ESAS Functions from the control room in the absence of any other initiation condition. This ESAS manual initiation capability is provided in the event the operator determines that an ESAS Function is needed and has not been automatically actuated. Furthermore, the ESAS manual initiation capability allows operators to rapidly initiate Engineered Safeguards (ES) Functions if the trend of unit parameters indicates that ES actuation will be needed.

LCO 3.3.6 covers only the system level manual initiation of these Functions. LCO 3.3.5, "Engineered Safeguards Actuation System (ESAS) Instrumentation," and LCO 3.3.7, "Engineered Safeguards Actuation System (ESAS) Actuation Logic," provide requirements on the portions of the ESAS that automatically initiate the Functions described earlier.

The ESAS manual initiation Function relies on the OPERABILITY of the digital actuation logic channels (LCO 3.3.7) to perform the actuation of the systems. A manual trip push button is provided on a control room console for each of the digital actuation logic channels. Operation of the push button energizes relays whose contacts perform a logical "<u>OR</u>" function with the automatic actuation.

The ESAS manual initiation channel is defined as the console switch and the instrumentation from the console switch to, but not including, the digital actuation logic channels, which actuate the end devices. Other means of manual initiation, such as controls for individual ES devices, may be available in the control room and other unit locations. These alternative means are not required by this LCO, nor may they be credited to fulfill the requirements of this LCO.

APPLICABLE SAFETY ANALYSES

The ESAS, in conjunction with the actuated equipment, provides protective functions necessary to mitigate Design Basis Accidents, specifically, the loss of coolant accident (LOCA), RB DBA and as a backup to mitigate the steam line break event.

The ESAS manual initiation ensures that the control room operator can rapidly initiate ES Functions. The manual initiation trip Function is required as a backup to automatic trip functions and allows operators to initiate ESAS whenever any parameter is rapidly trending toward its trip setpoint.

Operating experience has shown the ESAS manual initiation function to be significant to public health and safety, and therefore satisfy Criterion 4 of 10 CFR 50.36 (Ref. 1).

LCO

Two ESAS manual initiation channels of each ESAS Function shall be OPERABLE whenever conditions exist that could require ES protection of the reactor or RB. Two OPERABLE channels ensure that no single random failure will prevent system level manual initiation of any ESAS Function. The ESAS manual initiation Function allows the operator to initiate protective action prior to automatic initiation or in the event the automatic initiation does not occur.

The ESAS is divided into five Functions actuated by ten manual initiation channels as indicated in the following table:

Function	Associated Channels
High Pressure Injection	1 & 2
Low Pressure Injection	3 & 4
RB Cooling	5 & 6
RB Spray	7 & 8
Spray Additive	9 & 10

The ESAS High Pressure Injection (HPI) Function is actuated by ESAS Manual Initiation channels 1 and 2 and includes the following system actuations: HPI, a subset of reactor building (RB) isolation valves, diesel generators, and ES electrical alignment.

The ESAS Low Pressure Injection (LPI) Function is actuated by ESAS Manual Initiation channels 3 and 4 and includes the following system actuations: LPI, a subset of RB isolation valves, and emergency feedwater (EFW) through an ESAS signal provided to the Emergency Feedwater Isolation and Control (EFIC) System.

The ESAS RB Cooling Function is actuated by ESAS Manual Initiation channels 5 and 6 and includes the following system actuations: RB cooling, a subset of RB isolation valves, and RB penetration room ventilation system.

The ESAS RB Spray Function is actuated by ESAS Manual Initiation channels 7 and 8 and includes the following system actuations: RB spray.

The ESAS Spray Additive Function is actuated by ESAS Manual Initiation channels 9 and 10 and includes the following system actuations: spray additive.

APPLICABILITY

The ESAS manual initiation Functions shall be OPERABLE in MODES 1 and 2, and in MODES 3 and 4 when the associated ES equipment is required to be

OPERABLE. The manual initiation channels are required because ES Functions are designed to provide protection in these MODES. ESAS initiates systems that are either reconfigured for decay heat removal operation or disabled while in MODES 5 and 6. Accidents in these MODES are slow to develop and would be mitigated by manual operation of individual components. Time is available to evaluate unit conditions and to respond by manually operating the ES components, if required.

ACTIONS

A Note has been added to the ACTIONS indicating separate Condition entry is allowed for each ESAS manual initiation Function.

A.1

Condition A applies when one manual initiation channel of one or more ESAS Functions becomes inoperable. Required Action A.1 must be taken to restore the channel to OPERABLE status within the next 72 hours. The Completion Time of 72 hours is based on unit operating experience and administrative controls, which provide alternative means of ESAS Function initiation via individual component controls. The 72 hour Completion Time is generally consistent with the allowed outage time for the safety systems actuated by ESAS.

B.1 and B.2

If Required Action A.1 and the 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 MODES from full power conditions in an orderly manner and without challenging unit systems.

SURVEILLANCE REQUIREMENTS

SR 3.3.6.1

This SR requires the performance of a CHANNEL FUNCTIONAL TEST of the ESAS manual initiation. This test verifies that the initiating circuitry is OPERABLE and will actuate the digital actuation logic channels. The 18 month Frequency is based on the need to perform this Surveillance under the conditions that apply during a unit outage and the potential for an unplanned transient if the Surveillance were performed with the reactor at power. This Frequency is demonstrated to be sufficient, based on operating experience, which shows these components usually pass the Surveillance when performed on the 18 month Frequency.

REFERENCES

1. 10 CFR 50.36.

B 3.3 INSTRUMENTATION

B 3.3.7 Engineered Safeguards Actuation System (ESAS) Actuation Logic

BASES

BACKGROUND

The digital actuation logic channels of ESAS are defined as the instrumentation between, but not including, the buffers of the analog instrument channels and the unit controls that actuate ESAS equipment. Each of the components actuated by the ESAS Functions is associated with one or more digital actuation logic channels. If two-out-of-three ESAS analog instrument channels indicate a trip, or if channel level manual initiation occurs, the digital actuation logic channel is activated and the associated equipment is actuated.

The purpose of requiring OPERABILITY of the ESAS digital actuation logic channels is to ensure that the Functions of the ESAS can be automatically initiated in the event of an accident. Automatic actuation of some Functions is necessary to prevent the unit from exceeding the Emergency Core Cooling Systems (ECCS) limits in 10 CFR 50.46 (Ref. 1). It should be noted that OPERABLE digital actuation logic channels alone will not ensure that each Function can be activated; the analog instrument channels and actuated equipment associated with each Function must also be OPERABLE to ensure that the Functions can be automatically initiated during an accident.

LCO 3.3.7 covers only the digital actuation logic channels that initiate these Functions. LCO 3.3.5, "Engineered Safeguards Actuation System (ESAS) Instrumentation," and LCO 3.3.6, "Engineered Safeguards Actuation System (ESAS) Manual Initiation," provide requirements on the analog instrument and manual initiation channels that input to the digital actuation logic channels.

The ESAS, in conjunction with the actuated equipment, provides protective functions necessary to mitigate Design Basis Accidents (DBAs), specifically, the loss of coolant accident (LOCA) and steam line break (SLB) events. The ESAS relies on the OPERABILITY of the digital actuation logic for each component to perform the actuation of the selected systems.

The small and large break LOCA analyses assume a conservative delay time for the actuation of high pressure injection (HPI) and low pressure injection (LPI) in BAW-10103A, Rev. 3 (Ref. 2). This delay time includes allowances for diesel generator (DG) starts, DG loading, ECCS pump starts, and valve alignment. Similarly, the reactor building (RB) Cooling, RB Isolation, and RB Spray have been analyzed with delays appropriate for the entire system.

The ESAS automatic initiation of Engineered Safeguards (ES) Functions to mitigate accident conditions is assumed in the DBA analysis and is required to ensure that consequences of analyzed events do not exceed the accident analysis predictions.

Automatically actuated features include HPI, LPI, RB Cooling, RB Spray, RB Spray Additive, and RB Isolation.

APPLICABLE SAFETY ANALYSES

Accident analyses rely on automatic ESAS actuation for protection of the core and RB and for limiting off site dose levels following an accident. The digital actuation logic is an integral part of the ESAS.

The ESAS actuation logic satisfies Criterion 3 of 10 CFR 50.36 (Ref. 3) for operation in MODE 1. There are no specific safety analyses for operation in MODES 2, 3 and 4. However, industry operating experience has identified the ESAS actuation logic as significant to public health and safety during these operating conditions. Therefore, the ESAS actuation logic satisfies Criterion 4 of 10 CFR 50.36 for operation in MODES 2, 3 and 4.

LCO

The digital actuation logic channels are required to be OPERABLE whenever conditions exist that could require ES protection of the reactor or the RB. This ensures automatic initiation of the ES required to mitigate the consequences of accidents.

The ESAS is divided into five Functions actuated by ten digital actuation logic channels as indicated in the following table:

Function	Associated Channels
High Pressure Injection	1 & 2
Low Pressure Injection	3 & 4
RB Cooling	5 & 6
RB Spray	7 & 8
Spray Additive	9 & 10

The ESAS HPI Function is actuated by ESAS digital actuation logic channels 1 and 2 and includes the following system actuations: HPI, a subset of RB isolation valves, DGs, and ES electrical alignment. Digital actuation logic channels 1 and 2 are actuated by two-out-of-three RCS Pressure—Low analog instrument channels, or two-out-of-three RB Pressure—High analog instrument channels.

The ESAS LPI Function is actuated by ESAS digital actuation logic channels 3 and 4 and includes the following system actuations: LPI, a subset of RB isolation valves, and EFW through an ESAS signal provided to EFIC. Digital actuation logic channels 3 and 4 are actuated by two-out-of-three RCS Pressure—Low analog instrument channels, or two-out-of-three RB Pressure—High analog instrument channels.

The ESAS RB Isolation and Cooling Function is actuated by ESAS digital actuation logic channels 5 and 6 and includes the following system actuations: RB cooling, a subset of RB isolation valves, and RB penetration room ventilation system. Digital actuation logic channels 5 and 6 are actuated by two-out-of-three RB Pressure—High analog instrument channels.

The ESAS RB Spray Function is actuated by ESAS digital actuation logic channels 7 and 8 and includes the following system actuations: RB spray. Digital actuation logic channels 7 and 8 are actuated by two-out-of-three RB Pressure—High High analog instrument channels.

The ESAS Spray Additive Function is actuated by ESAS digital actuation logic channels 9 and 10 and includes the following system actuations: spray additive. Digital actuation logic channels 9 and 10 are actuated by two-out-of-three RB Pressure—High High analog instrument channels.

APPLICABILITY

The digital actuation logic channels shall be OPERABLE in MODES 1 and 2, and in MODES 3 and 4 when the associated ES equipment is required to be OPERABLE, because ES Functions are designed to provide protection in these MODES. Automatic actuation in MODE 5 or 6 is not required because the systems initiated by the ESAS are either reconfigured for decay heat removal operation or disabled. Accidents in these MODES are slow to develop and would be mitigated by manual operation of individual components. Time is available to evaluate unit conditions and respond by manually operating the ES components, if required.

ACTIONS

A Note has been added to the ACTIONS indicating separate Condition entry is allowed for each ESAS digital actuation logic channel.

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

When one or more digital actuation logic channel(s) are inoperable, the associated component(s) can be placed in their ES configuration. Required Action A.1 is equivalent to the digital actuation logic channel performing its safety function ahead of time. In some cases, placing the component in its ES configuration would violate unit safety or operational considerations. In these cases, the component status should not be changed, but the supported system component must be declared inoperable. Conditions which would preclude the placing of a component in its ES configuration include, but are not limited to, violation of system separation, activation of fluid systems that could lead to thermal shock, isolation of fluid systems that are normally functioning, and actuation of components which would not return to their actuated condition upon restoration of electrical power. The

Completion Time of 1 hour is based on operating experience and reflects the urgency associated with the inoperability of a safety system component.

Required Action A.2 requires entry into the Required Actions of the affected supported systems, since the true effect of digital actuation logic channel failure is inoperability of the supported system. The Completion Time of 1 hour is based on operating experience and reflects the urgency associated with the inoperability of a safety system component. A combination of Required Actions A.1 and A.2 may be used for different components associated with an inoperable ESAS digital actuation logic channel.

SURVEILLANCE REQUIREMENTS

SR 3.3.7.1

SR 3.3.7.1 is the performance of a CHANNEL FUNCTIONAL TEST on a 31 day Frequency. The test demonstrates that each digital actuation logic channel successfully performs the two-out-of-three logic combinations every 31 days. The test simulates the required one-out-of-three inputs to the logic circuit and verifies the successful operation of the digital actuation logic. The Frequency is based on operating experience that demonstrates the rarity of more than one channel failing within the same 31 day interval.

REFERENCES

1. 10 CFR 50.46.
2. BAW-10103A, Rev. 3, July 1977.
3. 10 CFR 50.36.

CTS DISCUSSION OF CHANGES

ITS Section 3.3B: Instrumentation - ESAS

Note: ITS Section 3.3B package includes the following ITS:

ITS 3.3.5	Engineered Safeguards Actuation System (ESAS) Instrumentation
ITS 3.3.6	ESAS Manual Initiation
ITS 3.3.7	ESAS Actuation Logic

which address the corresponding NUREG-1430 LCOs.

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 B&W Standard Technical Specification (RSTS), NUREG-1430, Revision 1. This change does not alter the requirements of the CTS or NUREG. 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 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 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.

CTS DISCUSSION OF CHANGES

A4 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.)

A5 The Notes which allow for separate entry into the ACTIONS of ITS 3.3.5, ITS 3.3.6, and ITS 3.3.7 have been adopted. These additions have been made to provide requirements in a format consistent with NUREG-1430. The addition of these Notes maintains allowances consistent with the use and application of the requirements of the corresponding portions of CTS Table 3.5.1-1. This change represents a change in presentation format only with no addition or deletion of requirements.

A6 Requirements for instrument channels presented in CTS Table 3.5.1-1 have been replaced by the requirements of ITS 3.3.5. This change maintains the requirement for three OPERABLE channels of instrumentation for each of the required parameters. It does represent a change in format for these requirements. However, no additional requirements have been added by this change and no current requirements have been deleted.

A7 The term Minimum Degree of Redundancy as presented in CTS, i.e., Table 3.5.1-1 Column 4, will not be retained in ITS. Omission of this term is not considered to result in any changes in requirements since the intent of this column is consistent with application of Table 3.5.1-1 Column 3, "Minimum Channels Operable," which is retained (although the format is changed per DOC A3). Removal of this term and its usage from the CTS does not represent any actual change in requirements, only a change in presentation.

CTS DISCUSSION OF CHANGES

A8 The CTS requirements for the ESAS manual trip pushbuttons found in CTS Table 3.5.1-1 have been replaced by the requirements of ITS 3.3.6. This change maintains the requirement for two OPERABLE channels of manual actuation instrumentation for each of the required Functions. It does however represent a change in format for these requirements, although no additional requirements have been added by this change and no current requirements have been deleted.

A9 CTS Table 3.5.1-1, Engineered Safeguards Actuation System (ESAS), Functional Units 1, 2, 3, 4, and 5 have been replaced by ITS LCO 3.3.7. Although the CTS does not clearly present these requirements as an LCO, the requirements of these portions of Table 3.5.1-1 are treated as such by ANO-1. The adoption of ITS LCO 3.3.7 represents a change in format. However, this change in format does not change the application of the requirements found in CTS as they relate to the ESAS Actuation Logic Channels.

A10 The requirement to test the ESAS Manual Trip Functions Logic on a monthly basis will no longer be individually specified as it is in CTS Table 4.1-1 Item 43 b. This CTS requirement is redundant to the testing requirements presented in CTS Table 4.1-1 Items 14, 16, 18, and 20. The design of the ESAS at ANO-1 is such that performance of the CHANNEL FUNCTIONAL TEST of the Actuation Logic Channels encompasses the manual actuation system logic test specified in CTS Table 4.1-1 Item 43 b. Testing of the ESAS Actuation Logic Channels, as required by ITS SR 3.3.7.1, will maintain the testing requirements consistent with CTS.

A11 The requirement to perform a CHANNEL CHECK on the reactor building (RB) pressure high-high instrument channels (Reactor Building Spray System Analog Channels, Reactor Building Pressure Channels) has been indicated as an addition to the CTS in Table 4.1-1 Item 21.a. Although this is a change in presentation it does not represent a change in requirements. The design of the ANO-1 ESAS instrument channels is such that the same three transmitters provide input to both the High and the High-High RB pressure functions. Because the indications available for the performance of the required CHANNEL CHECKS are shared by both the High and the High-High RB pressure functions, one performance of this check is sufficient for both functions. The additional CHANNEL CHECK requirement was indicated in the CTS to provide a more complete cross-reference to the ITS requirements. This change provides requirements consistent with NUREG-1430 both in presentation and in content.

A12 The allowance provided in CTS 3.5.3 to bypass the High Reactor Building Pressure and Low Reactor Coolant System Pressure Functional Units during reactor building leak rate tests is omitted. The revised Applicabilities for these Functions (see DOC L1) do not require them to be OPERABLE during the leak rate testing. Therefore, this change is considered administrative.

A13 This page is not yet approved as provided in this package. Therefore, the markup is dependent on the expected NRC approval of the August 18, 1999 (Ref. 1CAN089903) license amendment request (LAR) related to the ESAS RCS low pressure setpoint revision.

A14 CTS 4.1.c is omitted since it duplicates requirements provided in the regulations, i.e., 10 CFR 50, Appendix B, criteria XI, XVI, and XVII. 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. Further, changes to the requirements are controlled by the NRC. This change is consistent with NUREG-1430.

TECHNICAL CHANGE – MORE RESTRICTIVE

M1 Not used.

M2 CTS Table 3.5.1-1 Note 8 provides action requirements in the event any portion of an ESAS digital subsystem is inoperable. This action requirement is referenced from CTS Table 3.5.1-1 although no specific LCO requirement is provided. CTS Table 3.5.1-1 Note 8 indicates that the safety features associated with an inoperable ESAS digital subsystem are to be considered inoperable and that CTS 3.3 applies. It does not however, specify a Completion Time for this action requirement. ITS 3.3.7 Required Action A.2 and its associated Completion Time are adopted to replace the requirements of CTS 3.5.1-1 Note 8. The adoption of the 1 hour Completion Time provides more restrictive, but appropriate, requirements in that no time period for the performance of this action was specified in CTS. This change is consistent with NUREG-1430.

M3 CTS Table 3.5.1-1 Note 6 provides an allowance for continued operation by tripping an inoperable channel and reducing the 2 out of 3 logic to 1 out of the remaining 2 channels. However, no time is specified to complete this action. Therefore, Note 1 is applicable until the inoperable channel is tripped. Note 1 requires the unit to be in hot shutdown within 12 hours. Therefore, the unit essentially has 12 hours to trip the inoperable channel (and restore compliance) or be in MODE 3. ITS 3.3.5 Required Actions A.1 and B.1 will provide only one hour to trip the channel or be in MODE 3 within an additional 6 hours (see also DOC M5). This change represents more restrictive requirements in that ITS 3.3.5 Required Actions A.1 and B.1 specify 7 hours before the unit must be in MODE 3 where CTS allows 12 hours (if the channel is not placed in the tripped condition). Further, the 1 hour Completion Time to place the channel in a tripped condition is not specified in CTS and also represents a more restrictive requirement. This change provides an appropriate Completion Time for this Required Action consistent with NUREG-1430.

M4 CTS Table 3.5.1-1 Note 5 has been replaced by ITS 3.3.5 Required Action B.2.2 and ITS 3.3.6 Required Action B.2. CTS Table 3.5.1-1 Note 5, in conjunction with

CTS Table 3.5.1-1 Note 1, provides a total time of 84 hours, from failure to meet the LCO, to enter cold shutdown (MODE 5). ITS 3.3.5 Required Action B.2.2 and ITS 3.3.6 Required Action B.2 will require entry into MODE 5 within 36 hours of failure to meet the LCO. These more restrictive requirements minimize the time during which the safety function is degraded while providing sufficient time to accomplish an orderly shutdown. Additionally, this Completion Time is consistent with NUREG-1430.

M5 CTS Table 3.5.1-1 Note 1 has been replaced by ITS 3.3.5 Required Action B.1 and ITS 3.3.6 Required Action B.1. CTS Table 3.5.1-1 Note 1 provides a time of 12 hours, from failure to meet the LCO, to enter hot shutdown (MODE 3). ITS 3.3.5 Required Action B.1 and ITS 3.3.6 Required Action B.1 will require entry into MODE 3 within 6 hours of failure to meet the LCO. These more restrictive requirements minimize the time during which the safety function is degraded while providing sufficient time to accomplish an orderly shutdown. Additionally, this Completion Time is consistent with NUREG-1430.

TECHNICAL CHANGE -- LESS RESTRICTIVE

L1 Specific Applicability statements for each of the Parameters in ITS Table 3.3.5-1 have been adopted. An Applicability exists in CTS only as implied by the appropriate action requirements which are CTS Table 3.5.1-1 Notes 1 and 5. These requirements would result in the unit being placed in cold shutdown (MODE 5) if any of the ESAS instrumentation Parameters contained more than one inoperable channel. The adoption of the specific ITS Applicability statements is less restrictive in that the Reactor Coolant System Pressure-Low Setpoint Parameter instrument channels will only be required OPERABLE when RCS pressure is above 1750 psig. This specific Applicability is consistent with the design of the ESAS, which provides the capability of bypassing this function when RCS pressure is reduced below 1750 psig (with some margin for instrumentation capabilities) and automatically removing this bypass when pressure is raised back above setpoint (CTS 3.5.3, Note **).

Additionally, ITS 3.3.5 Required Action B.2.1 along with its Note and the Note modifying Required Action B.2.2 have been adopted. This change provides action requirements to remove the unit from the Applicability of the LCO.

These changes have been made to provide requirements appropriate for the design and licensing basis for the unit. Additionally, this Completion Time is consistent with NUREG-1430.

L2 CTS Table 3.5.1-1 Note 8 indicated that if any one component of an ESAS digital subsystem is inoperable then the entire subsystem is inoperable. The design of the digital subsystems of the ESAS is such that there are five actuation logic channels contained in each of the two digital subsystems. A failure which renders one actuation logic channel inoperable may or may not affect any other of the actuation logic channels contained within that digital subsystem. As a result, the requirement to declare equipment inoperable while it is fully capable of performing its design function is inconsistent with both the CTS and ITS definitions of OPERABLE-OPERABILITY. The requirements of CTS Table 3.5.1-1 Note 8 are replaced by the ACTIONS of ITS 3.3.7.

L3 NUREG-1430 3.3.7 Required Action A.1 and its associated Completion Time have been adopted in the ITS. This Required Action allows equipment associated with an inoperable ESAS Actuation Logic Channel to be placed in its actuated state. This is an alternative to Required Action A.2, and CTS Table 3.5.1-1 Note 8, which would require declaring the equipment inoperable and entering the associated Required Actions for that equipment. This change allows additional flexibility in unit operation by not requiring the performance of the Required Actions for equipment made inoperable by the inoperability of an ESAS Actuation Logic Channel. This change provides requirements consistent with NUREG-1430 and which maintain the safety function of the equipment associated with the ESAS Actuation Logic Channels.

L4 NUREG-1430 3.3.6 Required Action A.1 and its associated Completion Time have been adopted in the ITS. This change establishes a 72 hour period of time in which the unit may continue operation, with one or more ESAS Functions having one channel of the manual initiation feature inoperable, prior to entering an ACTION which results in the unit entering MODE 3. This change has been made to provide ACTION requirements consistent with the safety function of the system, considering the allowed outage time for the actuated system. Additionally, this change is consistent with NUREG-1430.

L5 The Applicability statements of ITS 3.3.6 and 3.3.7 have been adopted. The Applicability for requirements related to these instrument channels was established, in CTS, only by the action requirements of CTS Table 3.5.1-1 Notes 1 and 5. These Notes could have resulted in the unit being placed in cold shutdown (MODE 5). Adoption of the ITS Applicabilities will require OPERABILITY of this instrumentation only during the MODES in which its actuated equipment is required to be OPERABLE. This change is consistent with the philosophy of the NUREG and with the requirements of NUREG-1430 as modified to accommodate the specific Applicabilities of the actuated equipment.

L6 Not used.

L7 CTS Table 4.1-1, items 15.a and 17.a require monthly testing of the HPI and LPI analog channels which are initiated by RCS pressure. CTS Note (1) on each of these two items indicates that the channel is tested "including test of shutdown bypass function (ECCS bypass function)." This Note and its requirements are omitted in

ITS 3.3.5. The bypass provides for operational flexibility only by preventing the actuation of ECCS during a shutdown. This bypass provides no safety function in that if the channel does not provide the intended bypass, the system can still perform its required actuations. If the ESAS is somehow prevented from actuation of the required components by the bypass, the channel is inoperable and the unit cannot enter the Applicable conditions for ITS 3.3.5.

The Bases for NUREG 3.3.5, LCO section, state: "Failure of a bypass bistable or bypass circuitry, such that a trip channel cannot be bypassed, does not render the channel inoperable." This is acceptable only if the bypass performs no safety function. Further, for this to be true, the bypass is not required to be tested with the CHANNEL FUNCTIONAL TEST, since if it were included, and it failed, the SR would be failed. Pursuant to SR 3.0.1, with a failed SR, the LCO would not be met, i.e., the channel would be inoperable. Since the Bases clearly indicate the channel is not inoperable, the bypass must not be a required function, and therefore, is not included in the SR.

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 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 Section 5 of the proposed Technical Specifications. This change is consistent with NUREG-1430.

CTS Location	New Location
Table 3.5.1-1 Column 1 "Number of Channels"	Bases 3.3.5, LCO
Table 3.5.1-1 Column 2 "No. of Channels for System Trip"	Bases 3.3.5, BACKGROUND
3.5.3	Bases 3.3.5, BACKGROUND

LA2 The information provided in Table 4.1-1, Item 20, Note (1) has been moved to the Bases of ITS 3.3.7, which describe the RB Spray system and its automatic actuation. This information provides details of design or process which are not directly pertinent to the actual requirement, i.e., Surveillance Requirement, but rather only further describe the required equipment. 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 Section 5 of the proposed Technical Specifications. This change is consistent with NUREG-1430.

3.3.5
3.3.6
3.3.7

A1

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

A3

<LATER> (3.3A, 3.3C, 3.3D, 3.4B) & LATER

3.5.1.1 Startup and operation are not permitted unless the requirements 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 be the limit given under Table 3.5.1-1, Columns 3 and 4, operation shall be limited as specified in Column 5.

<LATER> (3.3A) LATER

3.5.1.3 For on-line testing or in the event of a protection instrument 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 ligh 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 or channel bypass key shall be accessible for use in the control r 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.

R TRM

3.5.1.4 The key operated shutdown bypass switch associated with each rea protection channel shall not be used during reactor power operat except during channel testing.

<LATER> (3.3A) LATER

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 mainta: 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 sour 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 minutes following detection. The condition will be corrected ar 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 per: the reactor shall be placed in the hot shutdown condition withi: additional four hours.

3.3.5
3.3.6
3.3.7

A2

Bases

Every reasonable effort will be made to maintain all safety instrumentation in operation. A startup is not permitted unless the requirements of Table 3.5.1-1, Columns 3 and 4, are met.

Operation at rated power is permitted as long as the systems have at least the redundancy requirements of Column 4 (Table 3.5.1-1). This is in agreement with redundancy and single failure criteria of IEEE 279 as described in FSAR, Section 7.

There are four reactor protection channels. Normal trip logic is two-out-of-four. Required trip logic for the power range instrumentation channels is two-out-of-three. Minimum trip logic on other instrumentation channels is one-out-of-two.

The four reactor protection channels were provided with key operated bypass switches to allow on-line testing or maintenance on only one channel at a time during power operation. Each channel is provided with alarm and lights to indicate when that channel is bypassed. There will be one reactor protection system channel bypass switch key permitted in the control room. Upon the discovery of inoperable functions in any one reactor protection channel, the effect of the failure on the reactor protection syst and other interconnected systems is evaluated. The affected reactor protect channel may be placed in channel bypass, remain in operation in a degraded condition, or placed in the tripped condition as determined by operating conditions and management judgment. This action allows placing the plant in the safest condition possible considering the extent of the failure, plant conditions, and guidance from plant management. Should the failure in the reactor protection channel prohibit the proper operation of another system, the appropriate actions for the affected system are implemented. Administra controls are established to preclude placing a reactor protection channel in channel bypass when any other reactor protection channel contains an inopera function in the untripped state.

Each reactor protection channel key operated shutdown bypass switch is provided with alarm and lights to indicate when the shutdown bypass switch is being used.

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The source range and intermediate range nuclear flux instrumentation scales overlap by one decade. This decade overlap will be achieved at 10^{-10} amps c the intermediate range scale.

A2

The ESAS employs three independent and identical analog channels, which supply trip signals to two independent, identical digital subsystems. In order to actuate the safeguards systems, two out of three analog channels must trip. This will cause both digital subsystems to trip. Tripping of either digital subsystem will actuate all safeguards systems associated with that digital subsystem.

Because only one digital subsystem is necessary to actuate the safeguards systems and these systems are capable of tripping even when they are being tested, a single failure in a digital subsystem cannot prevent protective action.

3.3.5
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3.3.7

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.

<Add 3.3.5 ACTIONS NOTE
Add 3.3.6 ACTIONS NOTE
Add 3.3.7 ACTIONS NOTE> A5 LA1 Bases A3 A7

Table 3.5.1-1 (Cont'd)

ENGINEERED SAFEGUARDS ACTUATION SYSTEM

	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	
3.3.7 LCO	1. High pressure injection system (Note 8) Actuation Logic Channels						A9
3.3.5 LCO Table 3.3.5-1 parameter 1. & LATER (3.3D)	a. Reactor coolant pressure instrument channels	3	2	3 (Note 6)	1	Notes 1, 5	A6 & LATER
Parameter 2.	b. Reactor building 4 psig instrument channels	3	2	3 (Note 6)	1	Notes 1, 5	A6
3.3.6 LCO #a.	c. Manual trip pushbutton	2	1	2	1	Notes 1, 5	A8
3.3.7 LCO	2. Low pressure injection system (Note 8) Actuation Logic Channels						A9
3.3.5 LCO Table 3.3.5-1 Parameter 1. & LATER (3.3D)	a. Reactor coolant pressure instrument channels	3	2	3 (Note 6)	1	Notes 1, 5	A6 & LATER
Parameter 2.	b. Reactor building 4 psig instrument channels	3	2	3 (Note 6)	1	Notes 1, 5	A6
3.3.6 LCO #b.	c. Manual trip pushbutton	2	1	2	1	Notes 1, 5	A8
3.3.7 LCO	3. Reactor building isolation and reactor building cooling system (Note 8) Actuation Logic Channels						A9
3.3.5 LCO Table 3.3.5-1 Parameter 2.	a. Reactor building 4 psig instrument channel	3	2	3 (Note 6)	1	Notes 1, 5	A6
3.3.6 LCO #c.	b. Manual trip pushbutton	2	1	2	1	Notes 1, 5	A8

<Add 3.3.5 RA B.2.2 Note>
<Add 3.3.5 Appl. & Table 3.3.5-1 Applicable MODES or Other Specified Condition> L1
<Add 3.3.5 RA B.2.1 with NOTE>

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Table 3.5.1-1 (Cont'd

LA1 Bases

A3

A7

ENGINEERED SAFEGUARDS ACTUATION 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	
3.3.7 LCO	4. Reactor building spray pumps (Note 8) Actuation Logic Channels						A9
3.3.5 LCO Table 3.3.5-1 Parameter 3.	a. Reactor building 30 psig instrument channel	3	2	3 (Note 6)	1	Notes 1, 5	A6
3.3.6 LCO d.	b. Manual trip pushbutton	2	1	2	1	Notes 1, 5	A8
3.3.7 LCO	5. Reactor building spray valves (Note 8) Actuation Logic Channels						A9
3.3.5 LCO Table 3.3.5-1 Parameter 3.	a. Reactor building 30 psig instrument channel	3	2	3 (Note 6)	1	Notes 1, 5	A6
3.3.6 LCO e.	b. Manual trip pushbutton	2	1	2	1	Notes 1, 5	A8
<LATER> (3.3C)	EMERGENCY FEEDWATER INITIATION AND CONTROL SYSTEM 1. EFW Initiation a. Manual	2	1	2	1	Note 1	LATER

45a

<ADD 3.3.6 Appl.>
<ADD 3.3.7 Appl.> L5

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3.3.6
3.3.7

<Add 3.3.6 RA A.1> L4

<Add 3.3.7 RA A.1> L3

A3

TABLE 3.5.1-1 (Cont'd)

Notes:

3.3.5 RA B.1 / 3.3.6 RA B.1 & <LATER> (3.3A/C/D & 3.4B) — 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 — 6 — M5 & LATER

<LATER> (3.3A) — 2. When 2 of 4 power range instrument channels are greater than 10% rated power, hot shutdown is not required. — LATER

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.

3.3.5 RA B.2.2 / 3.3.6 RA B.2 & LATER (3.4B) — 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 — 36 — M4 & LATER

3.3.5 RA A.1 & <LATER> (3.3A, 3.3C) — 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 within one hour. Otherwise, the actions required by Column 5 shall apply. — M3 & 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

3.3.7 RA A.2 — 8. If any one component of a digital subsystem is inoperable, the entire digital subsystem is considered inoperable. Hence, Declare the associated safety features are inoperable within 1 hour and Specification 3.3 applies. — L2 — M2 — A1

<LATER> (3.3D) — 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. — LATER

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.

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3.5.3 Safety Features Actuation System Setpoints

Applicability

This specification applies to the safety features actuation system actuation setpoints.

Objective

To provide for automatic initiation of the safety features actuation system in the event of a breach of reactor coolant system integrity.

Specification

The safety features actuation setpoints and permissible bypasses shall be as follows:

Allowable Values A1

	Functional Unit	Action	Setpoint A1
Table 3.3.5-1 Parameter 1	High Reactor Building Pressure* High High	Reactor Building Spray	≤30 psig (44.7 psia)
Parameter 2	Reactor Building Pressure High	High Pressure Injection	≤ 4 psig (18.7 psia)
		Start of Reactor Building Cooling and Reactor Building Isolation	≤ 4 psig (18.7 psia)
		Reactor Bldg. Ventilation	≤ 4 psig (18.7 psia)
		Low Pressure Injection	≤ 4 psig (18.7 psia)
		Penetration Room Ventilation	≤ 4 psig (18.7) psia)
Parameter 3	Low Reactor Coolant System Pressure**	High Pressure Injection	≥ 1585 psig
		Low Pressure Injection	≥ 1585 psig
		Start of Reactor Building Cooling and Reactor Building Isolation	≥ 1585 psig

LA1 BASES

*May be bypassed during reactor building leak rate test. A12

**May be bypassed below 1750 psig and is automatically reinstated above 1750 psig. L1

With the safety features actuation setpoints less conservative than the above values, declare the channel inoperable and apply the applicable Action requirements of Table 3.5.1-1 until the channel is restored to OPERABLE status with the trip setpoint adjusted consistent with the trip setpoint value. A1

A13

33B

Bases

High Reactor Building Pressure

The basis for the 30 psig and 4 psig setpoints for the high pressure signal is to establish a setting which would be reached in adequate time in the event of a DBA, cover a spectrum of break sizes and yet be far enough above normal operation maximum internal pressure to prevent spurious initiation.

Low Reactor Coolant System Pressure

The basis for the 1585 psig low reactor coolant pressure setpoint for high and low pressure injection initiation is to establish a value which is high enough such that protection is provided for the entire spectrum of break sizes and is far enough below normal operating pressure to prevent spurious initiation. (1)

REFERENCES

(1) FSAR, Section 14.2.2.5

(2) B&W Calculation 32-1158581

A2

LAR

A13

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3.3.6
3.3.7

SURVEILLANCE REQUIREMENTS (Continued)

4.0.5 (Continued)

<LATER> (5.0)

b. Surveillance intervals specified in Section XI of the ASME Boiler and Pressure Vessel Code and applicable Addenda for the inservice inspection and testing activities required by the ASME Boiler and Pressure Vessel Code and applicable Addenda shall be applicable as follows in these Technical Specifications: LATER

ASME Boiler and Pressure Vessel Code and applicable Addenda terminology for inservice inspection and testing activities	Required frequencies for performing inservice inspection and testing activities
Weekly	At least once per 7 days
Monthly	At least once per 31 days
Quarterly or every 3 months	At least once per 92 days
Semiannually or every 6 months	At least once per 184 days
Yearly or annually	At least once per 366 days

c. The provisions of Specification 4.0.2 are applicable to the above required frequencies for performing inservice inspection and test activities.

d. Performance of the above inservice inspection and testing activit shall be in addition to other specified Surveillance Requirements

e. Nothing in the ASME Boiler and Pressure Vessel Code shall be construed to supersede the requiremetns of any Technical Specification.

4.1 OPERATIONAL SAFETY ITEMS A1

Applicability

Applies to items directly related to safety limits and limiting conditions fo operation.

Objective

To specify the minimum frequency and type of surveillance to be applied to unit equipment and conditions.

Specification

A3 & R TRM

a. The minimum frequency and type of surveillance required for react protective system and engineered safeguards system instrumentatio when the reactor is critical shall be as stated in Table 4.1-1.

3.3.5
3.3.6
3.3.7

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OPERATIONAL SAFETY ITEMS (continued)

<LATER> (3.3A, 3.3C, 3.3D)

4.1 (Continued)

b. Equipment and sampling test shall be performed as detailed in Table 4.1-2 and 4.1-3.

R TRM

A14

c. Discrepancies noted during surveillance testing will be corrected and recorded.

<LATER> (3.2)

LATER

d. A power distribution map shall be made to verify the expected power distribution at periodic intervals at least every 10 effective full power days using the incore instrumentation detector system.

A2

BASES

4.0.1 through 4.0.5 Establish the general requirements applicable to Surveillance Requirements. These requirements are based on the Surveillance Requirements stated in the Code of Federal Regulations, 10CFR 50.36(c)(3):

"Surveillance Requirements are requirements relating to test, calibration, or inspection to ensure that the necessary quality of systems and components is maintained, that facility operation will be within safety limits, and that the limiting conditions of operation will be met."

4.0.1 Establishes the requirement that surveillances must be performed during the operational modes or other conditions for which the requirements of the Limiting Conditions for Operation apply unless otherwise stated in an individual Surveillance Requirement. The purpose of this specification is to ensure that surveillances are performed to verify the operational status of systems and components and that parameters are within specified limits to ensure safe operation of the facility when the plant is in a mode or other specified condition for which the associated Limiting Conditions for Operation are applicable. Surveillance Requirements do not have to be performed when the facility is in an operational mode for which the requirements of the associated Limiting Condition for Operation do not apply unless otherwise specified.

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BASES (continued)

Under the terms of this specification, the more restrictive requirements of the Technical Specifications take precedence over the ASME Boiler and Pressure Vessel Code and applicable Addenda. The requirements of Specification 4.0.4 to perform surveillance activities before entry into an operational mode or other specified condition takes precedence over the ASME Boiler and Pressure Vessel Code provision which allows pumps and valves to be tested up to one week after return to normal operation. The Technical Specification definition of OPERABLE does not allow a grace period before a component, that is not capable of performing its specified function, is declared inoperable and takes precedence over the ASME Boiler and Pressure Vessel Code provision which allows a valve to be incapable of performing its specified function for up to 24 hours before being declared inoperable. A2

4.1 Bases

Check

Failures such as blown instrument fuses, defective indicators, faulted amplifiers which result in "upscale" or "downscale" indication can be easily recognized by simple observation of the functioning of an instrument or system. Furthermore, such failures are, in many cases, revealed by alarm or annunciator Action. Comparison of output and/or state of independent channels measuring the same variable supplements this type of built-in surveillance. Based on experience in operation of both conventional and nuclear plant systems, when the plant is in operation, the minimum checking frequency stated is deemed adequate for reactor system instrumentation. A2 & R TRM

Calibration

Calibration shall be performed to assure the presentation and acquisition of accurate information. The nuclear flux (power range) channels shall be calibrated at least twice weekly (during steady state operating conditions) against a heat balance standard to compensate for instrumentation drift. During nonsteady state operation, the nuclear flux channels shall be calibrated daily to compensate for instrumentation drift and changing rod patterns and core physics parameters. A2

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Other channels are subject only to "drift" errors induced within the instrumentation itself and, consequently, can tolerate longer intervals between calibrations. Process system instrumentation errors induced by drift can be expected to remain within acceptable tolerances if recalibration is performed once every 18 months.

Substantial calibration shifts within a channel (essentially a channel failure) will be revealed during routine checking and testing procedures.

Thus, minimum calibration frequencies for the nuclear flux (power range) channels, and once every 18 months for the process system channels is considered acceptable.

Testing

On-line testing of reactor protective channel and EFIC channels is required once every 4 weeks on a rotational or staggered basis. The rotation scheme is designed to reduce the probability of an undetected failure existing within the system and to minimize the likelihood of the same systematic test errors being introduced into each redundant channel. A2

All reactor protective channels will be tested before startup if the individual channel rotational frequency has been discontinued or if outage activities could potentially have affected the operability of one or more channels. A rotation will then be established to test the first Channel one week after startup, the second Channel two weeks after startup, the third Channel three weeks after startup, and the fourth Channel four weeks after startup.

The established reactor protective system instrumentation and EFIC test cycle is continued with one channel's instrumentation tested each week. Upon detection of a failure that prevents trip action, all instrumentation associated with the protective channels will be tested after which the rotational test cycle is started again. If actuation of a safety channel occurs, assurance will be required that actuation was within the limiting safety system setting.

The protective channels coincidence logic and control rod drive trip breakers are trip tested every quarter. The trip test checks all logic combinations and is to be performed on a rotational basis. The logic and breakers of the four protective channels shall be trip tested prior to startup and their individual channels trip tested on a cyclic basis. Discovery of a failure requires the testing of all channel logic and breakers, after which the trip test cycle is started again.

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~~The equipment testing and system sampling frequencies specified in Table 4.1-2 and Table 4.1-3 are considered adequate to maintain the status of the equipment and systems to assure safe operation.~~

~~REFERENCE~~

~~FSAR Section 7.1.2.3.4~~

(A3)

Table 4.1-1 (cont.)

	Channel Description	Check	Test	Calibrate	Remarks
<LATER> (3.3A)	~~13. High Reactor Building Pressure Channel~~	~~S~~	~~M~~	~~R~~	LATER
3.3.7	14. High Pressure Injection Logic Channel	NA	M SR 3.3.7.1	NA	
3.3.5	15. High Pressure Injection Analog Channels				
<LATER> (3.3D)	a. Reactor Coolant Pressure Channel	S SR 3.3.5.1	M(1) SR 3.3.5.2	R SR 3.3.5.3	~~(1) Including test of shutdown bypass function (ECCS bypass function).~~ L7
	b. Reactor Building 4 psig Channel	S SR 3.3.5.1	M SR 3.3.5.2	R SR 3.3.5.3	
3.3.7	16. Low Pressure Injection Logic Channel	NA	M SR 3.3.7.1	NA	
3.3.5	17. Low Pressure Injection Analog Channels				
<LATER> (3.3D)	a. Reactor Coolant Pressure Channel	S SR 3.3.5.1	M(1) SR 3.3.5.2	R SR 3.3.5.3	~~(1) Including test of shutdown bypass function (ECCS bypass function).~~ L7
	b. Reactor Building 4 psig Channel	S SR 3.3.5.1	M SR 3.3.5.2	R SR 3.3.5.3	
3.3.7	18. Reactor Building Emergency Cooling and Isolation System Logic Channel	NA	M SR 3.3.7.1	NA	
3.3.5	19. Reactor Building Emergency Cooling and Isolation System Analog Channels				
	a. Reactor Building 4 psig Channels	S SR 3.3.5.1	M SR 3.3.5.2	R SR 3.3.5.3	

3.3.5
3.3.7

A3

Table 4.1-1 (Cont'd)

		Channel Description	Check	Test	Calibrate	Remarks	
3.3.7	20.	Reactor Building Spray System System Logic Channels	NA	M(1) SR 3.3.7.1	NA	~~(1) Including RB spray pump, spray valves, and chem. add. valve logic channels.~~	LA2 Bases
3.3.5	21.	Reactor Building Spray System Analog Channels	<Add SR 3.3.5.1>				A11
		a. Reactor Building Pressure Channels	NA	SR 3.3.5.2 M	SR 3.3.5.3 R		
<LATER> (3.5D)	22.	~~Pressurizer Temperature Channels~~	~~S~~	~~NA~~	~~R~~		LATER
<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~~	~~(1) Check functioning of self-checking feature on each detector.~~	
<LATER> (3.4B) (3.3D)	28.	~~Radiation Monitoring Systems other than containment high range monitors (item 57)~~					LATER
		~~a. Process Monitoring System~~	~~S~~	~~Q~~	~~R~~		
		~~b. Area Monitoring System~~	~~S~~	~~M(1)~~	~~R~~		
<LATER> (3.3D)		~~c. Main Steam Line Radiation Monitors~~	~~S~~	~~M~~	~~R~~		LATER 3.3.5 3.3.7

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(A3)

Table 4.1-1 (Cont.)

	Channel Description	Check	Test	Calibrate	Remarks
	43. ESAS Manual Trip Functions		SR 3.3.6.1		(A1)
3.3.6	a. Switches & Logic	NA	R	NA	
3.3.7	b. Logic	NA	M	NA	(A10)
			SR 3.3.7.1		
⟨LATER⟩ (3.3A)	~~44. Reactor Manual Trip~~	~~NA~~	~~P~~	~~NA~~	LATER
⟨LATER⟩ (3.4B)	~~45. Reactor Building Sump Level~~	~~NA~~	~~NA~~	~~R~~	LATER
⟨LATER⟩ (3.3D)	~~46. EFW Flow Indication~~	~~M~~	~~NA~~	~~R~~	LATER

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Table 4.1-1 (Cont.)

	Channel Description	Check	Test	Calibrate	Remarks
<LATER> (3.3C)	d. SG A High Range Level High-high	S	M	R	LATER
	e. SG B High Range Level High-high	S	M	R	
<LATER> (3.3D)	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	
<LATER> (3.4B)	60. Low Temperature Overpressure Protection Alarm Logic	NA	R	R	
<LATER> (3.3D)	61. Core-exit Thermocouples	M	NA	R	
<LATER> (3.3A)	62 Electronic (SCR) Trip Relays	NA	Q	NA	
<LATER> (3.3D)	63 RVLMS	M	NA	R	
	64 HDLMS	M	NA	R	

+<LATER> (3.3A, 3.3C, 3.3D, 3.4B)

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

A4 +LATER + R TRM

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3.3.7

NO SIGNIFICANT HAZARDS CONSIDERATIONS STATEMENTS

ITS Section 3.3B: Instrumentation - ESAS

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:

3.3B L1

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

The Applicability for the ESAS Reactor Coolant System Pressure-Low Setpoint Parameter has been changed from an implied above MODE 5 to a specified ≥ 1750 psig Reactor Coolant System (RCS) pressure. Similarly, the Required Actions have been revised to require only that the MODE of Applicability be exited. This change in Applicability and Required Actions for this instrumentation parameter does not result in any hardware changes. This change 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). Also, the changes do not change the assumed response of the equipment in performing its specified mitigation functions from that considered during the original Applicability since the trip functions associated with this parameter were allowed by CTS to be bypassed during the Conditions which will be omitted from the revised Applicability. Therefore, the changes do not significantly increase the consequences of an accident.

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

The proposed change does not 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 availability for the required instrumentation. 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 ESAS Reactor Coolant System Pressure-Low Setpoint Parameter instrumentation provides ESAS actuation functions under certain operating conditions. In the conditions to be excluded from the Applicability, the actuation functions are bypassed and provide no input to the safety analysis. Therefore, the changes do not involve a significant reduction in the margin of safety.

NO SIGNIFICANT HAZARDS CONSIDERATIONS STATEMENTS

3.3B L2

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

The requirement to declare an entire ESAS digital subsystem inoperable in the event any portion of that subsystem is inoperable has been replaced with a requirement to declare only the affected portions inoperable. This change in ACTION requirements for this instrumentation parameter does not result in any hardware changes, neither does it result in any change in the function of the equipment. Therefore, this change does not significantly increase the probability of occurrence of any analyzed event. Also, the changes do not change the assumed response of the equipment in performing its specified mitigation functions. Therefore, the changes do not significantly increase the consequences of an accident.

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

The proposed change does not 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 availability for the required instrumentation. 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 ESAS actuation instrumentation provides initiation of ESAS functions under certain operating conditions. This change does not affect any operational or safety parameters, but rather provides for maintaining the operability of equipment which is capable of performing its safety function (and which would be declared inoperable under CTS). Therefore, the changes do not involve a significant reduction in the margin of safety.

NO SIGNIFICANT HAZARDS CONSIDERATIONS STATEMENTS

3.3B L3

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

This change provides the addition of an allowance to place equipment affected by an inoperable ESAS Actuation Logic Channel in the actuated position rather than declaring the affected equipment inoperable. This change in ACTION requirements for this instrumentation parameter does not result in any hardware changes. This change 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). Also, the changes do not change the assumed response of the equipment in performing its specified mitigation functions. Therefore, the changes do not significantly increase the consequences of an accident.

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

The proposed change does not 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 availability for the required equipment. 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 an ESAS digital subsystem is based on availability and capability of the actuated equipment to perform its safety function. This change maintains the capability of the required equipment to perform its safety function even in the absence of its actuating instrumentation. Therefore, this change does not represent a significant reduction in the margin of safety.

NO SIGNIFICANT HAZARDS CONSIDERATIONS STATEMENTS

3.3B L4

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

An extension of the Completion Time for a Required Action does not result in any hardware changes. The Completion Time for performance of Required Actions 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 limits the impact on probability). Also, an extension of the Completion Time provides additional opportunity to restore compliance with the requirements and avoid the increased potential for a transient during the shutdown process. Further, the Completion Time for performance of Required Actions does not significantly increase the consequences of an accident because the change does not change the assumed response of the equipment in performing its specified mitigation functions from that considered during the previous evaluation of accidents.

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?

Prompt and appropriate Required Actions have been determined based on the safety analysis functions to be maintained. The proposed Completion Time has been determined appropriate based on a combination of the time required to perform the action, the relative importance of the function or parameter to be restored, and engineering judgment. Therefore, the short extension of the Completion Time interval involves no significant reduction in the margin of safety.

NO SIGNIFICANT HAZARDS CONSIDERATIONS STATEMENTS

3.3B L5

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

The Applicability for the ESAS Manual Initiation and ESAS Actuation Logic have been changed from an implied "above MODE 5" to a specified "MODES 1 and 2" and "MODES 3 and 4 when associated engineered safeguard equipment is required to be OPERABLE." This change in Applicability for this instrumentation parameter does not result in any hardware changes. This change 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). Also, the changes do not change the assumed response of the equipment in performing its specified mitigation functions from that considered during the original Applicability since the trip functions associated with this parameter continue to be required when the associated equipment is required. Therefore, the changes do not significantly increase the consequences of an accident.

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

The proposed change does not 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 availability for the required instrumentation whenever the actuated equipment is required. 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 ESAS Manual Initiation and ESAS Actuation Logic instrumentation provides ESAS actuation functions under certain operating conditions. In the conditions to be excluded from the Applicability, the actuation functions are not required since the associated equipment is not required and provides no input to the safety analysis. Therefore, the changes do not involve a significant reduction in the margin of safety.

3.3B L6 Not Used

3.3B L7

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

The Applicability for the ESAS Reactor Coolant System (RCS) Pressure-Low Setpoint Parameter is specified as ≥ 1750 psig RCS pressure. As such, the operational bypass that prevents ECCS actuation during a shutdown when the RCS pressure drops below this value does not function during the conditions for which the ESAS RCS Pressure-Low Setpoint Parameter is required to be OPERABLE. Therefore, this bypass is removed from the CTS as a required function. This change does not result in any hardware changes, and 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). Also, the revision does not change the assumed response of the equipment in performing its specified mitigation functions since the actuation function associated with this parameter will continue to be available and OPERABLE. Therefore, the changes do not significantly increase the consequences of an accident.

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

The proposed change does not 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 proper availability for the required instrumentation. 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 ESAS RCS Pressure-Low Setpoint Parameter instrumentation is assumed to provide ESAS actuation functions with the RCS pressure at ≥ 1750 psig. Below these conditions, the instrumentation is allowed to be bypassed for operational considerations; however, the bypass performs no safety function. Therefore, the margin of safety is not dependent on the bypass and the change does not involve a significant reduction in the margin of safety.

ITS DISCUSSION OF DIFFERENCES
ITS Section 3.3B: Instrumentation - ESAS

Note: The ITS Section 3.3B package addresses the following NUREG-1430 LCOs:
- 3.3.5 ESFAS Instrumentation
- 3.3.6 ESFAS Manual Initiation
- 3.3.7 ESFAS Automatic Actuation Logic

1 NUREG 3.3.5, 3.3.6, & 3.3.7.- The ANO-1 unit specific terminology has been inserted to replace generic usage in NUREG 3.3.5, 3.3.6, and 3.3.7. Specifically, the term "Engineered Safety Feature Actuation System (ESFAS)" has been replaced with the ANO-1 unit specific term "Engineered Safeguards Actuation System (ESAS)." The associated acronyms have also been changed as appropriate. The corresponding Bases have been marked to indicate these changes, however the reference to this discussion is generally made only once per page to conserve space and maintain readability. These changes were made to maintain the usage of terminology consistent with design and licensing basis documentation. This change is consistent with current license basis.

2 NUREG 3.3.5 – Condition B has been revised to specify that this Condition applies when more than one channel is inoperable for each of one or more Parameters. This change was made to maintain requirements consistent with CTS Table 3.5.1-1 Column 5 and Note 1 which provide specific requirements for the inoperability of more than one channel. This change is consistent with current license basis.

Without this addition, entry into the ACTION requirements of ITS LCO 3.0.3 would be required if more than one channel is inoperable for each of one or more Parameters. Entry into the Required Actions of ITS 3.3.5 Condition B rather than the ACTION requirements of LCO 3.0.3 is more appropriate because, specific Required Actions, which result in the unit exiting the Applicability for each ESAS instrumentation Parameter, are provided in ITS 3.3.5. These Required Actions consistently result in the unit exiting the specific Applicability within a specific Completion Time. For example, ITS LCO 3.0.3 ACTION requirements would not provide a specific Completion Time for reducing RCS pressure to less than 1750 psig, in the event more than one channel of the RCS Pressure--Low Setpoint Parameter was inoperable. This change is consistent with TSTF-217, Rev. 1, as revised to reflect plant specific terminology.

3 NUREG 3.3.5 - Response time testing of the Engineered Safeguards Actuation System (ESAS), i.e., NUREG SR 3.3.5.4, is not adopted in ITS. Testing of this type is not required by ANO-1 CTS. Deletion of these Surveillance Requirements maintains consistency with the current ANO-1 licensing basis and neither removes any current requirement nor adds any additional requirement.

4 NUREG 3.3.5 – Table 3.3.5-1 Parameter 2, "Reactor Coolant System Pressure-Low Low Setpoint," and the specific Required Action associated with that Parameter, NUREG 3.3.5 Required Action B.2.2, are not adopted in ITS. The design of the ANO-1 ESAS includes only one low reactor coolant system pressure setpoint. This setpoint is used as an actuation initiator for both High Pressure Injection and Low Pressure Injection Systems. These changes were made to provide requirements appropriate for the ANO-1 unit specific system design. This change is consistent with current license basis.

5 NUREG 3.3.5, 3.3.6, and 3.3.7 - The unit specific design of the ANO-1 ESAS provides for three analog instrument channels for each of the monitored parameters. These three analog instrument channels provide the required input to each of the ten digital actuation logic channels. Contrary to the system design depicted in the requirements of NUREG-1430, these three analog instrument channels provide input to both trains of digital actuation logic channels. This unit specific design difference required the deletion of the phrase "in each ESFAS train" from LCO 3.3.5 as well as appropriate changes to the Bases to designate unit specific digital and analog channel terminology. This change is consistent with current license basis.

6 NUREG 3.3.5 - Specific detail of the equipment actuated by each of the Parameters listed in NUREG Table 3.3.5-1 has been removed. Similar details are included in the appropriate Bases sections and are not specifically pertinent requirements of LCO 3.3.5. Removal of this information represents no actual change in requirements, only a change in presentation of amplifying information. This change is considered editorial.

7 NUREG 3.3.6 & 3.3.7 - The Applicability of ITS 3.3.6 and ITS 3.3.7 has been modified to only include the portions of MODE 3 in which the associated ES equipment is required to be OPERABLE. This change was made to reflect the fact that some ESAS actuated equipment is not required in either MODE 3 or MODE 4. For example, neither CTS nor the proposed ITS requires the RB Spray Additive (sodium hydroxide) system, which is actuated by the ESAS, to be OPERABLE below MODE 2. This change was made to provide Applicabilities for the ESAS requirements which are consistent with the Applicabilities of the actuated equipment. Without this change, the actuating components would be required to be Operable when the components being actuated are not required to be Operable. This change is consistent with the current license basis.

8 NUREG 3.3.7 - The Frequency of ITS SR 3.3.7.1 has been adopted as 31 days. The change to the NUREG-1430 Frequency of 31 days on a STAGGERED TEST BASIS was made to retain testing requirements on a Frequency consistent with the CTS. The CTS requires this testing monthly, which is considered administratively equivalent to the proposed 31 day Frequency. This change is consistent with current license basis.

9 NUREG 3.3.6 and 3.3.7 - The Functions specified in NUREG ITS 3.3.6 have been modified to match the Functions as presented in the CTS and in the ANO-1 SAR. Identification of the digital actuation logic channel numbers which correspond to the Functions stated in the LCO have been provided in the Bases for additional clarity. ITS 3.3.7 has also been modified to include ANO-1 unit specific terminology, e.g., analog instrument channels and digital actuation logic channels. These changes were made to provide requirements consistent with the design of the ANO-1 ESAS and consistent with the specific terminology and names associated with the ANO-1 ESAS. Additional associated Bases changes are also included in B3.3.5, B3.3.6, and B3.3.7. These changes are consistent with current license basis.

10 NUREG Bases 3.3.5, 3.3.6, and 3.3.7 - ANO-1 unit specific terminology and design details have been added to the Bases of ITS 3.3.5, 3.3.6, and 3.3.7. Additionally, information which is not specifically pertinent to the Bases discussion for these specifications and which may be duplicative of information contained in the SAR has been removed. These changes provide unit specific details of system design, maintain usage of terminology consistent with design and licensing basis documentation, e.g., development of trip setpoints and Allowable Values, and reduce duplication of discussion which is not specifically pertinent to the specifications. These changes are consistent with current license basis.

11 NUREG Bases 3.3.5 – SR 3.3.5.1 Bases discussion of the performance of CHANNEL CHECKS on off scale instruments was removed. The instrument channels associated with the ANO-1 ESAS are not expected to be in an off scale condition while performance of this SR is required, therefore this potentially confusing discussion is not adopted. This change is consistent with current license basis.

12 NUREG SR 3.3.5.2 and Bases - The Note allowing a channel to be placed in an inoperable status for the performance of the Channel Functional Test for a limited period of time is not incorporated in the ITS. The ANO-1 design of ESAS does not include a channel bypass for testing. The built-in test facilities permit an electrical trip test of each analog instrument string by the substitution of signals at the buffer amplifiers. When an analog instrument string is placed in test, all associated analog subsystem outputs go to the trip state. This assures that all protective action cannot be defeated by placing analog instrument strings in test. This change is consistent with the current license basis.

13 NUREG Bases 3.3.6 - The Applicable Safety Analyses discussion has been modified to remove the implication that the OPERABILITY of the ESAS Manual Initiation was required to be OPERABLE in all MODES, "at any time." Additionally, reference to the ability to specify the use of the ESAS manual initiation in operating procedures was removed. This ability exists for this and numerous other Functions. However, that fact is not pertinent to this discussion. Furthermore, it implies that specifying other functions in operating procedures might be inappropriate. This change is consistent with current license basis.

ITS DISCUSSION OF DIFFERENCES

14 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.

The 10 CFR 50.36 Criterion satisfied by the ITS LCOs was modified to preserve consistency with the ANO-1 license basis. The NUREG Criterion specified were modified to be consistent with the analysis assumptions regarding equipment availability and operating condition (i.e., MODE). For ITS 3.3.6, Manual ESAS Initiation is not credited in any ANO-1 safety analysis, therefore, Criterion 4 was identified. This change is consistent with current license basis and 10 CFR 50.36.

15 NUREG Bases 3.3.7 - Unit specific clarification of the use of ITS 3.3.7 Required Actions A.1 and A.2 has been added to the Bases (B3.3.7). These additions have been made to provide additional guidance and clarification on the proper usage of the Required Actions without changing the intent of the ACTIONS.

16 Not used.

CTS

①

3.3 INSTRUMENTATION

3.3.5 Engineered ~~Safety Feature~~ Safeguards Actuation System (~~ESFAS~~ ESAS) Instrumentation

⑤ Table 3.5.1-1 ESAS Functional Unit 1a, 1b, 2a, 2b, 3a, 4a, 5a

LCO 3.3.5 Three ESAS analog instrument channels ~~of ESFAS instrumentation~~ for each Parameter in Table 3.3.5-1 shall be OPERABLE ~~in each ESFAS train~~.

APPLICABILITY: According to Table 3.3.5-1. N/A

ACTIONS

-----------------------------------NOTE-----------------------------------
Separate Condition entry is allowed for each Parameter. N/A

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more Parameters with one analog instrument channel inoperable.	A.1 Place channel in trip.	1 hour
B. One or more Parameters with more than one analog instrument channel inoperable. OR Required Action and associated Completion Time not met.	B.1 Be in MODE 3. AND B.2 ~~.1~~ --------NOTE-------- Only required for RCS Pressure—Low setpoint. ------------------------- Reduce RCS pressure < ~~[1800]~~ 1750 psig. ~~AND~~	6 hours 36 hours

(continued)

Table 3.5.1-1 Note 6 ⑤

Table 3.5.1-1 Note 1

N/A

②

1

CTS

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
B. (continued)	B.2.2 --------NOTE--------- Only required for RCS Pressure—Low Low setpoint. ------------------------ Reduce RCS pressure < [900] psig.	36 hours
	AND	
	B.2.3 --------NOTE--------- Only required for Reactor Building Pressure High setpoint and High High setpoint. ------------------------ Be in MODE 5.	36 hours

4

EDIT

N/A

Table 3.5.1-1
Note 5

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
SR 3.3.5.1 Perform CHANNEL CHECK.	12 hours

(continued)

Table 4.1-1
"Check" Items
15a, b
17a, b
19a
NA

CTS

1

SURVEILLANCE REQUIREMENTS (continued)

	SURVEILLANCE	FREQUENCY
SR 3.3.5.2	~~NOTE: When an ESFAS channel is placed in an inoperable status solely for performance of this Surveillance, entry into associated Conditions and Required Actions may be delayed for up to 8 hours, provided the remaining two channels of ESFAS instrumentation are OPERABLE or tripped.~~ Perform CHANNEL FUNCTIONAL TEST.	31 days
SR 3.3.5.3	Perform CHANNEL CALIBRATION.	~~[18]~~ months
~~SR 3.3.5.4~~	~~Verify ESFAS RESPONSE TIME within limits.~~	~~[18] months on a STAGGERED TEST BASIS~~

N/A

12 Table 4.1-1 "Test" Items 15a,b 17a,b 19a, 21a

Table 4.1-1 "Calibrate" Items 15a,b 17a,b 19a,21a

3

CTS

Table 3.3.5-1 (page 1 of 1)
Engineered ~~Safety Feature~~ Safeguards Actuation System Instrumentation

Table 3.5.1-1
ESAS Functional
Unit 1a, 2a

PARAMETER	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	ALLOWABLE VALUE
1. Reactor Coolant System Pressure — Low Setpoint ~~(HPI Actuation, RB Isolation, RB Cooling, EDG Start)~~	≥ ~~[1800]~~ 1750 psig	≥ ~~[1600]~~ 1585 psig
~~2. Reactor Coolant System Pressure — Low Low Setpoint (HPI Actuation, LPI Actuation, RB Isolation, RB Cooling)~~	~~≥ [900] psig~~	~~≥ [400] psig~~
2 ~~3~~. Reactor Building (RB) Pressure — High Setpoint ~~(HPI Actuation, LPI Actuation, RB Isolation, RB Cooling)~~	1,2,3,4	≤ ~~[5] psig~~ 18.7 psia
3 ~~4~~. RB ~~Reactor Building~~ Pressure — High High Setpoint ~~(RB Spray Actuation)~~	1,2,3,4	≤ ~~[30] psig~~ 44.7 psia

3.5.3

NA

4

Table 3.5.1-1
ESAS Functional
Unit 1b, 2b, 3a

3.5.3

NA

edit

3.5.3

NA

Table 3.5.1-1
ESAS Function
Unit 4a, 5a

1

6

CTS

3.3 INSTRUMENTATION

3.3.6 Engineered ~~Safety Feature~~ Safeguards Actuation System (ESFAS) Manual Initiation

1

LCO 3.3.6 Two manual initiation channels of each one of the ESFAS Functions below shall be OPERABLE:

Table 3.5.1-1 Functional Units

a. High Pressure Injection (channels 1 and 2); 1.c

b. Low Pressure Injection (channels 3 and 4); 2.C

c. Reactor Building (RB) Cooling (channels 5 and 6); 3.b

d. RB Spray (channels 7 and 8); and 4.b

9

~~e. RB Isolation; and~~

~~f. Control Room Isolation.~~

e. Spray Additive (channels 9 and 10). 5.b

APPLICABILITY: MODES 1, 2, and 3, MODES 3 and 4 when associated engineered safeguards equipment is required to be OPERABLE. N/A edit

7

ACTIONS

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

N/A

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. One or more ESFAS Functions with one channel inoperable.	A.1	Restore channel to OPERABLE status.	72 hours
B. Required Action and associated Completion Time not met.	B.1 AND	Be in MODE 3.	6 hours (continued)

1

N/A

Table 3.5.1-1 Note 1

1

CTS

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
B. (continued)	B.2 Be in MODE 5.	36 hours

Table 3.5.1-1
Note 5

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
SR 3.3.6.1 Perform CHANNEL FUNCTIONAL TEST.	~~[18]~~ months

Table
4.1-1
Item 43a

3.3 INSTRUMENTATION 1

3.3.7 Engineered Safeguards ~~Safety Feature~~ Actuation System (ESFAS) Automatic Actuation Logic 1 9

Table 3.5.H
ESAS Functional Unit 1, 2, 3, 4, 5
edit

LCO 3.3.7 The ~~All the~~ ESFAS digital automatic actuation logic channels matrices shall be OPERABLE. 5

7

APPLICABILITY: MODES 1, 2, and 3,
MODES 3 and MODE 4 when associated engineered safeguards equipment is required to be OPERABLE. N/A

5

ACTIONS

9

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

Separate Condition entry is allowed for each digital automatic actuation logic channel matrix. N/A

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. One or more digital automatic actuation logic channels matrices inoperable.	A.1	Place associated component(s) in engineered safeguards configuration.	1 hour
	<u>OR</u>		
	A.2	Declare the associated component(s) inoperable.	1 hour

5 N/A 9

Table 3.5.1-1 Note 8

9 CTS

1

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.3.7.1	Perform digital ~~automatic~~ actuation logic CHANNEL FUNCTIONAL TEST.	~~31 days on a STAGGERED TEST BASIS~~

5 9 8 Table 4.1-1 Items 14, 16, 18, 20, 43b

1

B 3.3 INSTRUMENTATION

B 3.3.5 Engineered Safeguards ~~Safety Feature~~ Actuation System (ES~~F~~AS) Instrumentation

BASES

BACKGROUND

The ES~~F~~AS initiates necessary safety systems, based on the values of selected unit Parameters, to protect against violating core design limits and ~~reactor coolant pressure boundary and~~ to mitigate accidents. 10

~~ESFAS actuates the following systems:~~

- ~~High pressure injection (HPI) Actuation;~~
- ~~Low pressure injection (LPI) Actuation;~~
- ~~Reactor building (RB) Cooling;~~
- ~~RB Spray;~~
- ~~RB Isolation; and~~
- ~~Emergency diesel generator (EDG) Start.~~

~~ESFAS also provides a signal to the Emergency Feedwater Isolation and Control (EFIC) System. This signal initiates emergency feedwater (EFW) when HPI is initiated.~~ 10

The ES~~F~~AS operates in a distributed manner to initiate the appropriate systems. The ES~~F~~AS does this by determining the need for actuation in each of three analog instrument channels monitoring each actuation Parameter. Once the need for actuation is determined, the condition is transmitted to digital ~~automatic~~ actuation logic ~~s~~ Channels which perform the two-out-of-three logic to determine the actuation of each end device. ~~Each end device has its own automatic actuation logic, although all automatic actuation logics take their signals from the same point in each channel for each Parameter.~~ 5

edit

Three ~~Four~~ Parameters are used for actuation: 4

- Low Reactor Coolant System (RCS) Pressure;
- ~~Low Low RCS Pressure;~~
- High Reactor Building (RB) ~~RB~~ Pressure; and 10

(continued)

BASES

BACKGROUND (continued)

- High High RB Pressure.

LCO 3.3.5 covers only the instrumentation channels that measure these Parameters. These channels include all the intervening equipment necessary to produce actuation before the measured process Parameter exceeds the limits assumed by the accident analysis. This includes sensors, bistable devices, operational bypass circuitry, block timers, and output relays. LCO 3.3.6, "Engineered Safety Feature Actuation System (ESFAS) Manual Initiation," and LCO 3.3.7, "Engineered Safety Feature Actuation System (ESFAS) Automatic Actuation Logic," provide requirements on the manual initiation and automatic actuation logic Functions.

analog

an

signal input to the digital actuation logic

logic buffer modules

Safeguards

Safeguards

digital

The ESFAS consists of three protection channels. Each channel provides input to logics that initiate equipment with a two-out-of-three logic on each component. Each protection channel includes bistable inputs from one instrumentation channel of Low RB Pressure, Low Low RCS Pressure, High RB Pressure, and High High RB Pressure. Automatic actuation logics combine the three protection channel trips in each train to actuate the individual Engineered Safety Feature (ESF) components needed to initiate each ESF System. Figure [ ], FSAR, Chapter [7] (Ref. 1), illustrates how instrumentation channel trips combine to cause protection channel trips.

INSERT B3.3-46A

The RCS pressure sensors are common to both trains. Isolation is provided via separate bistables for each train. Separate RB pressure sensors are used for the high and high high pressure Functions in each train, and separate sensors are used for each train.

INSERT B3.3-46B

The following matrix identifies the measurement channels and the Function actuated by each Parameter.

systems

ESAS digital actuation logic

(1) (10) (9) (5) (4) (5) (10) (5) (6) (5)

(continued)

<INSERT B3.3-46A>

The ESAS monitors three parameters via analog instrument channels. Each analog instrument channel provides input to the appropriate digital actuation logic channels that initiate equipment with a two-out-of-three coincidence logic on each digital channel. Each digital actuation logic channel includes bistable inputs from all three analog instrument channels of one parameter, i.e., either Low RCS Pressure, High RB Pressure, or High High RB Pressure. The digital actuation logic combines the analog instrument channel trips to actuate the individual Engineered Safeguards (ES) components needed to initiate each ES System. Figure 7.6, SAR, Chapter 7 (Ref. 1), also illustrates how analog instrument channel trips combine to cause digital actuation logic channel trips.

<INSERT B3.3-46B>

The ESAS is divided into five Functions actuated by ten digital actuation logic channels.

The ESAS High Pressure Injection (HPI) Function is actuated by ESAS digital actuation logic channels 1 and 2 and includes the following system actuations: HPI, a subset of RB isolation valves, diesel generators (DGs), and ES electrical alignment. Digital actuation logic channels 1 and 2 are actuated by two-out-of-three RCS Pressure—Low analog instrument channels, or two-out-of-three RB Pressure—High analog instrument channels.

The ESAS Low Pressure Injection (LPI) Function is actuated by ESAS digital actuation logic channels 3 and 4 and includes the following system actuations: LPI, a subset of RB isolation valves, and emergency feedwater (EFW) through an ESAS signal provided to the Emergency Feedwater Initiation and Control (EFIC) Instrumentation System. Digital actuation logic channels 3 and 4 are actuated by two-out-of-three RCS Pressure—Low analog instrument channels, or two-out-of-three RB Pressure—High analog instrument channels.

The ESAS RB Cooling Function is actuated by ESAS digital actuation logic channels 5 and 6 and includes the following system actuations: RB cooling, a subset of RB isolation valves, and RB penetration room ventilation system. Digital actuation logic channels 5 and 6 are actuated by two-out-of-three RB Pressure—High analog instrument channels.

The ESAS RB Spray Function is actuated by ESAS digital actuation logic channels 7 and 8 and includes the following system actuations: RB spray. Digital actuation logic channels 7 and 8 are actuated by two-out-of-three RB Pressure—High High analog instrument channels.

The ESAS Spray Additive Function is actuated by ESAS digital actuation logic channels 9 and 10 and includes the following system actuations: spray additive. Digital actuation logic channels 9 and 10 are actuated by two-out-of-three RB Pressure—High High analog instrument channels.

⑥

BASES

BACKGROUND (continued)

INSERT B 3.3-47A

PARAMETER	LOW RCS PRESSURE	LOW LOW RCS PRESSURE	HIGH RB PRESSURE	HIGH HIGH RB PRESSURE
HPI	X	X	X	
LPI		X		X
RB Cooling	X	X	X	(b)
RB Spray	(b)			
RB Isolation[a]	X	X	X	
EDG Start	X	X	X	
Control Room Isolation			X	

(a) Only isolates systems not required for RB or RCS heat removal.

(b) Actuates on High High RB Pressure coincident with HPI actuation.

~~Engineered safeguards bus undervoltage will also sequence on the HPI loads started by the HPI block timers. However, HPI will not occur unless the ESFAS HPI signal is also present. LCO 3.3.8, "Emergency Diesel Generator (EDG) Loss of Power Start (LOPS)," contains the requirements for the undervoltage channels.~~ ⑩

The ES~~F~~ equipment is divided between the two redundant actuation trains ~~A and B~~. The division of the equipment between the two actuation trains is based on the equipment redundancy and function and is accomplished in such a manner that the failure of one of the digital actuation logic channels and the related safeguards equipment will not inhibit the overall ES~~F~~ Functions. ~~Where a motor operated or a solenoid operated valve is driven by either of two matrices, one is from actuation channel A and one from actuation channel B.~~ Redundant ES~~F~~ pumps are controlled from separate and independent digital actuation logic channels.

edit

⊢⑤

⑩

⊢⑤

The actuation of ES~~F~~ equipment is also available by manual actuation switches located on the control room console.

(continued)

<INSERT B3.3-47A>

ESAS Digital Actuation Logic Channels	Actuated Systems	Parameter		
		RCS Press. Low	RB Press. High	RB Press. High High
1 and 2	Subset of RB Isolation, ES Electrical Alignment, HPI, and DG Start.	X	X	
3 and 4	Subset of RB Isolation, LPI, and EFIC EFW.	X	X	
5 and 6	Subset of RB Isolation, RB Cooling, and Penetration Room Vent.		X	
7 and 8	RB Spray			X
9 and 10	Spray Additive			X

1

BASES

BACKGROUND (continued)

The ~~ESFAS~~ ESAS, in conjunction with the actuated equipment, provides protective functions necessary to mitigate Design Basis Accidents (DBAs), specifically the loss of coolant accident (LOCA), and steam line break (SLB) events. (Ref. 2) as a backup to mitigate the RB DBA. The ~~ESFAS~~ ESAS relies on the OPERABILITY of the ~~automatic~~ digital actuation logic ~~for each component~~ channels to perform the actuation of the selected systems ~~of LCO 3.3.7~~. 5 9

Engineered ~~Safety Feature~~ Safeguards Actuation System Bypasses

No provisions are made for maintenance bypass of ~~ESFAS~~ ESAS instrumentation channels. Operational bypass of certain channels is necessary to allow accident recovery actions to continue and, for some channels, to allow reactor shutdown without ~~spurious~~ ~~ESFAS~~ ESAS actuation.

The ~~ESFAS~~ ESAS RCS pressure analog instrument~~ation~~ channels include permissive bistables that allow manual bypass when reactor pressure is below the point at which the low ~~and low low~~ pressure trip ~~are~~ is required to be OPERABLE. Once permissive conditions are sensed, the RCS pressure trips may be manually bypassed. Bypasses are automatically removed when bypass permissive conditions are exceeded. 5 4

INSERT B 3.3-48A

~~Each High RB Pressure channel may be manually bypassed after the other two channels in the Parameter have tripped. The manual bypass allows operators to take manual control of ESF Functions after initiation to allow recovery actions. The bypass may be manually removed and is automatically removed when RB pressure returns to below the trip setpoint.~~ 10

Reactor Coolant System Pressure

The RCS pressure is monitored by three independent pressure transmitters located in the RB. These transmitters are separate from the transmitters that feed the Reactor Protection System (RPS). Each of the pressure signals generated by these transmitters is monitored by ~~four~~ two bistables to provide a ~~two~~ trip signal~~s~~, at ~~1500~~ ≥1685 psig ~~and 500 psig, and two~~ bypass permissive signal~~s~~, at ~~1700~~ ≤1750 psig~~, and 900 psig~~. (a) 10

The outputs of the three low RCS pressure trip bistables ~~associated with the low RCS pressure, 1500 psig, trip~~ drive relays in two sets

(continued)

<INSERT B3.3-48A>

This bypass provides an operational provision only outside the Applicability for this Parameter, and provides no safety function.

BASES

BACKGROUND

Reactor Coolant System Pressure (continued)

digital

~~(actuation channels A and B)~~ of identical and independent ~~channels. These two sets of HPI channels each use three logic channels used in two-out-of-three coincidence networks for HPI Actuation.~~ ~~The outputs of the three bistables associated with the Low Low RCS Pressure [500 psig] trip drive relays in two sets (actuation channels A and B) of identical and independent channels. These two sets of LPI channels each use three logic channels used in two-out-of-three coincidence networks for LPI Actuation. The outputs of the three Low Low RCS Pressure bistables also trip the drive relays in the corresponding HPI Actuation channel as previously described.~~

digital logic

INSERT B3.3-49A

Reactor Building Pressure

INSERT B 3.3-49B

~~RB pressure inputs to the ESFAS are provided by 12 pressure switches. Six pressure switches are used for the High RB Pressure Parameter, and six pressure switches are used for the High High Pressure Parameter.~~

~~The output contacts of six High RB Pressure switches are used in two sets of identical and independent actuation trains. These two trains each use three logic channels. The outputs of these channels are used in two-out-of-three coincidence networks. The output contacts of the six RB pressure switches also trip the drive relays in the corresponding HPI and LPI Actuation channels as previously described.~~

~~The output contacts of six High High RB Pressure switches are used in two sets of identical and independent actuation trains. These two trains each use three logic channels (RB4, RB5, and RB6). The outputs of these channels are used in two-out-of-three coincident networks for RB Spray Actuation. Each high high pressure train actuates one RB Spray train when the High High RB signal and the HPI signal are coincident in that train.~~

~~Trip Setpoints and Allowable Values~~

~~Trip setpoints are the nominal value at which the bistables are set. Any bistable is considered to be properly adjusted~~

(continued)

INSERT B3.3-49A

Each analog instrument channel can be tested online to verify that the signal and trip setpoint are within the specified allowance requirements of approved calibration procedures. The built-in test facilities permit an electrical trip test of each analog instrument string by the substitution of signals at the buffer amplifiers. When an analog instrument string is placed in test, all associated analog subsystem outputs go to the trip state. This assures that all protective action cannot be defeated by placing analog instrument strings in test.

<INSERT B3.3-49B>

The RB pressure is monitored by three independent pressure transmitters located inside the RB. These transmitters are separate from the transmitters that feed the Reactor Protection System (RPS). Each of the pressure signals generated by these transmitters is monitored by two bistables to provide trip signals. The outputs of the bistables, associated with the RB Pressure—High and RB Pressure—High High trips, drive relays in two sets of identical and independent digital instrument channels. These two sets of channels each use two-out-of-three coincidence digital logic for automatic actuation.

Each analog channel can be tested online to verify that the signal and trip setpoint are within the specified allowance requirements of approved calibration procedures. The built-in test facilities permit an electrical trip test of each analog instrument string by the substitution of signals at the buffer amplifiers. When an analog instrument string is placed in test, all associated analog subsystem outputs go to the trip state. This assures that all protective action cannot be defeated by placing analog instrument strings in test.

BASES

BACKGROUND

Trip Setpoints and Allowable Values (continued)

when the "as left" value is within the band for CHANNEL CALIBRATION accuracy (i.e., ± [rack calibration + comparator setting accuracy]).

The trip setpoints used in the bistables are based on the analytical limits stated in Figure [ ], FSAR, Chapter [7] (Ref. 1). The selection of these trip setpoints is such that adequate protection is provided when all sensor and processing time delays are taken into account. To allow for calibration tolerances, instrumentation uncertainties, instrument drift, and severe environment induced errors for those ESFAS channels that must function in harsh environments as defined by 10 CFR 50.49 (Ref. 2), the Allowable Values specified in Table 3.3.5-1 in the accompanying LCO are conservatively adjusted with respect to the analytical limits. A detailed description of the methodology used to calculate the trip setpoints, including their explicit uncertainties, is provided in the "Unit Specific Setpoint Methodology" (Ref. 3). The actual nominal trip setpoint entered into the bistable is more conservative than that specified by the Allowable Value to account for changes in random measurement errors detectable by a CHANNEL FUNCTIONAL TEST. One example of such a change in measurement error is drift during the surveillance interval. A channel is inoperable if its actual trip setpoint is not within its required Allowable Value.

10

Setpoints, in accordance with the Allowable Values, ensure that the consequences of DBAs will be acceptable, providing the unit is operated from within the LCOs at the onset of the DBA and the equipment functions as designed.

Each channel can be tested online to verify that the setpoint accuracy is within the specified allowance requirements of Reference 3. Once a designated channel is taken out of service for testing, a simulated signal is injected in place of the field instrument signal. The process equipment for the channel in test is then tested, verified, and calibrated.

The Allowable Values listed in Table 3.3.5-1 are based on the methodology described in FSAR, Chapter [14] (Ref. 4), which incorporates all of the known uncertainties applicable for each channel. The magnitudes of these uncertainties are

(continued)

BASES

BACKGROUND

~~Trip Setpoints and Allowable Values (continued)~~

~~factored into the determination of each trip setpoint. All field sensors and signal processing equipment for these channels are assumed to operate within the allowances of these uncertainty magnitudes.~~ (10)

~~[Reviewer's Note: The ESFAS LCOs in the BWOG Standard Technical Specifications are based on a system representative of the Crystal River Unit 3 design.] As discussed earlier, this arrangement involves measurement channels shared among all actuation functions, with separate actuation logic channels for each actuated component. In this arrangement, multiple components are affected by each instrumentation channel failure, but a single automatic actuation logic failure affects only one component. The organization of BWOG STS ESFAS LCOs reflects the described logic arrangement by identifying instrumentation requirements on an instrumentation channel rather than on a protective function basis. This greatly simplifies delineation of ESFAS LCOs. Furthermore, the LCO requirements on instrumentation channels, automatic actuation logics, and manual initiation are specified separately to reflect the different impact each has on ESFAS OPERABILITY.~~ edit

APPLICABLE SAFETY ANALYSES

The following ESFAS Functions have been assumed within the accident analyses.

High Pressure Injection

The ESFAS actuation of HPI has been assumed for core cooling in the LOCA analysis and is ~~credited with~~ available for boron addition in the SLB analysis. edit

Low Pressure Injection

The ESFAS actuation of LPI has been assumed for large break LOCAs.

(continued)

The MSLB analysis also credits ESAS actuation of RB Cooling and RB Spray.

BASES

APPLICABLE SAFETY ANALYSES (continued)

Reactor Building Spray, Reactor Building Cooling, and Reactor Building Isolation

The ESFAS actuation of the RB coolers and RB Spray have been credited in RB analysis for LOCAs, ~~both for RB performance and equipment environmental qualification pressure and temperature envelope definition~~. Accident dose calculations have credited RB Penetration Room Ventilation, RB Isolation, and RB Spray.

10

Emergency ~~Diesel Generator Start~~ Power

The ESFAS initiated EDG Start ~~has been assumed in the LOCA analysis~~ and ES electrical equipment alignment have been included in the design to ensure that emergency power is available throughout the limiting LOCA scenarios. (Ref. 2)

The small and large break LOCA analyses assume a conservative ~~38 second~~ delay time for the actuation of HPI and LPI ~~in FSAR Chapter 14 (Ref. 8)~~. This delay time includes allowances for EDG starting, EDG loading, Emergency Core Cooling Systems (ECCS) pump starts, and valve ~~openings~~ alignment. Similarly, the RB Cooling, RB Isolation, and RB Spray have been analyzed with delays appropriate for the entire system analyzed. ~~Typical values used in the analysis are 35 seconds for RB Cooling, 60 seconds for RB Isolation, and 56 seconds for RB Spray.~~ 10

Accident analyses rely on automatic ESFAS actuation for protection of the core temperature and containment pressure limits and for limiting off site dose levels following an accident. These include LOCA, SLB, and ~~feedwater line break~~ other events that result in RCS inventory reduction or severe loss of RCS cooling. 10

INSERT B3.3-52A

~~The ESFAS channels satisfy Criterion 3 of the NRC Policy Statement.~~ 14

LCO

The LCO requires three ~~channels of~~ ESFAS analog instrument~~ation~~ channels for each Parameter in Table 3.3.5-1 to be OPERABLE ~~in each ESFAS train~~. Failure of any analog instrument renders the affected channel(s) inoperable and reduces the reliability of the affected Functions. 5

(continued)

<INSERT B3.3-52A>

The ESAS instrumentation satisfies Criterion 3 of 10 CFR 50.36 (Ref. 3) for operation in MODE 1. There are no specific safety analyses for operation in MODES 2, 3 and 4. However, industry operating experience has identified the ESAS instrumentation as significant to public health and safety during these operating conditions. Therefore, the ESAS instrumentation satisfies Criterion 4 of 10 CFR 50.36 for operation in MODES 2, 3 and 4.

Provided

BASES

LCO (continued)

Only the Allowable Value is specified for each ~~E~~SFAS Function in the LCO. ~~Nominal~~ ~~t~~rip setpoints are ~~specified~~ in the ~~unit specific setpoint calculations~~. The ~~nominal~~ trip setpoints are selected to ensure the setpoints measured by CHANNEL FUNCTIONAL TESTS do not exceed the Allowable Value if the bistable is performing as required. ~~Operation with a trip setpoint less conservative than the nominal trip setpoint, but within its Allowable Value, is acceptable provided that operation and testing are consistent with the assumptions of the unit specific setpoint calculations.~~ Each Allowable Value specified is more conservative than ~~the~~ analytical limit assumed in the safety analysis to account for instrument uncertainties appropriate to the trip Parameter. ~~These uncertainties are defined in the "Unit Specific Setpoint Methodology"~~ (Ref. 3).

Calibration procedures

equal to or

any

Guidance used to calculate the uncertainties associated with the trip setpoints is provided in Instrument Loop Error Analysis Setpoint Methodology, Design Guide ID6-001

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10

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The ~~Allowable~~ Values for bypass removal functions are stated in the Applicable MODES or Other Specified Condition column of Table 3.3.5-1.

Three ESFAS instrument~~ation~~ channels shall be OPERABLE ~~in each ESFAS train~~ to ensure that a single failure in one channel will not result in loss of the ability to automatically actuate the required safety systems.

analog

5

~~The bases for the LCO on ESFAS Parameters include the following.~~

edit

Reactor Coolant System Pressure

Three channels ~~each~~ of RCS Pressure—Low ~~and RCS Pressure—Low Low~~ are required OPERABLE ~~in each train~~. Each channel includes a sensor, trip bistable, bypass bistable, bypass relays, output relays, ~~and block timers~~. ~~The analog portion of each pressure channel is common to both trains of both RCS Pressure Parameters. Therefore, failure of one analog channel renders one channel of the low pressure and low low pressure Functions in each train inoperable. The bistable portions of the channels are Function and train specific. Therefore, a bistable failure renders only one Function in one train inoperable.~~ Failure of a bypass bistable or bypass circuitry, such that ~~a trip~~ channel cannot be bypassed, does not render the channel inoperable. ~~Output relays and block timer relays are train specific but may be shared among Parameters. Therefore, output or block~~

4

5

and

10

an analog instrument

5

10

<INSERT B3.3-53A>

5

(continued)

<INSERT B3.3-53A>

since the channel is still capable of performing its safety function, i.e., this is not a safety related bypass function.

BASES

LCO

Reactor Coolant System Pressure (continued)

timer relay failure renders all affected Functions in one train inoperable. (5)

1. Reactor Coolant System Pressure—Low Setpoint

The RCS Pressure—Low Setpoint is based on HPI actuation for small break LOCAs. The setpoint ensures that the HPI will be actuated at a pressure greater than or equal to the value assumed in accident analyses plus the instrument uncertainties. The maximum value assumed for the setpoint of the RCS Pressure—Low trip of HPI in safety analyses is 1480 psig. The setpoint for the low RCS and Allowable Value of ≥ [1600] psig for the low pressure parameter is selected to ensure actuation occurs when actual RCS pressure is above 1480 psig. The RCS Pressure instrumentation must function while subject to the severe environment created by a LOCA. Therefore, the trip setpoint Allowable Value accounts for severe environment induced errors.

<INSERT B 3.3-54A> (10)

To ensure the RCS Pressure—Low trip is not bypassed when required to be OPERABLE by the safety analysis, each channel's bypass removal bistable must be set with an Allowable Value of ≤ [1800] psig. The bypass removal does not need to function for accidents initiated from RCS Pressures below the bypass removal setpoint. Therefore, the bypass removal setpoint Allowable Value need not account for severe environment induced errors.

2. Reactor Coolant System Pressure—Low Low Setpoint

The RCS Pressure—Low Low Setpoint LPI actuation occurs in sufficient time to ensure LPI flow prior to the emptying of the core flood tanks during a large break LOCA. The Allowable Value of ≥ [400] psig ensures sufficient overlap of the core flood tank flow and the LPI flow to keep the reactor vessel downcomer full during a large break LOCA. The RCS Pressure instrumentation must function while subject to the severe environment created by a LOCA. Therefore, the trip setpoint Allowable Value accounts for severe environment induced errors. (4)

(continued)

<INSERT B3.3-54A>

The trip setpoints are the nominal values at which the bistables are set. For the RCS Pressure—Low, the limiting safety analysis assumes the HPI, LPI, EFIC EFW, ES electrical alignment, and two subsets of RB isolation actuate at ≥ 1520 psig (≥ 1535 psia). The Allowable Value of ≥ 1585 psig includes considerations for instrumentation error and an allowance for margin. Allowances for instrument drift and additional margin are included in the trip setpoint.

Guidance used to calculate the uncertainties associated with the trip setpoints is provided in Instrument Loop Error Analysis and Setpoint Methodology Manual, Design Guide, IDG-001 (Ref. 4). The explicit uncertainties associated with each setpoint are addressed in the individual design calculations or calibration procedures. Setpoints in accordance with the Allowable Value in conjunction with the LCOs and administrative controls ensure that the consequences of DBAs will be acceptable, providing the unit is operated from within the LCOs at the onset of the DBA and the equipment functions as analyzed. An analog instrument channel is inoperable if its actual trip setpoint is not within its required Allowable Value.

BASES

LCO

2. Reactor Coolant System Pressure—Low Low Setpoint (continued)

To ensure the RCS Pressure—Low Low trip is not bypassed when assumed OPERABLE by the safety analysis, each channel's bypass removal bistable must be set with an Allowable Value of ≤ [900] psig. The bypass removal does not need to function for accidents initiated by RCS Pressure below the bypass removal setpoint. Therefore, the bypass removal setpoint Allowable Value need not account for severe environment induced errors. (4)

Reactor Building Pressure

and RB Pressure—High High

Three channels each of RCS Pressure—Low and RB Pressure—High are required to be OPERABLE in each train. (5) Each channel includes a pressure switch, bypass relays, and output relays. The high pressure channels also include block timers. Each pressure switch is Function and train specific, so there are 12 pressure switches total. Therefore, a pressure switch renders only one Function in one train inoperable. Output relays and block timer relays are train specific but may be shared among Parameters. Therefore, output or block timer relay failure renders all affected Functions in one train inoperable.

<INSERT B3.3-55A>

The RB Pressure switches may be subjected to high radiation conditions during the accidents that they are intended to mitigate. The sensor portion of the switches is also exposed to the steam environment present in the RB following a LOCA or high energy line break. Therefore, the trip setpoint Allowable Value accounts for measurement errors induced by these environments.

3. Reactor Building Pressure—High Setpoint

The RB Pressure—High Setpoint Allowable Value ≤ [5] psig was selected to be low enough to detect a rise in RB Pressure that would occur due to a small break LOCA, thus ensuring that the RB high pressure actuation of the safety systems will occur for a wide spectrum of break sizes. The trip setpoint also causes the RB coolers to shift to emergency mode to prevent damage to the cooler fans due to the increase (10)

(continued)

<INSERT B3.3-55A>

The trip setpoints are the nominal values at which the bistables are set. Credit is taken in the safety analyses for RB Pressure—High trip for the actuation of selected systems. The safety analyses for reactor building performance and equipment environmental qualification (pressure and temperature envelope definition) conservatively assume the RB cooling is not initiated until well beyond the expected actual automatic actuation time frame. Therefore, no additional consideration of the instrumentation uncertainties is warranted.

Credit is taken in the safety analyses for RB Pressure—High High trip for the actuation of selected systems. The safety analyses for reactor building performance and equipment environmental qualification (pressure and temperature envelope definition) conservatively assumes the RB spray is not initiated until well beyond the expected actual automatic actuation time frame. Therefore, no additional consideration of the instrumentation uncertainties is warranted.

Therefore, the bistable is considered to be properly adjusted when the "as left" value is consistent with the identified Allowable Value, i.e., for this parameter the trip setpoint and the Allowable Value are the same. Guidance used to calculate the uncertainties associated with the trip setpoints is provided in Instrument Loop Error Analysis and Setpoint Methodology Manual, Design Guide, IDG-001 (Ref. 4). Setpoints in accordance with the Allowable Value ensure that the consequences of DBAs will be acceptable, providing the unit is operated from within the LCOs at the onset of the DBA and the equipment functions as analyzed. An analog instrument channel is inoperable if its actual trip setpoint is not within its required Allowable Value.

BASES

LCO

1. Reactor Building Pressure—High Setpoint (continued)

in the density of the air steam mixture present in the containment following a LOCA.

2. Reactor Building Pressure—High High Setpoint

The RB Pressure—High High Setpoint Allowable Value ≤ [30] psig was chosen to be high enough to avoid actuation during an SLB, but also low enough to ensure a timely actuation during a large break LOCA.

10

APPLICABILITY

analog channels of the following

Three ~~channels of~~ ESFAS instrument~~ation~~ for each Parameters ~~listed next~~ shall be OPERABLE. ~~in each ESFAS train.~~ 5

1. Reactor Coolant System Pressure—Low Setpoint

at or

The RCS Pressure—Low Setpoint actuation Parameter shall be OPERABLE during operation above ~~1800~~ 1750 psig. This requirement ensures the capability to automatically actuate safety systems and components during conditions indicative of a LOCA or secondary unit overcooling. Below ~~1800~~ 1750 psig, the low RCS Pressure actuation Parameter can be bypassed to avoid actuation during normal unit cooldowns when safety systems actuations are not required. edit

The allowance for the bypass is consistent with the transition of the unit to a lower energy state, ~~providing greater~~ where there is more margin~~s~~ to safety limits. The unit response to any event, given that the reactor is already tripped, will be less severe and allows ~~sufficient~~ more time for operator action to provide manual safety system actuations than in higher energy states. This is even more appropriate during unit heatups when the primary system and core energy content is low, prior to power operation. 10 edit

In MODES 5 and 6, there is ~~adequate~~ more time for the operator to evaluate unit conditions and respond by manually starting individual systems, pumps, and other equipment to mitigate the consequences of an abnormal condition or accident than in higher MODES. ~~Plant~~ RCS pressure and temperature edit edit

(continued)

1

BASES

APPLICABILITY 1. Reactor Coolant System Pressure—Low Setpoint (continued)

are very low, and many ESF components are administratively locked out or otherwise prevented from actuating to prevent inadvertent overpressurization of unit systems.

controlled

edit

2. Reactor Coolant System Pressure—Low Low Setpoint

~~The RCS Pressure—Low Low Setpoint actuation Parameter shall be OPERABLE during operation above [900] psig. This requirement ensures the capability to automatically actuate safety systems and components during conditions indicative of a LOCA or secondary unit overcooling. Below [900] psig, the low low RCS Pressure actuation Parameter can be bypassed to avoid actuation during normal unit cooldowns when safety system actuations are not required.~~

~~The allowance for the bypass is consistent with the transition of the unit to a lower energy state, providing greater margins to safety limits. The unit response to any event, given that the reactor is already tripped, will be less severe and allows sufficient time for operator action to provide manual safety system actuations. This is even more appropriate during unit heatups when the primary system and core energy content is low, prior to power operation.~~

~~In MODES 5 and 6, there is adequate time for the operator to evaluate unit conditions and respond by manually starting individual systems, pumps, and other equipment to mitigate the consequences of an abnormal condition or accident. Plant pressure and temperature are very low, and many ESF components are administratively locked out or otherwise prevented from actuating to prevent inadvertent overpressurization of unit systems.~~

4

(continued)

H1

BASES

APPLICABILITY (continued)

2
3.~~4~~. <u>Reactor Building Pressure—High and Reactor Building Pressure—High High Setpoints</u>

The RB Pressure—High and RB Pressure—High High actuation Functions of ESFAS shall be OPERABLE in MODES 1, 2, 3, and 4 when the potential for a HELB exists. ~~In MODES 5 and 6, the unit conditions are such that there is insufficient energy in the primary and secondary systems to raise the containment pressure to either the RB Pressure—High or RB Pressure—High High Setpoints. Furthermore, in~~ MODES 5 and 6, there is ~~adequate~~ more time for the operator to evaluate unit conditions and respond by manually starting individual systems, pumps, and other equipment to mitigate than in higher MODES the consequences of an abnormal condition or accident. Plant pressure and temperature are very low and many ESF components are administratively ~~locked out~~ controlled or otherwise prevented from actuating to prevent inadvertent overpressurization of unit systems.

edit

ACTIONS

Required Actions A and B apply to ~~all~~ the ESFAS instrumentation Parameters listed in Table 3.3.5-1.

A Note has been added to the ACTIONS indicating separate Condition entry is allowed for each Parameter.

H5

If ~~a~~ an analog instrument channel's trip setpoint is found nonconservative with respect to the Allowable Value, or the transmitter, instrument loop, signal processing electronics, or ESFAS bistable is found inoperable, then all affected functions provided by that channel should be declared inoperable and the unit must enter the Conditions for the particular protection Parameter affected.

~~When the number of inoperable channels in a trip Parameter exceeds those specified, then the unit is outside the safety analysis. Therefore, LCO 3.0.3 shall be immediately entered if applicable in the current MODE of operation.~~ 2

(continued)

1

BASES

ACTIONS (continued)

A.1

analog instrument

Condition A applies when one channel becomes inoperable in one or more Parameters. If one ESFAS channel is inoperable, placing it in a tripped condition leaves the system in a one-out-of-two condition for actuation. Thus, if another analog instrument channel were to fail, the ESFAS instrumentation could still perform its actuation functions. This action is completed when all of the affected output relays and block timers are tripped. This can normally be accomplished by tripping the affected bistables, or tripping the individual output relays and block timers. [At this unit, the specific output relays associated with each ESFAS instrumentation channel are listed in the following document:]

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The 1 hour Completion Time is sufficient time to perform the Required Action.

B.1, B.2.1, B.2.2, and B.2.3

INSERT B3.3-59A

Condition B applies when Required Action A.1 is not met within the required Completion Time. If Required Action A.1 cannot be met within the required Completion Time, the unit must be brought to a MODE in which the LCO does not apply. To achieve this status, the unit must be brought to at least MODE 3 within 6 hours and, for the RCS Pressure—Low Setpoint, to < [1800] psig, for the RCS Pressure—Low Low Setpoint, to < [900] psig, and for the RB Pressure High Setpoint and High High Setpoint, to MODE 5 within 36 hours.

2

edit

4

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

The All ESFAS Parameters listed in Table 3.3.5-1 are subject to CHANNEL CHECK, CHANNEL FUNCTIONAL TEST, and CHANNEL CALIBRATION, and response time testing. The operational bypasses associated with each ESFAS instrumentation channel are also subject to these SRs to ensure OPERABILITY of the ESFAS instrumentation channel.

3

10

(continued)

<INSERT B3.3-59A>

Condition B applies when Required Action A.1 and its associated Completion Time are not met, or when one or more parameters have more than one analog instrument channel inoperable. If Condition B applies, the unit must be brought to a condition in which the LCO does not apply. To achieve this status, the unit must be brought to at least MODE 3 within 6 hours. Additionally, for the RCS Pressure—Low parameter, the unit must be brought to < 1750 psig within 36 hours, and for the RB Pressure—High and High High parameters, the unit must be brought to MODE 5 within 36 hours.

1

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.3.5.1

provides reasonable assurance for prompt identification of

Performance of the CHANNEL CHECK every 12 hours ~~ensures that~~ a gross failure of instrumentation ~~has not occurred~~. A CHANNEL CHECK is normally a comparison of the parameter indicated on one channel to a similar parameter on other channels. It is based on the assumption that instrument channels monitoring the same parameter should read approximately the same value. Significant deviations between the two instrument channels could be an indication of excessive instrument drift in one of the channels or of something even more serious. CHANNEL CHECK will detect gross channel failure; therefore, it is key in verifying that the instrumentation continues to operate properly between each CHANNEL CALIBRATION. edit

factors including

Agreement criteria are determined by the unit staff, based on a combination of ~~the~~ channel instrument uncertainties, ~~including isolation, indication, and readability~~. If a channel is outside the criteria, it may be an indication that the transmitter or the signal processing equipment has drifted outside its limit. ~~If the channels are normally off scale during times when surveillance is required, the CHANNEL CHECK will only verify that they are off scale in the same direction. Off scale low current loop channels are verified to be reading at the bottom of the range and not failed downscale.~~ 10 11

The Frequency ~~about once every shift~~ is based on operating experience that demonstrates channel failure is rare. Since the probability of two random failures in redundant channels in any 12 hour period is extremely low, the CHANNEL CHECK minimizes the chance of loss of protective function due to failure of redundant channels. The CHANNEL CHECK supplements less formal, but potentially more frequent, checks of channel ~~operability~~ OPERABILITY during normal operational use of the displays associated with the LCO's required channels. edit edit

SR 3.3.5.2

~~A Note defines a channel as being OPERABLE for up to 8 hours while bypassed for Surveillance testing provided the remaining two ESFAS channels are OPERABLE or tripped. The Note allows channel bypass for testing without defining it~~ 12

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.3.5.2 (continued)

~~as inoperable, although during this time period it cannot initiate ESFAS. This allowance is based on the inability to perform the Surveillance in the time permitted by the Required Actions. Eight hours is the average time required to perform the Surveillance. It is not acceptable to routinely remove channels from service for more than 8 hours to perform required Surveillance testing.~~ 12

A CHANNEL FUNCTIONAL TEST is performed on each required ~~ESFAS~~ analog instrument channel to ensure the entire channel will perform the intended functions. Any setpoint adjustment shall be consistent with the assumptions of the ~~current unit specific~~ setpoint ~~analysis.~~ calculations. 5 10

The Frequency of 31 days is based on unit operating experience, with regard to channel OPERABILITY and drift, which demonstrates that failure of more than one channel of a given function in any 31 day interval is a rare event.

SR 3.3.5.3

CHANNEL CALIBRATION is a complete check of the analog instrument channel, including the sensor. The test verifies that the channel responds to a measured parameter within the necessary range and accuracy. CHANNEL CALIBRATION leaves the channel adjusted to account for instrument drift to ensure that the analog instrument channel remains ~~operational~~ OPERABLE between successive tests. CHANNEL CALIBRATION shall find that measurement errors and bistable setpoint errors are within the assumptions of the ~~unit specific~~ setpoint ~~analysis.~~ calculations. CHANNEL CALIBRATIONS must be performed consistent with the assumptions of the ~~unit specific~~ setpoint ~~analysis.~~ 5 10

This Frequency is justified by the assumption of ~~an~~ at least ~~[18]~~ month calibration interval to determine the magnitude of equipment drift in the setpoint ~~analysis.~~ calculations. edit 10

~~SR 3.3.5.4~~

~~SR 3.3.5.4 ensures that the ESFAS actuation channel response times are less than or equal to the maximum times assumed in the accident analysis. The response time values are the~~ 3

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.3.5.4 (continued)

~~maximum values assumed in the safety analyses. Individual component response times are not modeled in the analyses. Response time testing acceptance criteria for this unit are included in Reference 1. The analyses model the overall or total elapsed time from the point at which the parameter exceeds the actuation setpoint value at the sensor to the point at which the end device is actuated. Thus, this SR encompasses the automatic actuation logic components covered by LCO 3.3.7 and the operation of the mechanical ESF components.~~

~~Response time tests are conducted on an [18] month STAGGERED TEST BASIS. Testing of the final actuation devices, which make up the bulk of the response time, is included in the testing of each channel. Therefore, staggered testing results in response time verification of these devices every [18] months. The 18 month test Frequency is based on unit operating experience, which shows that random failures of instrumentation components causing serious response time degradation but not channel failure are infrequent occurrences.~~ 3

REFERENCES

1. ~~F~~SAR, Chapter ~~[7]~~.

3 ~~2~~. 10 CFR 50.~~49~~ 36.

4 ~~3~~. ~~"Unit Specific Setpoint Methodology."~~

2 ~~4~~. FSAR, Chapter ~~[14]~~ 9 and Chapter 3A.

edit

1. Instrument Loop Error Analysis and Setpoint Methodology Manual, Design Guide, IDG-001.

1

B 3.3 INSTRUMENTATION

B 3.3.6 Engineered ~~Safety Feature~~ Safeguards Actuation System (ES~~F~~AS) Manual Initiation

BASES

BACKGROUND

The ES~~F~~AS manual initiation capability allows the operator to actuate ES~~F~~AS Functions from the ~~main~~ control room in the absence of any other initiation condition. ~~Manually actuated Functions include High Pressure Injection, Low Pressure Injection, Reactor Building (RB) Cooling, RB Spray, RB Isolation, and Control Room Isolation.~~ This ES~~F~~AS manual initiation capability is provided in the event the operator determines that an ES~~F~~AS Function is needed and has not been automatically actuated. Furthermore, the ES~~F~~AS manual initiation capability allows operators to rapidly initiate Engineered ~~Safety Feature (ESF)~~ Safeguards (ES) Functions if the trend of unit parameters indicates that ES~~F~~ actuation will be needed.

edit

10

LCO 3.3.6 covers only the system level manual initiation of these Functions. LCO 3.3.5, "Engineered ~~Safety Feature~~ Safeguards Actuation System (ES~~F~~AS) Instrumentation," and LCO 3.3.7, "Engineered ~~Safety Feature~~ Safeguards Actuation System (ES~~F~~AS) ~~Automatic~~ Actuation Logic," provide requirements on the portions of the ES~~F~~AS that automatically initiate the Functions described earlier.

9

The ES~~F~~AS manual initiation Function relies on the OPERABILITY of the ~~automatic~~ digital actuation logic channels (LCO 3.3.7) ~~for each component~~ to perform the actuation of the systems. A manual trip push button is provided on ~~the ESF panel of the~~ control room console for each of the ~~levels of protection for each actuation~~ (9) digital actuation logic channels. Operation of the push button energizes relays whose contacts perform a logical "OR" function with the ~~matrices of the~~ automatic actuation~~, except for the matrices which are part of the ESF buses loading sequence. Manual actuation of the ESF buses loading sequence is made by de-energizing the timed output relays. The power supply for the manual trip relays is taken from the station batteries. Different batteries are used for the two actuations.~~

9 5 10

The ES~~F~~AS manual ~~initiation~~ from channel is defined as the console switch and the instrumentation ~~between~~ to, but not including, the console switch ~~and~~ the ~~automatic~~ digital actuation logic channels, which actuate~~s~~ the end devices. Other means of manual initiation, such as controls for individual ES~~F~~ devices, may be available in the control room and other

5 9

(continued)

BASES

BACKGROUND (continued)

unit locations. These alternative means are not required by this LCO, nor may they be credited to fulfill the requirements of this LCO.

APPLICABLE SAFETY ANALYSES

The ESFAS, in conjunction with the actuated equipment, provides protective functions necessary to mitigate Design Basis Accidents, specifically, the loss of coolant accident (LOCA) and steam line break events.

reactor building (RB) DBA

as a backup to mitigate the

The ESFAS manual initiation ensures that the control room operator can rapidly initiate ESF Functions ~~at any time~~. The manual initiation trip Function is required as a backup to automatic trip functions and allows operators to initiate ESFAS whenever any parameter is rapidly trending toward its trip setpoint. ~~Furthermore, the ESFAS manual initiation may be specified in operating procedures for verification that ESF systems are running.~~ (13)

~~The ESFAS manual initiation functions satisfy Criterion 3 of the NRC Policy Statement.~~ (14)

<INSERT B 3.3-64A>

LCO

Two ESFAS manual initiation channels of each ESFAS Function shall be OPERABLE whenever conditions exist that could require ESF protection of the reactor or RB. Two OPERABLE channels ensure that no single random failure will prevent system level manual initiation of any ESFAS Function. The ESFAS manual initiation Function allows the operator to initiate protective action prior to automatic initiation or in the event the automatic initiation does not occur.

<INSERT B 3.3-64B> ⊢(9)

APPLICABILITY

MODES 3, and

The ESFAS manual initiation Functions shall be OPERABLE in MODES 1, 2, and 3, and in MODE 4 when the associated ~~engineered safeguard~~ ES equipment is required to be OPERABLE. The manual initiation channels are required because ESF Functions are designed to provide protection in these MODES. while in In MODES 5 and 6, ESFAS initiates systems that are either reconfigured ~~or disabled~~ for ~~shutdown cooling operation~~ decay heat removal. Accidents in these MODES are slow to develop and would be mitigated by manual operation of individual components. Adequate time is available to evaluate unit conditions and ⊢(7) edit

(continued)

<INSERT B3.3-64A>

Operating experience has shown the ESAS manual initiation function to be significant to public health and safety, and therefore satisfy Criterion 4 of 10 CFR 50.36 (Ref. 1).

<INSERT B3.3-64B>

The ESAS is divided into five Functions actuated by ten manual initiation channels as indicated in the following table:

Function	Associated Channels
High Pressure Injection	1 & 2
Low Pressure Injection	3 & 4
RB Cooling	5 & 6
RB Spray	7 & 8
Spray Additive	9 & 10

The ESAS High Pressure Injection (HPI) Function is actuated by ESAS Manual Initiation channels 1 and 2 and includes the following system actuations: HPI, a subset of reactor building (RB) isolation valves, diesel generators, and ES electrical alignment.

The ESAS Low Pressure Injection (LPI) Function is actuated by ESAS Manual Initiation channels 3 and 4 and includes the following system actuations: LPI, a subset of RB isolation valves, and emergency feedwater (EFW) through an ESAS signal provided to the Emergency Feedwater Isolation and Control (EFIC) System.

The ESAS RB Cooling Function is actuated by ESAS Manual Initiation channels 5 and 6 and includes the following system actuations: RB cooling, a subset of RB isolation valves, and RB penetration room ventilation system.

The ESAS RB Spray Function is actuated by ESAS Manual Initiation channels 7 and 8 and includes the following system actuations: RB spray.

The ESAS Spray Additive Function is actuated by ESAS Manual Initiation channels 9 and 10 and includes the following system actuations: spray additive.

BASES

APPLICABILITY (continued)	to respond by manually operating the ES~~F~~ components, if required.

ACTIONS

A Note has been added to the ACTIONS indicating separate Condition entry is allowed for each ES~~F~~AS manual initiation Function.

A.1

Condition A applies when one manual initiation channel of one or more ES~~F~~AS Functions becomes inoperable. Required Action A.1 must be taken to restore the channel to OPERABLE status within the next 72 hours. The Completion Time of 72 hours is based on unit operating experience and administrative controls, which provide alternative means of ES~~F~~AS Function initiation via individual component controls. The 72 hour Completion Time is generally consistent with the allowed outage time for the safety systems actuated by ES~~F~~AS. edit

B.1 and B.2

~~Required Action B.1 and Required Action B.2 apply if Required Action A.1 cannot be met within the required Completion Time.~~ If Required Action A.1 ~~cannot be met within the required~~ are not met and the associated Completion Time~~,~~ the unit must be ~~brought to a MODE in~~ which the LCO does not apply. To achieve this status, the unit must be brought to at least MODE 3 within 6 hours and to MODE 5 within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required MODES from full power conditions in an orderly manner and without challenging unit systems. edit

SURVEILLANCE REQUIREMENTS

SR 3.3.6.1

This SR requires the performance of a CHANNEL FUNCTIONAL TEST of the ES~~F~~AS manual initiation. This test verifies that the initiating circuitry is OPERABLE and will actuate the ~~end device (i.e., pump, valves, etc.)~~ digital actuation logic channels. The ~~[~~18~~]~~ month Frequency is based on the need to perform this Surveillance 10 9 5

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.3.6.1 (continued)

under the conditions that apply during a unit outage and the potential for an unplanned transient if the Surveillance were performed with the reactor at power. This Frequency is demonstrated to be sufficient, based on operating experience, which shows these components usually pass the Surveillance when performed on the [18] month Frequency.

edit

REFERENCES

~~None.~~ 1. 10CFR 50.36.

1

9

B 3.3 INSTRUMENTATION

B 3.3.7 Engineered Safety Feature Safeguards Actuation System (ESFAS) Automatic Actuation Logic

BASES

BACKGROUND

INSERT B3.3-67A

~~The automatic actuation logic channels of ESFAS are defined as the logic between the buffers of the sensing channels and the controllers that actuate ESFAS equipment. Each of the components actuated by the ESFAS Functions has an associated automatic actuation logic matrix. If two-out-of-three ESFAS instrumentation channels indicate a trip, or system level manual initiation occurs, the automatic actuation logic is activated and the associated component is actuated.~~ 9 edit

The purpose of requiring OPERABILITY of the ESFAS automatic digital actuation logic channels is to ensure that the Functions of the ESFAS can be automatically initiated in the event of an accident. Automatic actuation of some Functions is necessary to prevent the unit from exceeding the Emergency Core Cooling Systems (ECCS) limits in 10 CFR 50.46 (Ref. 1). It should be noted that OPERABLE automatic digital actuation logic channels alone will not ensure that each Function can be activated; the analog instrumentation channels and actuated equipment associated with each Function must also be OPERABLE to ensure that the Functions can be automatically initiated during an accident. 5 9 9 5

LCO 3.3.7 covers only the automatic digital actuation logic channels that initiates these Functions. LCO 3.3.5, "Engineered Safety Safeguards Feature Actuation System (ESFAS) Instrumentation," and LCO 3.3.6, "Engineered Safety Feature Actuation System (ESFAS) Manual Initiation," provide requirements on the analog instrumentation and manual initiation channels that input to the automatic digital actuation logic channels. 5 9

The ESFAS, in conjunction with the actuated equipment, provides protective functions necessary to mitigate Design Basis Accidents (DBAs), specifically, the loss of coolant accident (LOCA) and steam line break (SLB) events. The ESFAS relies on the OPERABILITY of the automatic digital actuation logic for each component to perform the actuation of the selected systems. 5 9

The small and large break LOCA analyses assume a conservative 35 second delay time for the actuation of high pressure injection (HPI) and low pressure injection (LPI) in 10

(continued)

<INSERT B3.3-67A>

The digital actuation logic channels of ESAS are defined as the instrumentation between, but not including, the buffers of the analog instrument channels and the unit controls that actuate ESAS equipment. Each of the components actuated by the ESAS Functions is associated with one or more digital actuation logic channels. If two-out-of-three ESAS analog instrument channels indicate a trip, or if channel level manual initiation occurs, the digital actuation logic channel is activated and the associated equipment is actuated.

1 9

BASES

alignment

BACKGROUND (continued)

BAW-10103A, Rev. 3 (Ref. 2). This delay time includes allowances for ~~emergency~~ diesel generator (EDG) starts, EDG loading, ECCS pump starts, and valve ~~openings~~. Similarly, the reactor building (RB) Cooling, RB Isolation, and RB Spray have been analyzed with delays appropriate for the entire system. 10

~~Typical values used in the analyses are 35 seconds for RB Cooling, 60 seconds for RB Isolation, and 58 seconds for RB Spray.~~ 10

Safeguards

The ESFAS automatic initiation of Engineered ~~Safety Feature~~ (ESF) Functions to mitigate accident conditions is assumed in the DBA analysis and is required to ensure that consequences of analyzed events do not exceed the accident analysis predictions. Automatically actuated features include HPI, LPI, RB Cooling, RB Spray, RB Spray Additive and RB Isolation.

~~The ESFAS LCOs in the BWOG Standard Technical Specifications (STS) are based on a system representative of the Crystal River Unit 3 design. As discussed earlier, this arrangement involves measurement channels shared among all actuation functions, with separate actuation logic channels for each actuated component. In this arrangement, multiple ESF components are affected by a measurement channel failure, but a single automatic actuation logic failure affects only one component. The organization of BWOG STS ESFAS LCOs reflect the described logic arrangement by linking actions for automatic actuation logic failures directly to the actions for the affected ESF component. The overall philosophy is that if an automatic actuation logic fails, the affected component is put into its engineered safeguard configuration. This action eliminates the need for the automatic actuation logic. If the affected component cannot be placed in its engineered safeguard configuration, actions are taken to address the inoperability of the supported system component. This greatly simplifies delineation of ESFAS LCOs. Furthermore, the LCO requirements on instrumentation channels, automatic actuation logics, and manual initiation are specified separately to reflect the different impact each has on ESFAS OPERABILITY.~~ edit

(continued)

(1) (9)

BASES (continued)

APPLICABLE SAFETY ANALYSES

Accident analyses rely on automatic ES~~F~~AS actuation for protection of the core and RB and for limiting off site dose levels following an accident. ~~These include LOCA, SLB, and feedwater line break events that result in Reactor Coolant System (RCS) inventory reduction or severe loss of RCS cooling.~~ (10) The ~~automatic~~ digital actuation logic is an integral part of the ES~~F~~AS. (5) (9)

INSERT B3.3-69A

~~The ESFAS automatic actuation logics satisfy Criterion 3 of the NRC Policy Statement.~~ (14)

LCO

The digital ~~automatic~~ actuation logic ~~matrix for each component actuated by the ESFAS is~~ channels are required to be OPERABLE whenever conditions exist that could require ES~~F~~ protection of the reactor or the RB. This ensures automatic initiation of the ES~~F~~ required to mitigate the consequences of accidents. (9) (5)

INSERT B3.3-69B (9)

APPLICABILITY

The digital ~~automatic~~ actuation logic ~~Function~~ channels shall be OPERABLE in MODES 1, ~~2,~~ and ~~3, and in MODE~~ 4 and in MODES 3 and when the associated ES ~~engineered safeguard~~ equipment is required to be OPERABLE, because ES~~F~~ Functions are designed to provide protection in these MODES. Automatic actuation in MODE 5 or 6 is not required because the systems initiated by the ES~~F~~AS are either reconfigured ~~or disabled~~ for ~~shutdown cooling~~ decay heat removal operation. Accidents in these MODES are slow to develop and would be mitigated by manual operation of individual components. ~~Adequate~~ Time is available to evaluate unit conditions and respond by manually operating the ES~~F~~ components, if required. (5) (7) (10) (10)

ACTIONS

A Note has been added to the ACTIONS indicating separate Condition entry is allowed for each ES~~F~~AS ~~automatic~~ digital actuation logic ~~matrix~~ channel. (5) (9)

A.1 and A.2

When one or more ~~automatic~~ digital actuation logic ~~matrices~~ channels are inoperable, the associated component(s) can be placed in ~~its~~ their ES ~~engineered safeguard~~ configuration. Required Action A.1 is (5)

(continued)

<INSERT B3.3-69A>

The ESAS actuation logic satisfies Criterion 3 of 10 CFR 50.36 (Ref. 3) for operation in MODE 1. There are no specific safety analyses for operation in MODES 2, 3 and 4. However, industry operating experience has identified the ESAS actuation logic as significant to public health and safety during these operating conditions. Therefore, the ESAS actuation logic satisfies Criterion 4 of 10 CFR 50.36 for operation in MODES 2, 3 and 4.

<INSERT B3.3-69B>

The ESAS is divided into five Functions actuated by ten digital actuation logic channels as indicated in the following table:

Function	Associated Channels
High Pressure Injection	1 & 2
Low Pressure Injection	3 & 4
RB Cooling	5 & 6
RB Spray	7 & 8
Spray Additive	9 & 10

The ESAS HPI Function is actuated by ESAS digital actuation logic channels 1 and 2 and includes the following system actuations: HPI, a subset of RB isolation valves, DGs, and ES electrical alignment. Digital actuation logic channels 1 and 2 are actuated by two-out-of-three RCS Pressure—Low analog instrument channels, or two-out-of-three RB Pressure-High analog instrument channels.

The ESAS LPI Function is actuated by ESAS digital actuation logic channels 3 and 4 and includes the following system actuations: LPI, a subset of RB isolation valves, and EFW through an ESAS signal provided to EFIC. Digital actuation logic channels 3 and 4 are actuated by two-out-of-three RCS Pressure—Low analog instrument channels, or two-out-of-three RB Pressure—High analog instrument channels.

The ESAS RB Isolation and Cooling Function is actuated by ESAS digital actuation logic channels 5 and 6 and includes the following system actuations: RB cooling, a subset of RB isolation valves, and RB penetration room ventilation system. Digital actuation logic channels 5 and 6 are actuated by two-out-of-three RB Pressure—High analog instrument channels.

The ESAS RB Spray Function is actuated by ESAS digital actuation logic channels 7 and 8 and includes the following system actuations: RB spray. Digital actuation logic channels 7 and 8 are actuated by two-out-of-three RB Pressure—High High analog instrument channels.

The ESAS Spray Additive Function is actuated by ESAS digital actuation logic channels 9 and 10 and includes the following system actuations: spray additive. Digital actuation logic channels 9 and 10 are actuated by two-out-of-three RB Pressure—High High analog instrument channels.

BASES

ACTIONS

A.1 and A.2 (continued) digital channel 9

equivalent to the automatic actuation logic performing its safety function ahead of time. In some cases, placing the 5

~~component in its~~ engineered safeguard configuration would 10

ES

violate unit safety or operational considerations. In these cases, the component status should not be changed, but the supported system component must be declared inoperable. Conditions which would preclude the placing of a component in its engineered safeguard configuration include, but are 10

not limited to, violation of system separation, activation of fluid systems that could lead to thermal shock, ~~or~~ isolation of fluid systems that are normally functioning. The Completion Time of 1 hour is based on operating experience and reflects the urgency associated with the inoperability of a safety system component.

and actuation of components which would not return to their actuated condition upon restoration of electrical power.

Required Action A.2 requires entry into the Required Actions of the affected supported systems, since the true effect of automatic actuation logic failure is inoperability of the supported system. The Completion Time of 1 hour is based on operating experience and reflects the urgency associated with the inoperability of a safety system component.

digital channel

A combination of Required Actions A.1 and A.2 may be used for different components associated with an inoperable ESAS digital actuation logic channel. 15

SURVEILLANCE REQUIREMENTS

SR 3.3.7.1

SR 3.3.7.1 is the performance of a CHANNEL FUNCTIONAL TEST on a 31 day ~~STAGGERED TEST BASIS~~. The test demonstrates that ~~every~~ automatic actuation logic ~~associated with one of the two safety system trains~~ successfully performs the two-out-of-three logic combinations every 31 days. ~~All automatic actuation logics are thus retested every 62 days.~~ The test simulates the required one-out-of-three inputs to the logic circuit and verifies the successful operation of the automatic actuation logic. The Frequency is based on operating experience that demonstrates the rarity of more than one channel failing within the same 31 day interval. 8 9 5

Frequency each digital channel digital

~~Automatic actuation logic response time testing is incorporated into the response time testing required by LCO 3.3.5.~~ 3

(continued)

1
9

BASES (continued)

REFERENCES

1. 10 CFR 50.46.
2. ~~BAW-10103A, Rev. 3, July 1977.~~
3. 10CFR 50.36.

EDIT