ITS DISCUSSION OF DIFFERENCES
ITS Section 3.1: Reactivity Control Systems

1 NUREG 3.1.1, 3.1.4, 3.1.5, 3.1.8, & 3.1.9 - Incorporated TSTF-009.

2 NUREG 3.1.8 - The Frequency of NUREG SR 3.1.8.3 (ITS SR 3.1.8.3) was changed to specify "Within 8 hours prior to performance of PHYSICS TESTS at each testing plateau." This Frequency requires the nuclear overpower trip setpoint be verified prior to the onset of PHYSICS TESTS which ensures that the established LCO conditions are satisfied, with respect to the trip function. The requirement to perform these NUREG SRs with a Frequency of 8 hours is excessively restrictive and unduly burdensome on the operation of the unit. The short time frame in which the unit is expected to be conducting PHYSICS TESTS requiring the exception to one or more LCOs does not warrant the increased verification requirements. Further, these SRs provide a verification of RPS system performance at a Frequency significantly shorter than that required of the RPS when operating in MODE 1 at RATED THERMAL POWER (ref. NUREG 3.3.1). No basis exists to imply that the RPS trip function, or its calibration, would behave differently than that observed during power operation. The CTS does not contain a similar surveillance requirement. The ANO-1 current license basis does require that the high flux trip (nuclear overpower trip) setpoints are administratively set, as stated in the CTS Bases associated with CTS 3.5.2. The proposed change in Frequency is considered to be consistent with ANO-1's current practice, as allowed by the current license basis. The Bases were changed to reflect this Frequency. 3.1-12

NUREG 3.1.9 - The Frequency of NUREG SR 3.1.9.2. (ITS SR 3.1.9.2) was changed to specify "Within 8 hours prior to performance of PHYSICS TESTS." This Frequency requires the nuclear overpower trip setpoint be verified prior to the onset of PHYSICS TESTS which ensures that the established LCO conditions are satisfied, with respect to the trip function. The requirement to perform these NUREG SRs with a Frequency of 8 hours is excessively restrictive and unduly burdensome on the operation of the unit. The short time frame in which the unit is expected to be conducting PHYSICS TESTS requiring the exception to one or more LCOs does not warrant the increased verification requirements. Further, these SRs provide a verification of RPS system performance at a Frequency significantly shorter than that required of the RPS when operating in MODE 1 at RATED THERMAL POWER (ref. NUREG 3.3.1). No basis exists to imply that the RPS trip function, or its calibration, would behave differently than that observed during power operation. The CTS does not contain a similar surveillance requirement. The ANO-1 current license basis does require that the high flux trip (nuclear overpower trip) setpoints are administratively set, as stated in the CTS Bases associated with CTS 3.5.2. The proposed change in Frequency is considered to be consistent with ANO-1's current practice, as allowed by the current license basis. The Bases were changed to reflect this Frequency.

3 NUREG 3.1.4 - Incorporated TSTF-143.

4 NUREG 3.1.3 - The Moderator Temperature Coefficient (MTC) limits in ITS 3.1.3 were modified to specify the current license requirements as presented in CTS 3.1.7.1. Because there is no MTC value presently specified in the ANO-1 COLR, nor is there a value to be relocated to the COLR, ITS 3.1.3 was revised to specify that the MTC shall be non-

positive whenever THERMAL POWER is greater than or equal to 95% of RTP and shall be less positive than 0.9×10^{-4} Δk/k/°F whenever THERMAL POWER is less than 95% RTP. These changes are in accordance with current license basis. Further, this change results in ITS 3.1.3 establishing a maximum positive limit that is consistent with NUREG-1430.

3.1-12 In SR 3.1.3.1, the phrase "within the upper limit specified in the COLR" was changed to "within the limits" to coincide with the LCO requirements.

SR 3.1.3.2 has been deleted because the CTS contains no lower limit on MTC. The lower limit for MTC will remain under licensee administrative control. This value is validated through observation of core physics parameters over the cycle duration. These parameters have historically indicated close agreement between core design assumptions and actual core parameters thus indicating agreement between the actual MTC values and those assumed in the cycle reload analyses. These changes are consistent with current license basis.

The Bases for 3.1.3 were similarly modified to reflect the above described changes. In addition, the 3.1.3 LCO Bases were modified to include CTS Bases guidance that the positive MTC limit below 95% RTP is to be corrected to the 95% RTP power level. This results in a linearly decreasing positive MTC value as power is increased from Hot Zero Power to 95% RTP. This change is consistent with current license basis.

5 NUREG 3.1.4 - ITS LCO 3.1.4 requires each CONTROL ROD to be OPERABLE and aligned to within 6.5% of its group average height, consistent with NUREG 3.1.4.

3.1-05 CTS 4.7.1 requires control rods to be declared inoperable if: 1) a CONTROL ROD trip insertion time is not met; 2) a CONTROL ROD is misaligned with its group average by more than 9 inches; or 3) a CONTROL ROD cannot be exercised or if it cannot be located with absolute or relative position indications or in or out limit lights. As discussed in 3.1DOC-A11, the requirement to declare a misaligned CONTROL ROD inoperable has not been retained due to the format of ITS 3.1.4, which requires the same actions for an inoperable CONTROL ROD, or a misaligned CONTROL ROD. In the ITS, a CONTROL ROD will be considered inoperable if: 1) the CONTROL ROD is not free to insert into the core within the required insertion time; or 2) the CONTROL ROD is required to be declared inoperable by LCO 3.1.6, "Position Indicator Channels." A misaligned CONTROL ROD, while requiring entry into the applicable Condition, will not result in declaring a CONTROL ROD inoperable. The ITS 3.1.4 LCO Bases have been revised to add a clarification for CONTROL ROD OPERABILITY.

The terms "trippable," "trippability" and "untrippable" as they relate to CONTROL RODS have been removed from several locations within ITS 3.1.4 and the supporting BASES. This change preserves the current license basis. The CTS does not distinguish between trippable (a CONTROL ROD that is declared inoperable because it can not be located) and untrippable (a CONTROL ROD that is not free to insert into the core) inoperable CONTROL RODS. The deletion of the words "trippable," "trippability" and "untrippable" is consistent with CTS and represents no change in intent or application from current license basis.

ITS DISCUSSION OF DIFFERENCES

NUREG ACTION D was deleted because ITS 3.1.4 ACTIONS A, B and C, with the indicated changes, provide the requirements for all inoperable CONTROL RODS. Inoperable CONTROL RODS will continue to be dealt with consistently whether "trippable" or "untrippable." This maintains requirements consistent with CTS.

These changes are acceptable because the negative reactivity worth of an untrippable CONTROL ROD can be easily compensated for in the SHUTDOWN MARGIN (SDM) verification. SDM verification is the first Required Action in ITS 3.1.4. Thus, core reactivity and SDM considerations during operation are preserved in accordance with safety analysis assumptions. Further, if the CONTROL ROD is aligned within limits of its group average position (and the group average position is within the limits of ITS 3.2.1), then the power distribution of the core is unaffected. This similarly preserves the initial power distribution conditions of the safety analysis. Therefore, ITS Conditions A, B and C provide appropriate actions for continued operation with either an untrippable CONTROL ROD or an otherwise trippable CONTROL ROD that has been declared inoperable for some other reason.

The Bases have been revised to be consistent with the above mentioned changes. In addition, the Bases for SR 3.1.4.3 were modified to include additional detail regarding the control rod drop time testing. This change is consistent with current license basis.

6 NUREG 3.1.2, 3.1.3, & 3.1.4 - The word "Once" has been added to the Frequency of SR 3.1.2.1, SR 3.1.3.1, and SR 3.1.4.3 in ITS Section 3.1. This addition has been made to provide consistency between this statement of Frequency and the information contained within NUREG Section 1.4, Frequency. Discussions within Section 1.4 repeatedly emphasize the use of the term "Once" in this type of statement of Frequency. This change has been made specifically for clarification and consistency, and is considered to be editorial.

7 NUREG 3.1.6 - The wording of ITS 3.1.6 LCO and Condition A was changed to be consistent with the statements presented in ITS 3.1.4 LCO and Condition A. This editorial change establishes consistency between similar LCOs within the ITS.

ITS 3.1.6 Applicability will be MODES 1 and 2 in accordance with TSTF-159, Rev1.

3.1-10

The Bases were revised as necessary to reflect these changes. In addition, the last paragraph of the LCO Bases was revised to remove reference to peaking factors, leaving reference only to LHRs. No change in intent is associated with this change which is consistent with changes made elsewhere in the ITS Bases.

ITS DISCUSSION OF DIFFERENCES

8 NUREG 3.1.4 - A portion of the methodology specified in NUREG SR 3.1.4.2 has been deleted. This change was made to maintain testing requirements consistent with the CTS. The CTS does not contain this level of detail with regard to CONTROL ROD testing. Specific methodology, including the minimum distance a CONTROL ROD must be moved during testing, is currently contained in documents under licensee control and for consistency will be maintained under licensee control. Removal of these details will not change the intent of the SR and will maintain current testing requirements.

Further, to maintain consistency with the NUREG Bases, the words "by moving" were replaced with the word "for." This change takes into account that more than one method of determining rod freedom or the basis for the inability to demonstrate movement of a CONTROL ROD exists. These changes preserve the intent of this SR which is to insure that the CONTROL RODS are capable of inserting into the core in the event of a reactor trip. Moreover, the NUREG SR 3.1.4.2 Bases attempt to establish exceptions to the SR which requires the freedom of movement be demonstrated "by moving." The Bases allow a determination of trippability that may be used to preserve CONTROL ROD OPERABILITY although the CONTROL ROD may not be capable of being moved. This constitutes an SR 3.0.1 exception established within the Bases which is inappropriate.

3.1-09 The amount of control rod movement has been revised for consistency with the current license basis. The Bases for CTS 4.7.1.3 direct that each control rod be exercised by a movement of approximately two inches. This has been incorporated into the Bases for ITS 3.4.1.2. In addition, since the control room indication reads out in percent, an additional value of approximately 1.5% has been incorporated to aid the operator in determining if the acceptance criteria have been met.

9 NUREG 3.1.4 - ITS SR 3.1.4.3 was modified to maintain CONTROL ROD drop time testing consistent with CTS 4.7.1.1 requirements. This change does not add new requirements nor does it change or remove any existing requirements.

The NOTE in NUREG SR 3.1.4.3 was modified to allow continued operation with reactor coolant pump combinations which provide less total reactor coolant system flow than the combination used during CONTROL ROD drop time testing. Continued operation is allowed provided the total reactor coolant flow is less than the total flow during testing. This allowance is appropriate due to the bounding nature of the test flow conditions. ANO-1 is currently licensed for limited operation in a one RCP per loop configuration. This change will allow for continued unit operation, to the extent allowed by CTS 3.1.1.1.A. Without this change to the Note, reducing the number of running RCPs from 3 to 2, with drop time testing having been performed with 3 RCPs running, would have required that all CONTROL RODS be declared inoperable. This declaration is unnecessarily restrictive due to the bounding nature of the test flow conditions.

A portion of the NUREG Bases for SR 3.1.4.3 was deleted because it established a condition requiring performance of the SR that was not consistent with the SR Frequency requirements. The ITS SR Frequency is given as "once prior to reactor criticality after each removal of the reactor vessel head." However, the Bases stated that the SR is required "after CONTROL ROD drive system maintenance or modification." This Bases condition is not included within the scope of the SR 3.1.4.3 Frequency and was therefore deleted.

10 NUREG 3.1.4 - Required Action A.2.3 has been shown as not adopted in the ITS. This item was not a requirement in the CTS for this Condition. The Required Action's reduction of the nuclear overpower trip setpoint does not actively contribute toward the mitigation of the negative effects of operation with a misaligned CONTROL ROD. This type of administrative action is better suited as a licensee controlled procedural action. Lastly, the Bases implication that this reduction in setpoint maintains core protection and operating margins is not supported. By not adopting this Required Action, requirements consistent with current license bases are being maintained.

BASES information for this Required Action has likewise been removed.

11 NUREG 3.1.4 - The Completion Time for Required Action A.1 (ITS 3.1.4 Required Action A.2.1) has been changed from 1 hour to 2 hours. The Required Action of realigning a misaligned CONTROL ROD is not specified in CTS. There is an implied Action presented by CTS 3.5.2.2.6. This specification allows for continued operation above 60% ALLOWABLE THERMAL POWER (ATP) if a previously misaligned CONTROL ROD is no longer misaligned. No Completion Time is specified for either this Specification or CTS 3.5.2.2.5 which requires the power reduction to less than 60% ATP. Due to the lack of current specified Completion Times for the Required Actions of reducing power to less than 60% ATP and realigning a misaligned CONTROL ROD, similar Completion Times of 2 hours have been adopted for both Required Actions. This 2 hour Completion Time along with ITS 3.1.4 Required Action A.1.1 ensures that, within 1 hour, proper SDM is verified or appropriate actions initiated, and within 2 hours, any misaligned CONTROL ROD is realigned or power is reduced below 60% ATP.

12 NUREG 3.1.4, 3.1.6, & 3.1.7 - Incorporated TSTF-110, Rev 2.

13 NUREG 3.1.8 & 3.1.9 - Incorporated TSTF-154, Rev 2. This generic change has been modified to reference the criterion of 10CFR50.36 instead of the NRC Policy Statement. This is an editorial change associated with implementation of the 10CFR50.36 rule changes after NUREG-1430, Rev 1 was issued.

14 NUREG 3.1.5 - Incorporated TSTF-216.

15 NUREG 3.1.8 & 3.1.9 - NUREG LCO 3.1.8 and LCO 3.1.9 were modified to include the allowance to suspend the requirements of ITS 3.2.2, "AXIAL POWER SHAPING ROD (APSR) Insertion Limits," during PHYSICS TESTS in MODES 1 and 2. The inclusion of this exception in the ITS is acceptable based on approved written procedures, administrative controls, the requirements of 10CFR50.59, and ITS LCO 3.1.8 and LCO 3.1.9 provisions in effect during the conduct of PHYSICS TESTS. This exception accommodates LCO 3.2.2 suspension that may be necessary to verify the fundamental characteristics of the nuclear reactor which is critical in demonstrating the adequacy of design, analytical models, and confirmation of analysis results. This change maintains requirements consistent with CTS 3.5.2.5.4.

Required changes to the Bases of ITS 3.1.8 and 3.1.9 were also made. An insert to the Bases was made to further clarify the basis for the acceptability of allowing PHYSICS TESTS exceptions. This Bases addition is entirely editorial in nature. Reference to Regulatory Guide 1.68, Revision 2, August 1978, and ANSI/ANS-19.6.1-1985, December 13, 1985, were deleted at each occurrence and replaced with reference to SAR Section 3A.9, "Startup Program - Physics Testing." ANO is not committed to Regulatory Guide 1.68 or ANSI/ANS-19.6.1. This change is consistent with current license basis.

16 NUREG 3.1.7 - The LCO, Actions and Note have been modified to maintain requirements consistent with the CTS requirements for CONTROL ROD and APSR position indication channel requirements. CTS 4.7.1.3 requires only one OPERABLE channel of position indication per rod. If this required channel is inoperable, the associated rod must be declared inoperable and the Actions of the rod's governing Specification must be completed. The CTS requirements are maintained by the indicated changes to ITS 3.1.7.

SR 3.1.7.1 was modified to match the requirements of ITS 3.1.7. This change was made to provide for Surveillance Requirements which adequately address the equipment required by the LCO. This change provides clarification of the inconsistency within the CTS with regard to the required channels of position indication and surveillance requirements. CTS Table 4.1-1, Items 23 and 24 required shiftly checks of both the absolute and relative rod position indication channels, while CTS 4.7.1.3 allowed for unrestricted operation with either or potentially both of these channels inoperable. This change ensures that only the channel which is being credited as providing the required indication need be checked.

ITS SR 3.1.7.2 was also added. This addition maintains testing requirements and Frequency consistent with CTS Table 4.1-1, Items 23 and 24.

17 NUREG 3.1.4 - The Required Actions for ITS 3.1.4 Condition A were reordered. This change was made due to the fact that inoperable and misaligned CONTROL RODS, whether trippable or not, are dealt with similarly by CTS and ITS (Reference DOD 5). Without this change in the order of the Required Actions, verification of proper SDM would not be required during operation with an inoperable (potentially untrippable) rod if it was aligned within 6.5% of its group average height as stipulated in NUREG Required Action A.1. The failure to verify adequate SDM is inappropriate in this condition. This change maintains requirements consistent with CTS requirements. Supporting changes to the order and content of BASES information were also made.

ITS DISCUSSION OF DIFFERENCES

18 NUREG Bases - The Criterion statement at the conclusion of the Applicable Safety Analysis section was modified at each occurrence to refer to 10CFR50.36 instead of the NRC Policy Statement. This is an editorial change associated with the implementation of the 10CFR50.36 rule changes after NUREG-1430, Revision 1 was issued.

For ITS LCOs 3.1.1, 3.1.3, 3.1.4, 3.1.5, and 3.1.7, the 10CFR50.36 Criterion satisfied by the respective ITS LCOs was modified to preserve consistency with the ANO-1 license basis. Specifically, ANO-1 safety analyses upon which ITS LCOs 3.1.1, 3.1.3, 3.1.4, 3.1.5, and 3.1.7 are based were performed with the reactor critical. The ITS Applicability for these Specifications will be MODES 1 and 2. Thus, the Criterion statement was revised to specify that the LCO parameter satisfies Criterion 2 of 10CFR50.36 when in MODES 1 and 2 while critical. When in MODE 2 with the reactor subcritical, the LCO parameter satisfies Criterion 4 of 10CFR50.36. This change is consistent with current license basis and 10CFR50.36.

19 NUREG Bases 3.1.4 - The Bases for ITS 3.1.4 were modified to refer to a Linear Heat Rate (LHR) verification rather than a power peaking factor verification. These changes are consistent with the Bases discussion for ITS 3.2.5, "Power Peaking." Although LHR will be specified, no change in intent is associated with these changes. This is true because LHR verification is direct confirmation using the incore detector system that the core is operating within the design thermal operating limits. For additional information regarding this change, refer to Section 3.2 DOD 31.

20 NUREG 3.1.8 - Item c of the LCO requirements for maintaining the Nuclear Heat Flux Hot Channel Factor and the Nuclear Enthalpy Rise Hot Channel Factor within the limits specified in the COLR was modified in the ITS to specify that the linear heat rate (LHR) be maintained within the limits specified in the COLR. This change is necessary to provide PHYSICS TESTS requirements that are consistent with ITS 3.2.5, "Power Peaking" requirements. This LCO 3.1.8 condition coupled with SR 3.1.8.2 provides acceptable assurance that excessive core LHRs will not exist such that the thermal design limits of the fuel are exceeded. Although the terminology is different, this LCO condition preserves operating restrictions during PHYSICS TESTS consistent with those established in NUREG-1430.

In addition to the terminology change, a Note was added to the LCO, Condition B and SR 3.1.8.2 that specifies that the LCO provision on LHR only applies when THERMAL POWER is greater than 20% RTP. This Note establishes consistency between the LCO provisions of ITS 3.1.8 and ITS 3.2.5. This change is consistent with TSTF-160, Rev 1.

The Bases for ITS 3.1.8 were revised to reflect these changes.

21 NUREG Bases 3.1.3 - Repeated reference to SAR Chapter 14 using multiple reference indications is unnecessary and duplicative. Adequate reference to the SAR is provided by the first words of the introduction into the Applicable Safety Analyses portion of this Bases section.

ITS DISCUSSION OF DIFFERENCES

22 NUREG Bases 3.1.2 - The Bases for ITS 3.1.2 were rewritten in their entirety to reflect the unit specific methodology of performing the reactivity anomaly determination. The NUREG Bases discussion centered around a comparison of the RCS boron concentration with a critical boron concentration curve (boron rundown curve) derived as part of the reload analyses. The ITS was written to reflect that ANO-1 performs a reactivity balance and then compares the value against a known reactivity condition (i.e., net reactivity of zero condition when the reactor is critical). Under critical conditions, a calculated net reactivity of a value other than zero would indicate the existence of a discrepancy in the reactivity parameters used in the calculation. This would then have to be evaluated in accordance with the discussion that was present in the NUREG Bases for LCO 3.1.2.

23 NUREG 3.1.9 - CTS 3.1.8.1 requires that the nuclear overpower trip be set at less than or equal to 5% RTP during the conduct of low power PHYSICS TESTS. Therefore, ITS 3.1.9 and SR 3.1.9.2 will specify that the Nuclear Overpower Trip Setpoint be set at 5% RTP rather than the 25% RTP value established by NUREG-1430. In addition, ITS 3.1.9.b was editorially modified to use terminology consistent with ITS 3.1.8.b and other locations in NUREG-1430. Specifically, ITS 3.1.9.b was modified to read that the "Nuclear overpower trip setpoint is set to ≤5% RTP."

24 NUREG 3.1.9 - The Applicability was modified to read as "During PHYSICS TESTS initiated in MODE 2." This Applicability is required in order to ensure that the Required Action A.1 is completed should THERMAL POWER exceed 5% RTP. As presently written in NUREG-1430, upon exceeding 5% RTP the unit is in MODE 1 and the LCO and its requirements no longer apply. This change is consistent with TSTF-256.

25 NUREG 3.1.9 - Incorporated TSTF-156, Rev 1.

26 NUREG Bases 3.1.9 - Bases information designated in NUREG-1430 as being applicable to SR 3.1.9.1 has been removed because the SR described by this Bases information does not appear in NUREG-1430. The subsequent Bases discussions of SR 3.1.9.2 through SR 3.1.9.4 were renumbered as appropriate due to this deletion.

27 Not used.

28 NUREG Bases 3.1.1 - The Bases for 3.1.1, SDM, was rewritten in its entirety to address ANO-1 current license and administrative requirements. ANO-1 CTS did not establish a required SHUTDOWN MARGIN (SDM) in MODES 3, 4 and 5. ANO-1 is a "hot shutdown" unit in that no safety analyses have been performed in MODES 3, 4 and 5. SAR analyses performed demonstrate the ability of the unit to establish hot shutdown conditions from operating conditions. Thus, all reference to analyses protected by the LCO 3.1.1 requirement was deleted from the Bases. SAR requirements are that the reactor be sufficiently shutdown to preclude inadvertent criticality in the shutdown condition.

ANO-1 has administratively verified adequate SHUTDOWN MARGIN during MODES 3, 4 and 5. In this verification, appropriate credit has been given to withdrawn CONTROL RODS (cocked rod protection), RPS operating mode (interpreted as whether the RPS was in Shutdown Bypass mode) and potential reactivity effects associated with the current plant operating condition. The required degree of subcriticality is maintained through boration, as necessary.

SR 3.1.1.1 is the method of verification of adequate SDM and is referenced from numerous MODE 1 and 2 LCOs. As such, the information in the Bases must support the derivation of SDM in MODES 1 through 5. Thus, additional reactivity parameters associated with unit operation above the point of adding heat have been added. The specific methodology for performing a SDM calculation will be maintained under licensee administrative control.

29 NUREG 3.1.2 - Incorporated TSTF-142.

30 NUREG Bases 3.1.1 - Reference to a specific volumetric flow rate, a specific boron concentration and a specific differential boron worth in deriving an example for approximate boration duration is inappropriate. All of these factors are a function of system operating characteristics, limitations, time in core life or available boration source. The more appropriate method is to establish boration from an appropriate source and to maximize the injection to the extent possible with consideration for reactor coolant system inventory and makeup and letdown system capacities. Further, this boration is required to continue until the boron concentration is verified to be sufficient to achieve the required shutdown margin.

31 Not used.

32 NUREG Bases 3.1.2 - The NUREG Bases statement that ITS 3.1.2 does not apply in MODE 6 was modified to remove reference to post-criticality testing that verifies the SDM. The verification of SDM in MODE 2 is of little benefit in assuring adequate SDM in MODE 6. The statement that fuel loading continually changes the reactivity condition of the core is correct and a portion of the basis for the SDM requirements in MODE 6 as stated.

33 NUREG 3.1.4 - A Note was added to precede ITS 3.1.4 Required Action A.2.2.3 (NUREG 3.1.4 Required Action A.2.5) that specifies the performance of SR 3.2.5.1 for verification of core power distribution only applies when THERMAL POWER is greater than 20% RTP. This Note is necessary to establish a correlation between the minimum power level at which the incore detector system can be reliably used to provide accurate indication of core power distribution and when the SR is required to be performed. This Note establishes consistency between the Required Action and ITS 3.2.5. This change is consistent with TSTF-160, Rev 1.

The Bases were similarly modified to include the Note.

34 NUREG Bases 3.1.6 - The Applicable Safety Analysis discussion for ITS 3.1.6 is revised to reflect ANO plant specific design and analysis. There are no explicit safety analyses associated with misaligned APSRs. Limits on their alignment are specified in the ITS to preserve assumptions used in the power distribution analysis that supports ITS LCO 3.2.1, LCO 3.2.3 and LCO 3.2.4. This change is consistent with current license basis.

35 NUREG Bases 3.1.4 - The entire discussion of a second type of CONTROL ROD misalignment was deleted from the Bases. The NUREG Bases identified a second type of misalignment associated with a failure of one CONTROL ROD to insert (i.e. remain fully withdrawn) while all other CONTROL RODS insert fully. This discussion is inappropriate for the Bases of an LCO having Applicability in MODES 1 and 2 because: 1) the misalignment does not result in power peaking such that thermal design limits of the fuel would be exceeded, and 2) the misalignment is already discussed and provided for in the Bases for LCO 3.1.1, "Shutdown Margin (SDM)."

36 NUREG Bases 3.1.6 - The indicated changes remove all reference to a dropped APSR. The APSR mechanical design precludes its dropping into the reactor should its associated Control Rod Drive Mechanism become deenergized. It is non-credible for an APSR to drop into the reactor or become misaligned from its group due to dropping. The removal of these sentences does not alter the intent of the remaining passages or the Specification.

37 NUREG Bases 3.1.4 - The indicated changes represent clarification of the logic associated with the relationship between the relative position indicator and the power supply to the CONTROL ROD drives. Individual rods and groups may receive power from their associated group power supply, DC hold power supply or from the auxiliary power supply (as appropriate). Different power supply alignments to individual rods within a group could result in variations in the relative position indication for the rods within the group. The intent of the Bases statements remain the same. This change reflects unit design characteristics and is consistent with the current license basis.

38 NUREG Bases - The NUREG statement concerning the GDC criteria is modified in the ITS to reference the current licensing basis description contained in SAR Section 1.4.

39 NUREG Bases 3.1.8 - NUREG SR 3.1.8.4 (ITS SR 3.1.8.3) material describing the verification of SDM was erroneous. The listing of reactivity effects included parameters supporting the derivation of the SDM while subcritical or while critical below the point of adding heat. Neither is the case during the MODE 1 Applicability established for LCO 3.1.8. The reactivity effects listing was altered to incorporate the Doppler defect associated with heating of the fuel, Moderator defect associated with the heating of the reactor coolant and removal of the isothermal temperature coefficient (ITC) and RCS average temperature. The paragraph describing the necessity of using the isothermal temperature coefficient because the reactor is subcritical is deleted because it is obviously wrong in MODE 1.

Similarly, NUREG SR 3.1.9.4 (ITS SR 3.1.9.3) material describing the verification of SDM was also erroneous. The listing of reactivity effects included parameters supporting the derivation of the SDM while subcritical or while critical below the point of adding heat but did not support derivation of SDM when operating above the point of adding heat. The reactivity effects listing was altered to incorporate the Doppler defect associated with heating of the fuel and Moderator defect associated with the heating of the reactor coolant. The paragraph describing the necessity of using the isothermal temperature coefficient because the reactor is subcritical was modified to reflect that critical conditions may also exist. This change is consistent with TSTF-249.

40 NUREG 3.1.5 - Incorporated TSTF-158, Rev 1.

3.1-10

41 3.1.6 - Incorporates TSTF-220.

Discussions between ANO and Framatome Cogema Fuels (FCF) have indicated that Required Action A.1 incorporated by TSTF-220 may not be sufficient to detect all anomalies in the event of a misaligned APSR. Axial power imbalance is a global parameter. Inoperability of a single APSR is a more localized condition which can result in an increase in power peaking in the fuel assembly containing the APSR, or in an adjacent fuel assembly. Therefore, ANO has submitted a generic change for evaluation by the NEI TSTF process. This change revises TSTF-220 Required Action A.1 to require performance of SR 3.2.5.1 and is currently being tracked as ANO-1-063, pending assignment of a TSTF number.

42 NUREG LCO 3.1.9 allows LCO 3.2.1 "restricted operation region only" requirements to be suspended during PHYSICS TESTS. This exception is modified in the ITS 3.1.9 to allow suspension of LCO 3.2.1 requirements, consistent with CTS provisions which allow exception to position limit (does not limit to regulating rods inserted in the restricted region only) and overlap and sequence limits. This is acceptable since limits on THERMAL POWER and shutdown capability maintained during the PHYSICS TESTS ensure fuel damage criteria are preserved even if an accident were to occur with the LCO suspended.

3.1-09

43 Additional information has been incorporated to clarify that the value provided in the ITS 3.1.4 and 3.1.6 LCOs and SR 3.1.4.2 account for all necessary uncertainties and that the implementing procedures are not required to account for any additional uncertainties. This is consistent with the interpretation of the current requirements associated with Control Rod and APSR misalignment and Control Rod exercises.

44 The NUREG Bases 3.1.5 Applicable Safety Analysis discussion has been revised to properly characterize the ANO acceptance criteria for the safety and regulating rod group insertion limits and operability or misalignment. The SAR does not state the acceptance criteria that the core remains subcritical for this event. However, B & W has placed a design objective in the cycle reload methodologies that the core will remain subcritical. A reference to the B & W topical report has also been added. This change is consistent with the current license basis.

3.1-03

CTS

3.1 REACTIVITY CONTROL SYSTEMS

3.1.1 SHUTDOWN MARGIN (SDM)

within

LCO 3.1.1 The SDM shall be ~~greater than or equal to~~ the limit specified in the COLR. ~~The minimum limit shall be ≥ [1.0]% Δk/k.~~

N/A

1

APPLICABILITY: MODES 3, 4, and 5.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. SDM not within limit.	A.1 Initiate boration to restore SDM to within limit.	15 minutes

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
SR 3.1.1.1 Verify SDM greater than or equal to the limit specified in the COLR.	24 hours

3.1 REACTIVITY CONTROL SYSTEMS

3.1.2 Reactivity Balance

CTS

LCO 3.1.2 The measured core reactivity balance shall be within ± 1% $\Delta k/k$ of predicted values. 4.9

APPLICABILITY: MODES 1 and 2. 4.9

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME	
A. Measured core reactivity balance not within limit.	A.1 Re-evaluate core design and safety analysis and determine that the reactor core is acceptable for continued operation.	~~72 hours~~ 7 days	4.9 29
	AND		
	A.2 Establish appropriate operating restrictions and SRs.	7 days ~~72 hours~~	N/A
B. Required Action and associated Completion Time not met.	B.1 Be in MODE 3.	6 hours	N/A

CTS

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY	CTS
SR 3.1.2.1	-------------------NOTES------------------- 1. The predicted reactivity values may be adjusted (normalized) to correspond to the measured core reactivity prior to exceeding a fuel burnup of 60 effective full power days (EFPD) after each fuel loading. 2. This Surveillance is not required to be performed prior to entry into MODE 2. --		NA
	Verify measured core reactivity balance is within ± 1% Δk/k of predicted values.	Once 6 Prior to entering MODE 1 after each fuel loading AND -----NOTE------ Only required after 60 EFPD --------------- 31 EFPD thereafter	4.9 NA NA 4.9

CTS

3.1 REACTIVITY CONTROL SYSTEMS

3.1.3 Moderator Temperature Coefficient (MTC)

LCO 3.1.3 The MTC shall be ~~maintained within the limits specified in the COLR. The maximum positive limit shall be [≤ [ ] Δk/k/°F at RTP].~~ 3.1.7.1

4

non-positive whenever THERMAL POWER is ≥ 95% RTP and shall be less positive than 0.9×10^{-4} ΔK/K/°F whenever THERMAL POWER is < 95% RTP. 3.1.7.1

APPLICABILITY: MODES 1 and 2.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. MTC not within limits.	A.1 Be in MODE 3.	6 hours

3.1.7.3

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
SR 3.1.3.1 Verify MTC is within the ~~upper~~ limits. ~~specified in the COLR.~~	~~Prior to~~ Once entering MODE 1 after each fuel loading

3.1.7.2

6

4

~~(continued)~~

SURVEILLANCE REQUIREMENTS (continued)

	SURVEILLANCE	FREQUENCY
SR 3.1.3.2	-------------------NOTES------------------- 1. This SR is not required to be performed prior to entry into MODE 1 or 2. 2. If the MTC is more negative than the COLR limit when extrapolated to the end of cycle, SR 3.1.3.2 may be repeated. Shutdown must occur prior to exceeding the minimum allowable boron concentration at which MTC is projected to exceed the lower limit. --- Verify extrapolated MTC is within the lower limit specified in the COLR.	Each fuel cycle within 7 EFPDs after reaching an equilibrium boron concentration equivalent to 300 ppm

4

CTS

3.1 REACTIVITY CONTROL SYSTEMS

3.1.4 CONTROL ROD Group Alignment Limits

LCO 3.1.4 Each CONTROL ROD shall be OPERABLE and aligned to within ~~[6.5]%~~ 6.5% of its group average height. 4.7.1.2

APPLICABILITY: MODES 1 and 2. 3.5.2

ACTIONS 5

CONDITION	REQUIRED ACTION	COMPLETION TIME	
A. One ~~trippable~~ CONTROL ROD inoperable, or not aligned to within ~~[6.5]%~~ 6.5% of its group average height, or both.	A.2.1 ~~A.1~~ ~~Align all CONTROL RODS in the group to within [6.5]% of the group average height, while maintaining the rod insertion, group sequence, and group overlap limits in accordance with LCO 3.2.1, "Regulating Rod Insertion Limits."~~ Restore CONTROL ROD alignment.	~~2~~ 2 hours 3	3.5.2.2.6 N/A 11 17 3
	OR		
	A.1.1 ~~A.2.1.1~~ Verify SDM ~~is ≥ [1]% Δk/k.~~ to be within the limit provided in the COLR.	1 hour AND Once per 12 hours thereafter	17 3.5.2.2.2 3.5.2.2.3 1 N/A
	OR		
	A.1.2 ~~A.2.1.2~~ Initiate boration to restore SDM to within limit.	1 hour	17 3.5.2.2.2 3.5.2.2.3
	AND		
		(continued)	

Note to Reviewers
See Insert A for clarification of format of ACTION A.

INSERT A - Reviewer Clarification - LCO 3.1.4

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One CONTROL ROD inoperable, or not aligned to within 6.5% of its group average height, or both.	A.1.1 Verify SDM to be within the limit provided in the COLR.	1 hour AND Once per 12 hours thereafter
	OR	
	A.1.2 Initiate boration to restore SDM to within limit.	1 hour
	AND	
	A.2.1 Restore CONTROL ROD alignment.	2 hours
	OR	
	A.2.2.1 Reduce THERMAL POWER to ≤ 60% of the ALLOWABLE THERMAL POWER.	2 hours
	AND	
	A.2.2.2 Verify the potential ejected rod worth is within the assumptions of the rod ejection analysis.	72 hours
	AND	
	A.2.2.3 -------NOTE-------- Only required when THERMAL POWER is > 20% RTP. ------------------ Perform SR 3.2.5.1.	72 hours

Insert after page 3.1-6

CTS

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME	
A. (continued)	A.2.2.1 Reduce THERMAL POWER to ≤ 60% of the ALLOWABLE THERMAL POWER.	2 hours	3.5.2.2.5 N/A 17
	~~AND~~		
	~~A.2.3 Reduce the nuclear overpower trip setpoint to ≤ 70% of the ALLOWABLE THERMAL POWER.~~	~~10 hours~~	10
	AND		
	A.2.2.2 ~~A.2.4~~ Verify the potential ejected rod worth is within the assumptions of the rod ejection analysis.	72 hours	17 3.5.2.3 N/A
<INSERT 3.1-7A>	AND		33
	A.2.2.3 ~~A.2.5~~ Perform SR 3.2.5.1.	72 hours	17 N/A
B. Required Action and associated Completion Time for Condition A not met.	B.1 Be in MODE 3.	6 hours	3.5.2.2.3 N/A
C. More than one ~~trippable~~ CONTROL ROD inoperable, or not aligned within ~~[6.5]%~~ 6.5% of its group average height, or both.	C.1.1 Verify SDM ~~is~~ to be ~~≥ [1]% Δk/k~~ within the limit provided in the COLR.	1 hour	5 3.5.2.2.2 3.5.2.2.3 1
	OR		
		(continued)	

<INSERT 3.1-7A>

A.2.2.3 ------------Note---------------
Only required when THERMAL POWER is > 20% RTP.

CTS

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME	CTS
C. (continued)	C.1.2 Initiate boration to restore SDM to within limit. AND C.2 Be in MODE 3.	1 hour 6 hours	3.5.2.2.2 3.5.2.2.3 3.5.2.2.1
~~D. One or more rods untrippable.~~	~~D.1.1 Verify SDM is~~ to be ~~≥ [1]% Δk/k.~~ within the limit provided in the COLR OR ~~D.1.2 Initiate boration to restore SDM to within limit.~~ AND ~~D.2 Be in MODE 3.~~	~~1 hour~~ ~~1 hour~~ ~~6 hours~~	1 5

CTS

SURVEILLANCE REQUIREMENTS

	SURVEILLANCE	FREQUENCY	CTS
SR 3.1.4.1	Verify individual CONTROL ROD positions are within ~~[6.5]%~~ 6.5% of their group average height.	~~4 hours when the asymmetric CONTROL ROD alarm is inoperable~~ ~~AND~~ 12 hours ~~when the asymmetric CONTROL ROD alarm is OPERABLE~~ (12)	N/A
SR 3.1.4.2	Verify CONTROL ROD freedom of movement for ~~(trippability)~~ ~~by moving~~ each individual CONTROL ROD that is not fully inserted ~~≥ 3% in any direction.~~ (5) (8)	92 days	Table 4.1-2 Item 2
SR 3.1.4.3	-------------------NOTE------------------- With rod drop times determined with at least one but less than four reactor coolant pumps operating, operation may proceed provided operation is restricted to the pump combination operating during the rod drop time determination or pump combinations providing less total reactor coolant flow. (9) -- Verify the rod drop time for each CONTROL ROD, from the fully withdrawn position, is ≤ ~~[1.66]~~ 1.66 seconds from power interruption at the CONTROL ROD drive breakers to $\frac{3}{4}$ insertion (25% withdrawn position) with $T_{avg} \geq 525°F$.	Once ~~Prior~~ prior to reactor criticality after each removal of the reactor vessel head (6)	N/A 4.7.1.1 Table 4.1-2 Item 1

CTS

3.1 REACTIVITY CONTROL SYSTEMS

3.1.5 Safety Rod Insertion Limits

LCO 3.1.5 Each safety rod shall be fully withdrawn. 3.5.2.1 3.1.3.5

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. 3.5.2.5.1
Not required for any safety rod inserted to perform
-- 14

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME	
A. One safety rod not fully withdrawn.	~~A.1 Withdraw the rod fully.~~ ~~OR~~	~~1 hour~~	40
	~~A.2.1.1~~ A.1.1 Verify SDM ~~is~~ ~~≥ 1% Δk/k~~ to be within the limit provided in the COLR. OR	1 hour	3.5.2.1 1
	~~A.2.1.2~~ A.1.2 Initiate boration to restore SDM to within limit. AND	1 hour	3.5.2.1
	~~A.2.2~~ A.2 Declare the rod inoperable.	1 hour	3.1.3.7

(continued)

CTS

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME	CTS
B. More than one safety rod not fully withdrawn.	B.1.1 Verify SDM ~~is ≥ 1% Δk/k~~ to be within the limit provided in the COLR. OR	1 hour	3.5.2.1 ①
	B.1.2 Initiate boration to restore SDM to within limit. AND	1 hour	3.5.2.1
	B.2 Be in MODE 3.	6 hours	3.5.2.2.1

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY	
SR 3.1.5.1 Verify each safety rod is fully withdrawn.	12 hours	N/A

CTS

3.1 REACTIVITY CONTROL SYSTEMS

3.1.6 AXIAL POWER SHAPING ROD (APSR) Alignment Limits

LCO 3.1.6 Each APSR shall be OPERABLE and aligned ~~within [6.5]%~~ to within 6.5% of its group average height. (7) 4.7.1.2

APPLICABILITY: MODES 1 and 2 ~~when the APSRs are not fully withdrawn.~~ (7) 3.5.2

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One APSR inoperable ~~or~~ ~~not aligned within its limits~~ ~~or both.~~ to not aligned within 6.5% of its group average height, Perform ~~SR 3.1.3.1~~ SR 3.2.5.1	A.1 ~~Align the APSR group to within [6.5]% of the inoperable or misaligned rod, while maintaining the APSR insertion limits in the COLR.~~ ~~AND~~ ~~A.2 Prevent movement of the APSR group, while the rod remains inoperable or misaligned.~~	2 hours AND 2 hours after each APSR movement ~~2 hours~~
B. Required Action and associated Completion Time not met.	B.1 Be in MODE 3.	6 hours

N/A (7) (41) N/A

3.1-10

3.1-10

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY	CTS
SR 3.1.6.1 Verify position of each APSR is within ~~≤~~ 6.5 ~~%~~ of the group average height.	~~4 hours when the asymmetric CONTROL ROD alarm is inoperable~~ ~~AND~~ 12 hours ~~when the asymmetric CONTROL ROD alarm is OPERABLE~~	N/A (12)

CTS

3.1 REACTIVITY CONTROL SYSTEMS

3.1.7 Position Indicator Channels

LCO 3.1.7 One ~~The absolute position indicator channel and the relative~~ position indicator channel for each CONTROL ROD and APSR shall be OPERABLE. (16) N/A

APPLICABILITY: MODES 1 and 2. N/A

ACTIONS

-----------------------------------NOTE-----------------------------------
Separate Condition entry is allowed for each ~~inoperable position indicator channel~~ CONTROL ROD and APSR. (16) N/A

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. The required ~~relative~~ position indicator channel inoperable for one or more rods.	~~A.1 Determine the absolute position indicator channel for the rod(s) is OPERABLE.~~	~~8 hours AND Once per 8 hours thereafter~~
~~B. The absolute position indicator channel inoperable for one or more rods.~~	~~B.1.1 Determine position of the rods with inoperable absolute position indicator by actuating the affected rod's zone position reference indicators.~~ ~~AND~~	~~8 hours~~ (continued)

4.7.1.3

(16)

CTS

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
B. (continued)	B.1.2 Determine rods with inoperable position indicators are maintained at the zone reference indicator position and within the limits specified in LCO 3.1.5, "Safety Rod Insertion Limit"; LCO 3.2.1, "Regulating Rod Insertion Limits"; or LCO 3.2.2, "AXIAL POWER SHAPING ROD (APSR) Insertion Limits," as applicable.	8 hours **AND** Once per 8 hours thereafter
	OR	
	B.2.1 Place the control groups with nonindicating rods under manual control.	8 hours
	AND	
	B.2.2 Determine the position of the nonindicating rods indirectly with fixed incore instrumentation.	8 hours **AND** Once per 8 hours thereafter **AND** (continued)

16

CTS

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
~~B. (continued)~~		~~-----NOTE----- Not applicable during first 8 hour period -------------- 1 hour after motion of nonindicating rods, which exceeds [15 inches] in one direction since the last determination of the rod's position~~
~~C. The absolute position indicator channel and the relative position indicator channel inoperable for one or more rods. OR Required Action and associated Completion Time not met.~~	A.1 ~~C.1~~ Declare the rod(s) inoperable.	Immediately

16

4.7.1.3

CTS

SURVEILLANCE REQUIREMENTS

	SURVEILLANCE	FREQUENCY
SR 3.1.7.1	Verify ~~the~~ absolute ~~position~~ indicator ~~channels~~ and the ~~relative~~ ~~position~~ ~~indicator~~ ~~channels~~ agree ~~within~~ ~~the~~ ~~limit~~ specified ~~in~~ the ~~COLR~~. Perform CHANNEL CHECK of required position indicator channel.	~~4 hours when the asymmetric CONTROL ROD alarm is inoperable~~ ~~AND~~ 12 hours ~~when the asymmetric CONTROL ROD alarm is OPERABLE~~

Table 4.1-1 Items 23 & 24.

12

16

<INSERT 3.1-17 A SR 3.1.7.2> 16

Table 4.1-1 Items 23 & 24

<INSERT 3.1-17A>

SR 3.1.7.2	Perform CHANNEL CALIBRATION of required position indicator channel.	18 months

 CTS.

3.1 REACTIVITY CONTROL SYSTEMS

3.1.8 PHYSICS TESTS Exceptions—MODE 1

LCO 3.1.8 During the performance of ~~PHYSICS~~ TESTS, the requirements of — edit

LCO 3.1.4, "CONTROL ROD Group Alignment Limits"; — N/A
LCO 3.1.5, "Safety Rod Insertion Limits"; — 3.1.3.5
LCO 3.1.6, "AXIAL POWER SHAPING ROD (APSR) Alignment Limits"; — N/A
LCO 3.2.1, "Regulating Rod Insertion Limits," for the restricted operation region only; — 3.5.2.5.2, 3.5.2.5.3, 3.5.2.5.4
LCO 3.2.2, "AXIAL POWER SHAPING ROD (APSR) Insertion Limits";
LCO 3.2.3, "AXIAL POWER IMBALANCE Operating Limits"; and — 3.5.2.6
LCO 3.2.4, "QUADRANT POWER TILT (QPT)" — 3.5.2.4.1, 3.5.2.4.2, 3.5.2.4.3

may be suspended, provided: (15)

a. THERMAL POWER is maintained ≤ 85% RTP; N/A

b. Nuclear overpower trip setpoint is ≤ 10% RTP higher than the THERMAL POWER at which the test is performed, with a maximum setting of 90% RTP; N/A

<INSERT 3.1-18A>

c. Linear Heat Rate (LHR) is ~~$F_Q(Z)$ and $F^N_{\Delta H}$ are~~ maintained within the limits specified ~~in the COLR; and~~ (20) N/A

3.1.8.3

d. SDM is ~~≥ [1.0]% Δk/k~~ within the limit provided in the COLR. (1)

APPLICABILITY: MODE 1 during PHYSICS TESTS. N/A

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. SDM not within limit.	A.1 Initiate boration to restore SDM to within limit. AND A.2 Suspend PHYSICS TESTS exceptions.	15 minutes 1 hour

N/A

(continued)

<INSERT 3.1-18A>

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

CTS

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
B. THERMAL POWER > 85% RTP. OR Nuclear overpower trip setpoint > 10% higher than PHYSICS TESTS power level. OR Nuclear overpower trip setpoint > 90% RTP. OR <INSERT 3.1-19A> ~~$F_Q(Z)$ or $F^N_{\Delta H}$~~ LHR not within limits.	B.1 Suspend PHYSICS TESTS exceptions.	1 hour

N/A

20

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.1.8.1	Verify THERMAL POWER is ≤ 85% RTP.	1 hour
<INSERT 3.1-19B> SR 3.1.8.2	Perform SR 3.2.5.1.	2 hours
SR 3.1.8.3	Verify nuclear overpower trip setpoint is ≤ 10% RTP higher than the THERMAL POWER at which the test is performed, with a maximum setting of 90% RTP.	Within 8 hours prior to performance of PHYSICS TESTS at each testing plateau

N/A

20 N/A

2 N/A

3.1-12

(continued)

<INSERT 3.1-19A>

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

<INSERT 3.1-19B>

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

CTS

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.1.8.4	Verify SDM ~~is ≥ [1.0]% Δk/k~~ To be within the limit provided in the COLR	24 hours

NA

1

CTS

3.1 REACTIVITY CONTROL SYSTEMS

3.1.9 PHYSICS TESTS Exceptions—MODE 2

LCO 3.1.9 During performance of PHYSICS TESTS, the requirements of

LCO 3.1.3, "Moderator Temperature Coefficient (MTC)"; — N/A
LCO 3.1.4, "CONTROL ROD Group Alignment Limits"; — N/A
LCO 3.1.5, "Safety Rod Insertion Limits"; — 3.1.3.5 (42)
LCO 3.1.6, "AXIAL POWER SHAPING ROD (APSR) Alignment Limits"; — N/A
LCO 3.2.1, "Regulating Rod Insertion Limits," ~~for the restricted operation region only~~; and — 3.5.2.5.2, 3.5.2.5.3, 3.5.2.5.4
LCO 3.2.2, "AXIAL POWER SHAPING ROD (APSR) Insertion Limits";
~~and~~ LCO 3.4.2, "RCS Minimum Temperature for Criticality"~~;~~ — 3.1.3.1

may be suspended, provided: — (5)

a. THERMAL POWER is ≤ 5% RTP; — 3.1.8.1.A, 3.1.8.1.B

b. ~~Reactor trip setpoints on the~~ OPERABLE nuclear overpower ~~channels are~~ trip setpoint is set to ≤ ~~25%~~ 5% RTP; — 3.1.8.1.A, 3.1.8.1.B (23)

c. Nuclear instrumentation/~~source range and intermediate range~~ high startup rate CONTROL ROD withdrawal inhibit is ~~are~~ OPERABLE; and — (25) 3.1.8.2

d. SDM is ~~≥ [1.0]% Δk/k~~ within the limit provided in the COLR. — 3.1.8.3 (1)

APPLICABILITY: ~~MODE 2~~ (D) during PHYSICS TESTS initiated in MODE 2. — (24) N/A

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. THERMAL POWER not within limit.	A.1 Open control rod drive trip breakers.	Immediately

N/A

(continued)

CTS

ACTIONS (continued)

CONDITION	REQUIRED ACTION		COMPLETION TIME
B. SDM not within limit.	B.1	Initiate boration to restore SDM to within limit.	15 minutes
	AND		
	B.2	Suspend PHYSICS TESTS exceptions.	1 hour
C. Nuclear overpower trip setpoint is not within limit. OR Nuclear instrumentation ~~source and intermediate range~~ high startup rate CONTROL ROD withdrawal inhibit inoperable.	C.1	Suspend PHYSICS TESTS exceptions.	1 hour

N/A

N/A

25

SURVEILLANCE REQUIREMENTS

SURVEILLANCE			FREQUENCY
SR	3.1.9.1	Verify THERMAL POWER is ≤ 5% RTP.	1 hour
SR	3.1.9.2	Verify nuclear overpower trip setpoint is ≤ ~~9.5%~~ 5% RTP.	Within ~~8 hours~~ Prior to performance of PHYSICS TESTS

N/A

N/A

2

23

(continued)

3.1-12

CTS

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE	FREQUENCY
SR 3.1.9.3 Verify SDM ~~is ≥ [1.0]% Δk/k.~~ to be within the limit provided in the COLR	24 hours

N/A

1

B 3.1 REACTIVITY CONTROL SYSTEMS

B 3.1.1 SHUTDOWN MARGIN (SDM)

BASES

BACKGROUND

The reactivity control systems must be redundant and capable of holding the reactor core subcritical when shut down under cold conditions per GDC 26 (Ref. 1). SDM requirements provide sufficient reactivity margin to ensure that acceptable fuel design limits will not be exceeded for normal shutdown and anticipated operational occurrences (AOOs). In MODES 3, 4, and 5, the SDM defines the degree of subcriticality that would be obtained immediately following the insertion of all safety and regulating rods, assuming the single CONTROL ROD assembly of highest reactivity worth is fully withdrawn.

In MODES 3, 4, and 5

<INSERT B3.1-1A>

28

The system design requires that two independent reactivity control systems be provided, and that one of these systems be capable of maintaining the core subcritical under cold conditions. These requirements are provided by the use of ~~movable control assemblies~~ CONTROL RODS and soluble boric acid in the Reactor Coolant System (RCS). In MODES 1 and 2, the CONTROL RODS can compensate for the reactivity effects of the fuel and water temperature changes accompanying power level changes over the range from full load to no load. In addition, for analyzed events initiated in MODES 1 and 2, the CONTROL RODS, together with the Chemical Addition and Makeup and Purification System, provide SDM during power operation and are capable of making the core subcritical rapidly enough to prevent exceeding acceptable fuel damage limits, assuming that the rod of highest reactivity worth remains fully withdrawn (Ref. 1).

edit

28

edit

edit

The Chemical Addition and Makeup and Purification System can compensate for fuel depletion, during operation and all xenon burnout reactivity changes, and maintain the reactor subcritical under cold conditions (Ref. 1).

edit

edit

During ~~power~~ operation in MODES 1 and 2, SDM control is ensured by operating with the safety rods fully withdrawn (LCO 3.1.5, "Safety Rod Insertion Limits") and the regulating rods within the limits of LCO 3.2.1, "Regulating Rod Insertion Limits." In MODE 3, consideration must be given to the position of the safety rods and whether the RPS is in Shutdown Bypass in determining the required SDM. When the unit is in ~~the shutdown and refueling modes~~ MODES 3, 4, or 5, the SDM requirements are met by means of adjustments to the RCS boron concentration. Adjusted SDM limits defined in the COLR preclude recriticality in the event of a main steam line break (MSLB) in MODE 3, 4, or 5 when high steam generator levels exist.

Shutdown boron concentration requirements assume the highest worth rod is stuck in the fully withdrawn position to account for a postulated inoperable CONTROL ROD prior to reactor shutdown.

28

(continued)

<INSERT B 3.1-1A>

maintain the core subcritical during these conditions.

In MODES 1 and 2 while critical, SDM requirements are met by the worth of the withdrawn CONTROL RODS which provide sufficient reactivity margin to ensure that acceptable fuel design limits will not be exceeded for normal shutdown and abnormalities. In MODE 2 while subcritical and in MODE 3, with all safety rods withdrawn and the RPS not in Shutdown Bypass, the SDM defines the degree of subcriticality that would be obtained immediately following the insertion of all CONTROL RODS , assuming the single CONTROL ROD of highest reactivity worth is fully withdrawn. In MODES 3, 4, or 5, when all safety rods are not fully withdrawn or the RPS is in Shutdown Bypass, the SDM defines the degree of subcriticality required to be maintained, assuming the CONTROL ROD of highest reactivity worth is fully withdrawn.

For analyzed events in MODES 1 and 2 while critical,

BASES (continued)

APPLICABLE SAFETY ANALYSES

The minimum required SDM is assumed as an initial condition in safety analysis. The safety analysis (Ref. 2) establishes an SDM that ensures specified acceptable fuel design limits are not exceeded for normal operation and AOOs, with assumption of the highest worth rod stuck out following a reactor trip. abnormalities 28 edit

In MODES 1 and 2 while critical The acceptance criteria for SDM requirements are that specified acceptable fuel design limits are maintained. The SDM requirements must ensure that: 28

a. The reactor can be made subcritical from all operating conditions, transients, and Design Basis Events; and

b. The reactivity transients associated with postulated accident conditions are controllable with acceptable limits (departure from nucleate boiling ratio (DNBR), fuel centerline temperature limits for AOOs, and ≤ 280 cal/gm energy deposition for the rod ejection accident)~~; and~~. abnormalities edit

In MODES 3, 4, and 5, the SDM requirements must ensure that ~~c.~~ The reactor will be maintained sufficiently subcritical to preclude inadvertent criticality in the shutdown condition. 28

~~The most limiting accident for the SDM requirements is based on an MSLB, as described in the accident analysis (Ref. 2).~~

~~In addition to the limiting MSLB transient, the SDM requirement must also protect against:~~

~~a. Inadvertent boron dilution;~~

~~b. An uncontrolled rod withdrawal from a subcritical or low power condition;~~

~~c. Startup of an inactive reactor coolant pump;~~

~~d. Rod ejection; and~~

~~e. Return to criticality if an MSLB occurs during high steam generator level operations in MODE 3, 4, or 5.~~

~~The basis for the shutdown requirement when high steam generator levels exist is the heat removal potential in the~~ 28

(continued)

In MODE 2 while subcritical and in MODES 3, 4, and 5, SDM satisfies Criterion 4 of 10 CFR 50.36.

BASES

APPLICABLE SAFETY ANALYSES (continued)

~~secondary system fluid and the negative reactivity added via MTC. At any given initial primary system temperature and its associated secondary system pressure, the secondary system liquid levels can be equated to a final primary system temperature assuming the entire mass is boiled. The resulting RCS temperature determines the required SDM.~~ 28

In MODES 1 and 2 while critical,

SDM satisfies Criterion 2 of ~~the NRC Policy Statement~~. 10 CFR 50.36 (Ref. 1) 18

LCO

~~Shutdown boron concentration requirements assume the highest worth rod is stuck in the fully withdrawn position to account for a postulated inoperable or untrippable rod prior to reactor shutdown.~~ 28

In MODES 1 and 2, and in MODE 3 when all safety rods are fully withdrawn and the RPS is not in Shutdown Bypass

SDM is a core design condition that can be ensured through CONTROL ROD positioning (control regulating and shutdown safety groups) and through the soluble boron concentration. 28

In MODE 3 when all safety rods are not fully withdrawn or the RPS is in Shutdown Bypass, and in MODES 4 and 5, SDM represents a required degree of subcriticality that assumes the highest reactivity worth CONTROL ROD is fully withdrawn.

~~The MSLB (Ref. 2) accident is the most limiting analysis that establishes the SDM value of the LCO.~~

~~For MSLB accidents, if the LCO is violated, there is a potential to exceed the DNBR limit and to exceed 10 CFR 100 limits (Ref. 3).~~

~~To compensate for the potential heat removal associated with an MSLB accident when high steam generator levels exist during secondary system chemistry control and steam generator cleaning, the initial SDM in the core must be adjusted. The figure in the COLR represents a series of initial conditions that ensure the core will remain subcritical following an MSLB accident from those conditions.~~ 28

APPLICABILITY

In MODES 3, 4, and 5, the SDM requirements are applicable to provide sufficient negative reactivity to ~~meet the assumptions of the safety analysis discussed above. The figure in the COLR is used to define the SDM when high steam generator levels exist during secondary system chemistry control and steam generator cleaning in MODES 3, 4, and 5.~~ ensure that the reactor remains subcritical. 28

¶ In MODES 1 and 2, SDM is ensured by complying with LCO 3.1.5

(continued)

BASES

APPLICABILITY (continued)

and LCO 3.2.1. In MODE 6, the shutdown reactivity requirements are given in LCO 3.9.1, "Boron Concentration."

ACTIONS

A.1

If the SDM requirements are not met, boration must be initiated promptly. A Completion Time of 15 minutes is adequate for an operator to correctly align and start the required systems and components. It is assumed that boration will be continued until the SDM requirements are met. ~~If the SDM is below the limit for the steam generator level and RCS temperature specified in the COLR, RCS boration must be continued until the limit specified in the COLR is met.~~ 28

edit

In the determination of the required combination of boration flow rate and boron source concentration, there is no unique requirement that must be satisfied. Since it is imperative to raise the boron concentration of the RCS as soon as possible, the boron concentration should be a highly concentrated solution, such as that normally found in the boric acid ~~storage~~ addition tank (BAAT) or the borated water storage tank (BWST). The operator should borate with the best source available for the ~~plant~~ unit conditions.

edit

~~In determining the boration flow rate, the time in core life must be considered. For instance, the most difficult time in core life to increase the RCS boron concentration is at the beginning of cycle, when the boron concentration may approach or exceed 2000 ppm. Assuming that a value of [1]% Δk/k must be recovered and a boration flow rate is [ ] gpm, it is possible to increase the boron concentration of the RCS by 100 ppm in approximately 35 minutes. If a boron worth of 10 pcm/ppm is assumed, this combination of parameters will increase the SDM by [1]% Δk/k. These boration parameters of [ ] gpm and [ ] ppm represent typical values and are provided for the purpose of offering a specific example.~~ 30

(continued)

BASES (continued)

SURVEILLANCE REQUIREMENTS

. The reactivity effects that are considered in the reactivity balance are dependent upon the operational MODE of the unit. In general, the reactivity balance includes the following

SR 3.1.1.1

The SDM is verified by performing a reactivity balance calculation~~, considering the listed~~ reactivity effects: 28

a. RCS boron concentration;

b. ~~Regulating rod~~ CONTROL ROD position; edit

c. RCS average temperature;

d. Fuel burnup based on gross thermal energy generation;

e. Xenon concentration;

f. Samarium concentration; ~~and~~

g. Isothermal temperature coefficient (ITC) 28

; h. Moderator temperature coefficient (MTC); and i. Doppler defect

Using the ITC accounts for Doppler reactivity in this calculation ~~because~~ when the reactor is subcritical, and the fuel temperature will be changing at the same rate as the RCS. or critical but below the point of adding heat (POAH)

Using the MTC and Doppler defect accounts for the reactivity effects of power operation above the POAH.

The Frequency of 24 hours is based on the generally slow change in required boron concentration, and also allows sufficient time for the operator to collect the required data, which includes performing a boron concentration analysis, and complete the calculation.

REFERENCES

1. ~~10 CFR 50, Appendix A,~~ SAR, Section 1.4 GDC 26. 38

2. ~~F~~SAR, Chapter ~~[14]~~ 3 edit

3. 10 CFR ~~100, "Reactor Site Criteria."~~ 50.36. 18

B 3.1 REACTIVITY CONTROL SYSTEMS

B 3.1.2 Reactivity Balance

BASES

BACKGROUND

According to GDC 26, GDC 28, and GDC 29 (Ref. 1), reactivity shall be controllable, such that subcriticality is maintained under cold conditions, and acceptable fuel design limits are not exceeded during normal operation and ~~anticipated operational occurrences~~ abnormalities. Therefore, the reactivity balance is used as a measure of the agreement between the predicted ~~versus measured~~ core reactivity and the actual during power operation. The periodic confirmation of the predicted core reactivity is necessary to ensure that safety analyses of design basis transients and accidents remain valid. A large reactivity difference could be the result of unanticipated changes in fuel, CONTROL ROD, or burnable poison worth, or operation at conditions not consistent with those assumed in the predictions of core reactivity. These could potentially result in a loss of SDM or violation of acceptable fuel design limits. Comparing The predicted ~~versus measured~~ core reactivity with the actual core reactivity validates the nuclear methods used in the safety analysis and supports the SDM demonstrations ~~(LCO 3.1.1, "SHUTDOWN MARGIN (SDM)")~~ in ensuring the reactor can be brought safely to cold, subcritical conditions. The difference between the actual and predicted core reactivity is commonly referred to as a reactivity anomaly.

When the reactor ~~core~~ is critical ~~or~~ in ~~normal power operation~~ MODE 1 or 2, a reactivity balance exists ~~and the net reactivity is zero~~ (referred to as the actual core reactivity state) and the net reactivity is known to be zero. A comparison of predicted Core reactivity and ~~measured~~ the actual Core reactivity is convenient under such a balance, since parameters are being maintained relatively stable under steady state power conditions where. The positive reactivity inherent in the core design is balanced by the negative reactivity of the control components, thermal feedback, neutron leakage, and materials in the core that absorb neutrons, such as Soluble boron and burnable absorbers, producing zero net reactivity. ~~Excess reactivity can be inferred from the boron letdown curve (or critical boron curve), which provides an indication of the soluble boron concentration in the Reactor Coolant System (RCS) versus cycle burnup. Periodic measurement of the RCS boron concentration for comparison with the predicted value with other variables fixed, (such as rod height, temperature, pressure, and power), provides a convenient method of ensuring that core reactivity is within design expectations, and that the~~

edit 22

22

edit 22

22

(continued)

BASES

BACKGROUND (continued)

~~calculational models used to generate the safety analysis are adequate.~~ (22)

In order to achieve the required fuel cycle energy output, the uranium enrichment in the new fuel loading and the fuel remaining from the previous cycle provides excess positive reactivity beyond that required to sustain steady state operation throughout the cycle. When the reactor is critical ~~at RTP and moderator temperature~~, the excess positive reactivity of the fuel is compensated by burnable absorbers ~~(if any)~~, CONTROL RODS, APSRs, thermal feedback from the fuel and moderator, fission product ~~whatever neutron~~ poisons (mainly xenon and samarium) ~~are present in the fuel~~, epithermal energy neutron absorbers, neutron leakage, and the ~~RCS~~ Reactor Coolant System boron concentration.

During Cycle operation ~~When the core is producing THERMAL POWER~~, the fuel is being depleted and excess reactivity is decreasing. As the fuel depletes, the primary method of compensating for the reduction in excess reactivity is through a reduction in the RCS boron concentration ~~is reduced to decrease negative reactivity and maintain constant THERMAL POWER. The boron letdown curve is based on steady state operation at RTP. Therefore, deviations from the predicted boron letdown curve may indicate deficiencies in the design analysis, deficiencies in the calculational models, or abnormal core conditions, and must be evaluated~~. (22)

APPLICABLE SAFETY ANALYSES

The acceptance criteria for core reactivity are the establishment of the reactivity balance limit to ensure that ~~plant~~ unit operation is maintained within the assumptions of the safety analyses. edit

Accurate prediction of core reactivity is either an explicit or implicit assumption in the accident analysis evaluations. Every accident evaluation ~~(Ref. 2)~~ is, therefore, dependent upon an accurate evaluation of ~~core~~ reactivity. In particular, SDM and reactivity transients, such as CONTROL ROD withdrawal accidents or rod ejection accidents, are very sensitive to accurate prediction of core reactivity (Ref. 2). These accident analysis evaluations rely on computer codes which have been qualified against available test data, operating ~~plant~~ unit data, and analytical benchmarks. Monitoring the core reactivity balance ensures that the nuclear methods provide an accurate representation of the core reactivity. edit (22) edit edit

Design calculations and safety analyses are performed for each fuel cycle for the purpose of predetermining reactivity

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

behavior and the ~~RCS boron concentration~~ requirements for reactivity control during ~~fuel depletion~~. the operating cycle

22

the actual reactivity condition of the critical reactor

The comparison between ~~measured~~ and predicted initial core reactivity provides a normalization ~~for~~ the calculational models used to predict core reactivity. If the ~~measured and~~ predicted ~~RCS boron concentrations for identical~~ core conditions at beginning of cycle (BOC) do not agree, then the assumptions used in the reload cycle design analysis or the calculational models used to predict ~~soluble boron~~ reactivity requirements may not be accurate. If reasonable agreement between ~~measured~~ and predicted core reactivity exists at BOC, then the prediction may be normalized to the measured boron concentration. Thereafter, any significant deviations in the ~~measured boron concentration from the predicted boron letdown curve, which is developed during fuel depletion,~~ may be an indication that the calculational model is not adequate for ~~core burnups beyond BOC~~, or that an unexpected change in core conditions has occurred. the operating cycle

the / of / the

n opportunity for the

Core reactivity and the actual core reactivity at reference

the actual

predicted reactivity condition from the actual reactivity condition during the operating cycle

22

The normalization of the predicted ~~RCS boron concentration~~ to the ~~measured~~ value is typically performed after reaching RTP following startup from a refueling outage, with the CONTROL RODS in their ~~normal~~ positions ~~for power operation~~. The normalization is performed at BOC conditions, so that core reactivity relative to predicted values can be continually monitored and evaluated, as core conditions change during the cycle.

reactivity parameters

actual reactivity

and APSRs

reference

and fission product poisons at their expected equilibrium concentrations

RCS temperature

22

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

18

LCO

Long term core reactivity behavior is a result of the core physics design and cannot be easily controlled, once the core design is fixed. During operation, therefore, the conditions of the LCO can only be ensured through measurement and tracking, and appropriate actions taken as necessary. Large differences between actual and predicted core reactivity may indicate that the assumptions of the ~~Design Basis Accident (DBA) and transient~~ analyses are no longer valid, or that the uncertainties in the nuclear design methodology are larger than expected. A limit on the reactivity of ± 1% Δk/k has been established, based on engineering judgment. A ± 1% Δk/k deviation in reactivity

accident

22

the predicted

(continued)

BASES

the actual reactivity condition of the reactor

LCO (continued)

from ~~that predicted~~ is larger than expected for normal operation and should therefore be evaluated.

the predicted

actual reactivity

When ~~measured~~ core reactivity is within 1% $\Delta k/k$ of the ~~predicted~~ value at steady state thermal conditions, the core is considered to be operating within acceptable design limits. ~~Since deviations from the limit are normally detected by comparing predicted and measured steady state RCS critical boron concentrations, the difference between measured and predicted values would be approximately 100 ppm (depending on the boron worth) before the limit is reached. These values are well within the uncertainty limits for analysis of boron concentration samples, so that spurious violations of the limit due to uncertainty in measuring the RCS boron concentration are unlikely.~~ 22

APPLICABILITY

In MODES 1 and 2 ~~during fuel cycle operation with $k_{eff} \geq 1$~~, the limits on core reactivity must be maintained ~~because a reactivity balance must exist when the reactor is critical or producing THERMAL POWER.~~ 22

the

balance

to ensure an acceptable SDM and continued adherence to the assumptions used in the accident analysis

As the fuel depletes, core conditions are changing, and confirmation of the reactivity balance ensures the core is operating as designed.

This Specification does not apply in MODES 3, 4, and 5, because the reactor is shutdown and changes to core reactivity due to fuel depletion cannot occur.

the net reactivity condition of the reactor can not easily be determined and

In MODE 6, ~~fuel loading results in a continually changing core reactivity.~~ Boron concentration requirements (LCO 3.9.1, "Refueling Boron Concentration") ensure that fuel movements are performed within the bounds of the safety analysis~~, and an SDM demonstration is required during the first startup following operations that could have altered core reactivity (e.g., fuel movement or CONTROL ROD replacement or shuffling)~~. edit 32

acceptable

ACTIONS

A.1 and A.2

the actual core reactivity

the

Should an anomaly develop between ~~measured~~ and predicted core reactivity, an evaluation of ~~the core~~ design and safety analysis must be performed. Core conditions are evaluated to determine their consistency with input ~~to~~ design 22

the

assumptions used in the core

(continued)

BASES

ACTIONS

A.1 and A.2 (continued)

calculations. Measured core and process parameters are evaluated to determine that they are within the bounds of the safety analysis, and safety analysis calculational models are reviewed to verify that they are adequate for representation of the core conditions. The required Completion Time of ~~72 hours~~ 7 days is based on the low probability of ~~a DBA~~ an abnormality or accident occurring during this period, and allows sufficient time to assess the physical condition of the ~~reactor~~ core and complete the evaluation of the core design and safety analysis. 29 22 edit

Following evaluations of the core design and safety analysis, the cause of the reactivity anomaly may be resolved. If the cause of the reactivity anomaly is a mismatch in core reference conditions at the time of ~~RCS boron concentration sampling~~ the reactivity balance, then a recalculation of ~~the RCS boron concentration requirements~~ may be performed to demonstrate that core reactivity is behaving as expected. If an unexpected physical change in the condition of the core has occurred, it must be evaluated and corrected, if possible. If the cause of the reactivity anomaly is in the calculation technique, then the calculational models must be revised to provide more accurate predictions. If any of these results are demonstrated, and it is concluded that the reactor core is acceptable for continued operation, then the ~~boron letdown curve~~ appropriate reactivity parameter may be renormalized, and ~~power~~ operation may continue. In MODE 1 If operational restrictions or additional surveillance requirements are necessary to ensure the reactor core is acceptable for continued operation, then they must be defined. 22

The required Completion Time of ~~72 hours~~ 7 days is adequate for preparing operating restrictions or surveillances that may be required to allow continued reactor operation. 29

B.1

If the core reactivity balance cannot be restored to within the 1% Δk/k limit, the unit must be brought to a MODE in which the LCO does not apply. ~~To achieve this status~~ As a conservative measure, the unit must be brought to at least MODE 3 within 6 hours. If the SDM for MODE 3 is not met, then boration required by Required Action A.1 of LCO 3.1.1 would occur. The allowed 22

(continued)

BASES

ACTIONS

B.1 (continued)

Completion Time of 6 hours is reasonable, based on operating experience to reach the required unit conditions from full power conditions in an orderly manner and without challenging unit systems.

RTP

edit

SURVEILLANCE REQUIREMENTS

SR 3.1.2.1

Core reactivity is verified by periodic ~~comparisons of measured and predicted RCS boron concentrations~~. The comparison is made considering that other core conditions are fixed or stable, including CONTROL ROD positions, moderator temperature, fuel temperature, fuel depletion, xenon concentration, and samarium concentration. The Surveillance is performed prior to entering MODE 1 as an initial check on core conditions and design calculations at BOC. A Note is included in the SR to indicate that the normalization of predicted core reactivity to the measured value must take place within the first 60 effective full power days (EFPD) after each fuel loading. ~~This allows sufficient time for core conditions to reach steady state, but prevents operation for a large fraction of the fuel cycle without establishing a benchmark for the design calculations.~~ The required subsequent Frequency of 31 EFPD, following the initial 60 EFPD after entering MODE 1 is acceptable, based on the slow rate of core reactivity changes due to fuel depletion and the presence of other indicators (QPT, etc.) for prompt indication of an anomaly. Another Note is included in the SRs to indicate that the performance of the Surveillance is not required for entry into MODE 2.

a

reactivity balance calculation that compares the predicted core reactivity to the actual core reactivity condition (net reactivity of zero condition)

and APSR

once

reactivity

after each fuel loading

may

22

The 60 EFPD after entering MODE 1 allows sufficient time for core conditions to reach steady state, but prevents operation for a large fraction of the fuel cycle without establishing a benchmark for the design calculations.

REFERENCES

1. ~~10 CFR 50, Appendix A~~ SAR, Section 1.4, GDC 26, GDC 28, and GDC 29.

2. ~~F~~SAR, Chapter ~~14~~ 3A and 14.

3. 10 CFR 50.36

38

edit

18

B 3.1 REACTIVITY CONTROL SYSTEMS

B 3.1.3 Moderator Temperature Coefficient (MTC)

BASES

associated

BACKGROUND

So that in the power operating range the net effect of the prompt inherent nuclear feedback characteristic tends to compensate for a rapid increase in reactivity

According to GDC 11 (Ref. 1), the reactor core and its interaction with the Reactor Coolant System (RCS) must be designed for inherently stable power operation, even in the possible event of an accident. In particular, the net reactivity feedback in the system must compensate for any unintended reactivity increases. 5 coil —38

The MTC relates a change in core reactivity to a change in reactor coolant temperature (a positive MTC means that reactivity increases with increasing moderator temperature; conversely, a negative MTC means that reactivity decreases with increasing moderator temperature). The reactor is designed to operate with a negative MTC over the largest possible range of fuel cycle operation. Therefore, with a negative MTC a coolant temperature increase will cause a reactivity decrease, so that the coolant temperature tends to return toward its initial value. Reactivity increases that cause a coolant temperature increase will thus be self limiting, and stable power operation will result. The same characteristic is true when the MTC is positive and coolant temperature decreases occur. edit edit edit

MTC values are predicted at selected burnups during the safety evaluation analysis and are confirmed to be acceptable by measurements. Both initial and reload cores are designed so that the beginning of cycle (BOC) MTC is less than or equal to zero when THERMAL POWER is 95% RTP or greater. The actual value of the MTC is dependent on core characteristics, such as fuel loading and reactor coolant soluble boron concentration. The core design may require additional burnable absorbers to yield an MTC at BOC within the range analyzed in the plant accident analysis. The end of cycle (EOC) MTC is also limited by the requirements of the accident analysis. Fuel cycles that are designed to achieve high burnups or that have changes to other characteristics are evaluated to ensure the MTC does not exceed the EOC limit. become more negative than the value assumed in the safety analyses. edit —4 edit edit

(continued)

BASES (continued)

APPLICABLE SAFETY ANALYSES

Reference 2 contains analyses of accidents that result in both overheating and overcooling of the reactor core. MTC is one of the controlling parameters for core reactivity in these accidents. Both the most positive value and most negative value of the MTC are initial conditions in the safety analyses, and both values must be bounded. Values used in the analyses consider worst case conditions, such as ~~very large soluble boron concentrations~~ for overheating events, to ensure the accident results are bounding. ~~(Ref. 3).~~ EDIT. 21

The acceptance criteria for the specified MTC are:

a. The MTC values must remain within the bounds of those used in the accident analysis ~~(Ref. 2)~~; and 21

b. The MTC must be such that inherently stable power operations result during normal operation and accidents, such as overheating and overcooling events.

Accidents that cause core overheating (either decreased heat removal or increased power production) must be evaluated for results when the MTC is positive. Reactivity accidents that cause increased power production include the CONTROL ROD withdrawal transient from either zero or full THERMAL POWER. The limiting overheating event relative to plant response is based on the maximum difference between core power and steam generator heat removal during a transient. The most limiting event with respect to positive MTC is ~~a rod withdrawal accident from zero power, also referred to as a~~ the startup accident. ~~(Ref. 4).~~ EDIT. 21

Accidents that cause core overcooling must be evaluated for results when the MTC is most negative. The event that produces the most rapid cooldown of the RCS, and is therefore the most limiting event with respect to the negative MTC, is a steam line break (SLB) event. Following the reactor trip for the postulated EOC SLB event, the large moderator temperature reduction, combined with the large negative MTC, may produce reactivity increases that are as much as the shutdown reactivity. When this occurs, a substantial fraction of core power ~~is~~ may be produced with all CONTROL ROD assemblies inserted, except the most reactive one. Even if the reactivity increase produces slightly subcritical conditions, a large fraction of core power may be produced through the effects of subcritical neutron multiplication. EDIT.

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

MTC values are bounded in reload safety evaluations, assuming steady state conditions at BOC and EOC. ~~A near EOC measurement is conducted at conditions when the RCS boron concentration reaches approximately 300 ppm. The measured value may be extrapolated to project the EOC value, in order to confirm reload/design predictions.~~ 4

In MODES 1 and 2 while critical, MTC satisfies Criterion 2 of ~~the NRC Policy Statement~~ 10CFR 50.36 (Ref. 3). 18

In MODE 2 while sub-critical, MTC satisfies Criterion 4 of 10CFR 50.36.

LCO

LCO 3.1.3 requires the MTC to be within specified limits ~~of the COLR~~ to ensure the core operates within the assumptions of the accident analysis. During the reload core safety evaluation, the MTC is analyzed to determine that its values remain within the bounds of the original accident analysis during operation. The LCO establishes a maximum positive value that can not be exceeded. The limit of +0.9E-4 ~~%~~ Δk/k/°F (corrected to 95% RTP) on positive MTC, when THERMAL POWER is < 95% RTP, ensures that core overheating accidents will not violate the accident analysis assumptions. The requirement for a ~~negative~~ non-positive MTC, when THERMAL POWER is ≥ 95% RTP, ensures that core operation will be stable. ~~The negative MTC limit for EOC specified in the COLR ensures that core overcooling accidents will not violate the accident analysis assumptions.~~ 4 edit 4 4 4

MTC is a core physics parameter determined by the fuel and fuel cycle design and cannot be ~~easily~~ directly controlled once the core design is fixed during operation, therefore, the LCO can only be ensured through measurement. The surveillance check~~s~~ at BOC ~~and EOC~~ on MTC provide~~s~~ confirmation that the MTC is behaving as anticipated, so that the acceptance criteria are met. edit 4

APPLICABILITY

In MODE 1, the limits on MTC must be maintained to ensure that any accident initiated from ~~THERMAL POWER~~ power operation will not violate the design assumptions of the accident analysis. In MODE 2, the limits must also be maintained to ensure that startup and subcritical accidents, such as the uncontrolled CONTROL ROD ~~assembly~~ or group withdrawal, will not violate the assumptions of the accident analysis. In MODES 3, 4, 5, and 6, this LCO is not applicable, since no Design Basis Accidents (DBAs) using the MTC as an analysis edit edit

(continued)

BASES

APPLICABILITY (continued)

assumption are initiated from these MODES. However, the variation of MTC with temperature in MODES 3, 4, and 5 for DBAs initiated in MODES 1 and 2 is accounted for in the subject accident analysis. The variation of MTC with temperature assumed in the safety analysis, is accepted as valid once the BOC ~~and middle of cycle~~ measurement~~s~~ ~~are~~ is used for normalization. 4

ACTIONS

A.1

MTC is a ~~function of~~ core physics parameter determined by the fuel and fuel cycle designs, and cannot be controlled directly once the designs have been implemented in the core. If MTC exceeds its limits, the reactor must be placed in MODE 3. This eliminates the potential for violation of the accident analysis ~~bounds~~ assumptions. EDIT. The associated Completion Time of 6 hours is reasonable, considering the probability of an accident occurring during the time period that would require an MTC value within the LCO limits, for reaching MODE 3 conditions from ~~full power conditions~~ RTP EDIT. in an orderly manner and without challenging ~~plant~~ unit systems. EDIT.

EDIT.

SURVEILLANCE REQUIREMENTS

move

The ~~following two~~ SR~~s~~ for measurement of the MTC at the beginning ~~and end~~ of each fuel cycle provide~~s~~ for confirmation of the limiting MTC values. The MTC changes slowly from most positive (least negative) to most negative value during fuel cycle operation, as the RCS boron concentration is reduced with fuel depletion. 4

SR 3.1.3.1 EDIT.

The requirement for measurement, prior to initial operation ~~above 5% RTP~~ in MODE 1, satisfies the confirmatory check on the most positive (least negative) MTC value. EDIT.

~~SR 3.1.3.2~~

~~The requirement for measurement, within 7 effective full power days (EFPD) after reaching an equilibrium boron concentration of 300 ppm for RTP, satisfies the confirmatory~~ 4

~~(continued)~~

(continued) EDIT.

BASES

SURVEILLANCE REQUIREMENTS

SR 3.1.3.2 (continued)

check on the most negative (least positive) MTC value. The measurement is performed at any THERMAL POWER equivalent to an RCS boron concentration of 300 ppm (for steady state operation at RTP with all CONTROL RODS fully withdrawn) so that the projected EOC MTC may be evaluated before the reactor actually reaches the EOC condition. MTC values are extrapolated and compensated to permit direct comparison to the specified MTC limits.

4

The SR is modified by two Notes. Note 1 indicates performance of SR 3.1.3.2 is not required prior to entering MODE 1 or 2. Although this Surveillance is applicable in MODES 1 and 2, the reactor must be critical before the Surveillance can be completed. Therefore, entry into the applicable MODE, prior to accomplishing the Surveillance, is necessary.

Note 2 indicates that SR 3.1.3.2 may be repeated, and shutdown must occur, prior to exceeding the minimum allowable boron concentration at which MTC is projected to exceed the lower limit. The minimum allowable boron concentration is obtained from the EOC MTC versus boron concentration slope with appropriate conservatisms. Thus, the projected EOC MTC is evaluated before the lower limit is actually reached.

4

REFERENCES

1. SAR, Section 1.4, ~~10 CFR 50, Appendix A,~~ GDC 11. 38
2. ~~F~~SAR, Chapter ~~[14]~~ 3A and 14. EDIT
3. ~~3. FSAR, Section [ ].~~ 21
3. ~~4.~~ ~~FSAR, Section [ ].~~ 10CFR50.36. 18

B 3.1 REACTIVITY CONTROL SYSTEMS

B 3.1.4 CONTROL ROD Group Alignment Limits

BASES

BACKGROUND

The OPERABILITY ~~(e.g., trippability)~~ of the CONTROL RODS ~~(safety rods and regulating rods)~~ is an initial condition assumption in all safety analyses that assume rod insertion upon reactor trip. Maximum rod misalignment is an initial condition assumption in the safety analysis that directly affects core power distributions and assumptions of available SDM. 5 EDIT. EDIT.

The applicable criteria for these design requirements are ~~10 CFR 50, Appendix A,~~ GDC 10, "Reactor Design," and GDC 26, "Reactivity Control System Redundancy-and Capability" (Ref. 1), and 10 CFR 50.46, "Acceptance Criteria for Emergency Core Cooling Systems for Light Water Nuclear Power Plants" (Ref. 2). H 38

Mechanical or electrical failures may cause a CONTROL ROD to become inoperable or to become misaligned from its group. CONTROL ROD inoperability or misalignment may cause increased power peaking, due to the asymmetric reactivity distribution and a reduction in the total available ~~rod~~ CONTROL ROD worth for reactor shutdown. Therefore, CONTROL ROD alignment and OPERABILITY are related to core operation within design power peaking limits and the core design requirement of a minimum SDM. EDIT.

Limits on CONTROL ROD alignment and OPERABILITY have been established, and all ~~rod~~ CONTROL 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 preserved. EDIT.

CONTROL RODS are moved by their CONTROL ROD drive mechanisms (CRDMs). Each CRDM moves its rod ¼ inch for one revolution of the leadscrew, but at varying rates depending on the signal output from the Control Rod Drive Control System (CRDCS). 1c 1c EDIT.

The CONTROL RODS are arranged into rod groups that are radially symmetric. Therefore, movement of the CONTROL RODS does not introduce radial asymmetries in the core power distribution. The ~~safety rods and the regulating rods~~ CONTROL RODS EDIT.

(continued)

BASES

BACKGROUND (continued)

provide required negative reactivity worth for immediate reactor shutdown upon a reactor trip. The regulating rods provide reactivity ~~(power level)~~ control during normal operation and transients, and their movement is normally ~~governed by the~~ controlled in automatic control system. edit edit edit edit

The axial position of ~~safety rods and regulating rods~~ the CONTROL RODS is indicated by ~~two separate and~~ three independent systems, which are the relative position indicator~~s~~ ~~transducers~~ ~~and~~, the absolute position indicator~~s~~ ~~transducers~~ (see LCO 3.1.7, "Position Indicator Channels"), and the zone reference indicators. edit edit

The relative position indicator transducer is a potentiometer that is driven by electrical pulses from the CRDCS. There is one counter for each CONTROL ROD drive. on a rod Individual rods in a group, when aligned to the same power supply, all receive the same signal to move; therefore, the counters for all rods in a group should normally indicate the same position. The Relative Position Indicator System is considered highly precise. ~~(one rotation of the leadscrew is ¼ inch in rod motion).~~ ~~If~~ However, if a rod does not move for each demand pulse, the counter will still count the pulse and incorrectly reflect the position of the rod. edit 37

The Absolute Position Indicator System provides a highly accurate indication of actual CONTROL ROD position, but at a lower precision than the relative position indicators. This system is based on ~~inductive analog~~ signals from a series of reed switches spaced along a tube. ~~with a center to center distance of 3.75 inches.~~ edit edit edit

<INSERT B 3.1-18 A>> edit

APPLICABLE SAFETY ANALYSES

CONTROL ROD misalignment and inoperability accidents are analyzed in the safety analysis (Ref. 3). The acceptance criteria for addressing CONTROL ROD inoperability or misalignment are that:

a. There shall be no violations of:

1. specified acceptable fuel design limits, or
2. Reactor Coolant System (RCS) pressure boundary ~~damage~~ integrity; and edit

b. The core must remain subcritical after an abnormality or accident. ~~transients.~~ edit

(continued)

<INSERT B3.1-18A>

Other reed switches included in the same tube with the absolute position indicator matrix provide full in and full out limit indications, and position indications at 0%, 25%, 50%, 75%, and 100% travel. This series of seven indicators are called zone reference indicators.

BASES

APPLICABLE SAFETY ANALYSES (continued)

Two ~~Three~~ types of misalignment are distinguished. during MODES 1 and 2. During movement of a CONTROL ROD group, one rod may stop moving, while the other rods in the group continue. This condition may cause excessive power peaking. ~~The second type of misalignment occurs if one rod fails to insert upon a reactor trip and remains stuck fully withdrawn. This condition requires an evaluation to determine that sufficient reactivity worth is held in the CONTROL RODS to meet the SDM requirement with the maximum worth rod stuck fully withdrawn. If a CONTROL ROD is stuck in the fully withdrawn position, its worth is accounted for in the calculation of SDM, since the safety analysis does not take two stuck rods into account.~~ 35 The Second ~~third~~ type of misalignment occurs when one CONTROL ROD ~~rod~~ drops partially or fully into the reactor core. This event causes an initial power reduction followed by a return towards the original power due to positive reactivity feedback from the negative moderator temperature coefficient. Increased peaking during the power increase may result in excessive local linear heat rates (LHRs).

The accident analysis and reload safety evaluations define regulating rod insertion limits that ensure the required SDM can always be achieved if the maximum worth CONTROL ROD is stuck fully withdrawn (Ref. 3 ~~4~~). If a CONTROL ROD is stuck in or dropped in, continued operation is permitted if the increase in local LHR is within the design limits. The Required Action statements in the LCOs provide conservative reductions in THERMAL POWER and verification of SDM to ensure continued operation remains within the bounds of the safety analysis (Ref. 3 ~~5~~). EDIT. EDIT.

Continued operation of the reactor with a misaligned or dropped CONTROL ROD is allowed if the local core LHRs ~~$F_Q(Z)$ and the $F_{\Delta H}^N$~~ are verified to be within their limits in the COLR. When a CONTROL ROD is misaligned, the assumptions that are used to determine the regulating rod insertion limits, APSR insertion limits, AXIAL POWER IMBALANCE limits, and QPT limits are not preserved. Therefore, the limits may not preserve the design peaking factors, and ~~$F_Q(Z)$ and $F_{\Delta H}^N$~~ must be verified directly by incore mapping. Bases Section 3.2, "Power Distribution Limits," contains a more complete discussion of the relation of LHR ~~$F_Q(Z)$ and $F_{\Delta H}^N$~~ to the operating limits. 19 edit

(continued)

BASES

In MODES 1 and 2 while critical,

In MODE 2 while subcritical, the CONTROL ROD group alignment limits satisfy Criterion 4 of 10 CFR 50.36.

APPLICABLE SAFETY ANALYSES (continued)

~~T~~it The CONTROL ROD group alignment limits satisfy Criterion 2 of ~~the NRC Policy Statement~~ 10CFR50.36 (Ref. 4). 18

LCO

The limits on CONTROL ROD group alignment, safety rod insertion, and APSR alignment, together with the limits on regulating rod insertion, APSR insertion, AXIAL POWER IMBALANCE, and QPT, ensure the reactor will operate within the fuel design criteria. The Required Actions in these LCOs ensure that deviations from the alignment limits will either be corrected or that THERMAL POWER will be adjusted, so that excessive local LHRs will not occur and the requirements on SDM and ejected rod worth are preserved.

The limit for individual CONTROL ROD misalignment is approximately ~~6.5%~~ 6.5% (9 inches) deviation from the group average position. This value is established, based on the distance between reed switches, with additional allowances for uncertainty in the absolute position indicator amplifiers, group ~~maximum or minimum synthesizer~~ average position calculator, and asymmetric alarm or fault detector outputs. ~~The~~ Therefore, no additional uncertainties are required to be incorporated in the implementing procedures. For the purpose of complying with this LCO, the position of ~~an inoperable~~ a misaligned rod is not included in the calculation of the rod group average position. <INSERT B3.1-20A> EDIT. 43 EDIT. 5

3.1-05

Failure to meet the requirements of this LCO may produce unacceptable ~~power peaking factors and~~ LHRs, or unacceptable SDM or ejected rod worth, all of which may constitute initial conditions inconsistent with the safety analysis. 19

APPLICABILITY

The requirements on CONTROL ROD OPERABILITY and alignment are applicable in MODES 1 and 2 because these are the only MODES in which significant neutron (or fission) power is generated, and the OPERABILITY ~~(i.e., trippability)~~ and alignment of rods have the potential to affect the safety of the plant. In MODES 3, 4, 5, and 6, the alignment limits do not apply because the ~~CONTROL RODS are typically bottomed and the reactor is shut down and not producing fission power~~ and resultant local power peaking would not exceed fuel design limits. In ~~the shutdown~~ 3, 4, 5, and 6 MODES, the OPERABILITY of the ~~safety and regulating rods~~ CONTROL RODS has the potential to affect the required SDM, but this effect can be compensated for by an increase in the boron concentration of the RCS. See LCO 3.1.1, "SHUTDOWN MARGIN (SDM)," for SDM in MODES 3, 4, and 5, and 5 EDIT. EDIT. EDIT. EDIT. EDIT.

(continued)

<INSERT B3.1-20A>

3.1-05

A CONTROL ROD is not considered to be inoperable due solely to misalignment. A CONTROL ROD is considered to be inoperable if it is not free to insert into the core within the required insertion time, or as directed by LCO 3.1.7, "Position Indicator Channels."

BASES

APPLICABILITY (continued)

LCO 3.9.1, "Boron Concentration," for boron concentration requirements during ~~refueling~~ MODE 6.

EDIT.

ACTIONS

~~A.1~~ A.2.1

Alignment of the inoperable or misaligned CONTROL ROD may be accomplished by either moving the single CONTROL ROD to the group average position, or by moving the remainder of the group to the position of the single inoperable or misaligned CONTROL ROD. Either action can be used to restore the CONTROL RODS to a radially symmetric pattern. However, this must be done without violating the CONTROL ROD group sequence, overlap, and insertion limits of LCO 3.2.1, "Regulating Rod Insertion Limits," given in the COLR. THERMAL POWER must also be restricted, as necessary, to the value allowed by the insertion limits of LCO 3.2.1. The required Completion Time of ~~1 hour~~ 2 hours is acceptable because local xenon redistribution during this short interval will not cause a significant increase in LHR. This option of inserting the group to the position of the misaligned rod is not available if a safety rod is misaligned, since the limits of LCO 3.1.5, "Safety Rod Insertion Limits," would be violated.

MOVE DOWN TO FOLLOW A.1.2.

11

EDIT

~~A.2.1.1~~ A.1.1

Compliance with Required Actions of Condition A ~~A.2.1.1 through A.2.5~~ allows for continued power operation with one CONTROL ROD inoperable ~~but trippable~~, or misaligned from its group average position, or both. ~~These Required Actions comprise the final alternate for Condition A.~~

5

17

MOVE

If realignment of the CONTROL ROD to the group average or alignment of the group to the misaligned CONTROL ROD is not completed within 1 hour ~~(Required Action A.1 not met)~~, the rod ~~should~~ shall be considered inoperable. Since the rod may be inserted farther than the group average insertion for a long time, SDM must be evaluated. Ensuring the SDM meets the minimum requirement established in the COLR within 1 hour is adequate to determine that further degradation of the SDM is not occurring.

move

EDIT.

(continued)

BASES

ACTIONS (continued)

~~A.2.1.2~~ A.1.2

If the SDM is less than the limit specified in the COLR, then the

~~Restoration of the required SDM requires~~ increasing the RCS boron concentration, since the CONTROL ROD may remain misaligned and not be providing its normal negative reactivity on tripping. RCS boration must occur as described in Bases Section 3.1.1. The required Completion Time of 1 hour to initiate boration is reasonable, based on the time required for potential xenon redistribution, the low probability of an accident occurring, and the steps required to complete the action. This allows the operator sufficient time for aligning the required valves and starting the boric acid pumps. Boration will continue until the required SDM is restored.

1c — EDIT. — 17

<INSERT A.2.1> from previous page

A.2.~~2~~.1 — 17

Reduction of THERMAL POWER to ≤ 60% ALLOWABLE THERMAL POWER ensures that local LHR increases, due to a misaligned rod, will not cause the core design criteria to be exceeded. The required Completion Time of 2 hours allows the operator sufficient time for reducing THERMAL POWER.

~~A.2.3~~

~~Reduction of the nuclear overpower trip setpoint to ≤ 70% ALLOWABLE THERMAL POWER, after THERMAL POWER has been reduced to 60% ALLOWABLE THERMAL POWER, maintains both core protection and an operating margin at reduced power similar to that at RTP. The required Completion Time of 10 hours allows the operator 8 additional hours after completion of the THERMAL POWER reduction in Required Action A.2.2 to adjust the trip setpoint.~~ — 10

A.2.~~4~~ 2.2 — 17

The existing CONTROL ROD configuration must not cause an ejected rod to exceed the limit of 0.65% Δk/k at RTP or 1.00% Δk/k at zero power (Ref. ~~6~~). This evaluation may require a computer calculation of the maximum ejected rod worth based on nonstandard configurations of the CONTROL ROD groups. The evaluation must determine the ejected rod worth for the ~~remainder of the fuel cycle to ensure a valid~~ duration of time that operation is expected to continue with a misaligned rod. (continued)

3 — EDIT. — EDIT.

BASES

ACTIONS

~~A.2.4~~ A.2.2.2 (continued)

~~evaluation~~, CAP should fuel cycle conditions at some later time become more bounding than those at the time of the rod misalignment, additional evaluation will be required to verify the continued acceptability of operation. The required Completion Time of 72 hours is acceptable because LHRs are limited by the THERMAL POWER reduction and sufficient time is provided to perform the required evaluation. 17

~~A.2.5~~ A.2.2.3 17

Performance of SR 3.2.5.1 provides a determination of the ~~power peaking factors~~ local core LHRs using the Incore Detector System. ~~Verification of the $F_Q(Z)$ and $F^N_{\Delta H}$~~ from an incore power distribution map is necessary to ensure that excessive local LHRs will not occur due to CONTROL ROD misalignment. This is necessary because the assumption that all CONTROL RODS are aligned (used to determine the regulating rod insertion, AXIAL POWER IMBALANCE, and QPT limits) is not valid when the CONTROL RODS are not aligned. The required Completion Time of 72 hours is acceptable because LHRs are limited by the THERMAL POWER reduction and adequate time is allowed to obtain an incore power distribution map. 19

<INSERT B3.1-23A> 33

B.1

If the Required Actions and associated Completion Times for Condition A ~~cannot be~~ are not met, the ~~plant~~ unit must be brought to a MODE in which the LCO does not apply. To achieve this status, the ~~plant~~ unit must be brought to at least MODE 3 within 6 hours. The allowed Completion Time of 6 hours is reasonable, based on operating experience, for reaching MODE 3 from ~~full power conditions~~ RTP in an orderly manner and without challenging ~~plant~~ unit systems. EDIT. EDIT. EDIT. EDIT.

C.1.1

More than one ~~trippable~~ CONTROL ROD becoming inoperable or misaligned ~~, or both inoperable but trippable and misaligned from their group average position,~~ or both, is not expected and may violate the minimum SDM requirement. Therefore, SDM must be evaluated. Ensuring the SDM meets the minimum requirement 5 EDIT.

(continued)

<INSERT B3.1-23A>

Required Action A.2.2.3 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

C.1.1 (continued)

within 1 hour allows the operator adequate time to determine the SDM.

C.1.2

If the SDM is less than the limit specified in the COLR, then the EDIT.

(1c) ~~Restoration of the required SDM requires increasing~~ the RCS boron concentration to provide negative reactivity. RCS boration must occur as described in Bases Section 3.1.1. The required Completion Time of 1 hour for initiating boration is reasonable, based on the time required for potential xenon redistribution, the low probability of an accident occurring, and the steps required to complete the action. This allows the operator sufficient time for aligning the required valves and starting the boric acid pumps. Boration will continue until the required SDM is restored.

C.2

(5)

If more than one ~~trippable~~ CONTROL ROD is inoperable or misaligned from their group average position, continued operation of the reactor may cause the misalignment to increase, as the regulating rods insert or withdraw to control reactivity. If the CONTROL ROD misalignment increases, local power peaking may also increase, and local LHRs will also increase if the reactor continues operation at THERMAL POWER. The SDM is decreased when one or more CONTROL RODS become inoperable at a given THERMAL POWER level, or if one or more CONTROL RODS become misaligned by insertion from the group average position.

Therefore, it is prudent to place the reactor in MODE 3. LCO 3.1.4 does not apply in MODE 3 since excessive power peaking cannot occur. ~~and the minimum required SDM is ensured.~~ The allowed Completion Time of 6 hours is reasonable, based on operating experience, for reaching MODE 3 from ~~full power conditions~~ RTP in an orderly manner and without challenging ~~plant~~ Unit systems. EDIT. EDIT.

(continued)

BASES

ACTIONS (continued)

~~D.1.1 and D.1.2~~

~~When one or more rods are untrippable, the SDM may be adversely affected. Under these conditions, it is important to determine the SDM and, if it is less than the required value, initiate boration until the required SDM is recovered. The Completion Time of 1 hour is adequate for determining SDM and, if necessary, for initiating emergency boration to restore SDM.~~

~~In this situation, SDM verification must include the worth of the untrippable rod as well as a rod of maximum worth.~~

~~D.2~~

~~If the untrippable rod(s) cannot be restored to OPERABLE status, the plant must be brought to a MODE or condition in which the LCO requirements are not applicable. To achieve this status, the plant must be brought to at least MODE 3 within 6 hours.~~

~~The allowed Completion Time is reasonable, based on operating experience, for reaching MODE 3 from full power conditions in an orderly manner and without challenging plant systems.~~

5

SURVEILLANCE REQUIREMENTS

SR 3.1.4.1

CONTROL RODS

6.5%

EDIT

Verification that individual ~~rods~~ are aligned within ~~[6.5]%~~ of their group average height limits at a 12 hour Frequency allows the operator to detect a rod that is beginning to deviate from its expected position. ~~If the asymmetric CONTROL ROD alarm is inoperable, a Frequency of 4 hours is reasonable to prevent large deviations in CONTROL ROD alignment from occurring without detection.~~ The specified Frequency takes into account other ~~rod~~ position information that is continuously available to the operator in the control room, so that during actual ~~rod~~ motion, deviations can immediately be detected.

12

EDIT

CONTROL ROD

EDIT.

(continued)

3.1-09

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.1.4.2

approximately 1.5% (approximately 2 inches)

8

No additional allowances for instrument uncertainties are required to be incorporated in the implementing procedures for this parameter.

Verifying each CONTROL ROD is OPERABLE would require that each rod be tripped. However, in MODES 1 and 2, tripping each CONTROL ROD could result in radial tilts. Exercising each individual CONTROL ROD every 92 days provides increased confidence that all rods continue to be OPERABLE without exceeding the alignment limit, even if they are not regularly tripped. Moving each CONTROL ROD by ~~3%~~ will not cause radial or axial power tilts, or oscillations, to occur. The 92 day Frequency takes into consideration other information available to the operator in the control room and SR 3.1.4.1, which is performed more frequently and adds to the determination of OPERABILITY of the rods. Between ~~required~~ typical performances of SR 3.1.4.2 (determination of CONTROL ROD OPERABILITY by movement), if a CONTROL ROD(S) is discovered to be immovable, but is otherwise determined to be ~~trippable and aligned~~ Capable of being fully inserted, the CONTROL ROD(S) ~~is~~ may continue to be considered ~~to~~ ~~be~~ OPERABLE unless inoperable for some other reason. At any time, if a CONTROL ROD(S) is immovable, a determination of the ~~trippability (OPERABILITY) of~~ capability to fully insert (OPERABILITY) the CONTROL ROD(S) must be made, and appropriate action taken.

43

5

SR 3.1.4.3

<INSERT B3.1-26A>

~~Verification of rod drop time allows the operator to determine that the maximum rod drop time permitted is consistent with the assumed rod drop time used in the safety analysis. The rod drop time given in the safety analysis is 1.4 seconds to ⅔ insertion. Using the identical rod drop curve gives a value of [1.66] seconds to ¾ insertion. The latter value is used in the Surveillance because the zone reference lights are located at 25% insertion intervals. The zone reference lights will activate at ¾ insertion to give an indication of the rod drop time and rod location.~~

5

¶ Measuring ~~rod~~ CONTROL ROD drop times, prior to reactor criticality after reactor vessel head removal ~~and after CONTROL ROD drive system maintenance or modification~~, ensures that the reactor internals and CRDM will not interfere with CONTROL ROD motion or ~~rod~~ drop time. This Surveillance is performed during a ~~plant~~ unit outage, due to the ~~plant~~ conditions needed to perform the SR and the potential for an unplanned ~~plant~~ unit transient if the Surveillance were performed with the reactor at power.

9

EDIT

EDIT

(continued)

<INSERT B 3.1-26A>

Verification of CONTROL ROD drop time allows the operator to determine that the maximum CONTROL ROD drop time permitted is consistent with the assumed CONTROL ROD drop time used in the safety analysis. The CONTROL ROD drop time given in the safety analysis is 1.66 seconds to 3/4 position insertion (Ref. 5). This 1.66 seconds includes 0.14 seconds delay time for opening of the CRD breakers and for CRDM unlatch. Using the CONTROL ROD position versus time and time versus reactivity insertion curves gives a value of 1.4 seconds to 2/3 reactivity insertion upon which the accident analysis is based (Ref. 3). The former value is used in the Surveillance because the zone reference lights are located at 25% insertion intervals. The zone reference lights will activate at 3/4 insertion to give an indication of the CONTROL ROD drop time and CONTROL ROD location. The CONTROL ROD drop time is the total elapsed time from the loss of power to the control rod drive (CRD) breaker under voltage coils until the CONTROL ROD has completed approximately 104 inches of travel from the fully withdrawn position. The safety analysis has included a CRD breaker time delay of 0.080 seconds in SAR Chapter 14 (Ref. 3). If the trip test measurement is begun with the opening of the CRD breakers, the required trip insertion time shall be reduced to 1.58 seconds and the CRD breaker time delay shall be verified to be less than or equal to 0.080 seconds.

BASES

SURVEILLANCE REQUIREMENTS

SR 3.1.4.3 (continued)

This testing is normally performed with all reactor coolant pumps operating and average moderator temperature ≥ 525°F to simulate a reactor trip under actual conditions. However, if the CONTROL ROD ~~rod~~ drop times are determined with less than four reactor coolant pumps operating, a Note allows ~~power~~ operation to continue, provided operation is restricted to the pump combination utilized during the CONTROL ROD ~~rod~~ drop time determination or pump combinations providing less total reactor coolant flow. edit 9

REFERENCES

1. ~~10 CFR 50, Appendix A,~~ GDC 10 and GDC 26. SAR, Section 1.4, 38
2. 10 CFR 50.46.
3. ~~F~~SAR, Chapter ~~[14].~~ 3A and 14 EDIT.
4. ~~FSAR, Section [ ].~~ 10CFR50.36. 18
5. ~~F~~SAR, ~~Section [ ].~~ Chapter 3 EDIT.
6. ~~FSAR, Section [ ].~~ EDIT.

B 3.1 REACTIVITY CONTROL SYSTEMS

B 3.1.5 Safety Rod Insertion Limit

BASES

CONTROL RODS

BACKGROUND

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

CONTROL ROD

The applicable criteria for the reactivity and power distribution design requirements are 10 CFR 50, Appendix A, GDC 10, "Reactor Design," GDC 26, "Reactivity Control System Redundancy and Capability," GDC 28, "Reactivity Limits" (Ref. 1), and 10 CFR 50.46, "Acceptance Criteria for Emergency Core Cooling Systems for Light Water Nuclear Power Reactors" (Ref. 2). 38 edit

CONTROL ROD

in MODES 1 and 2

Limits on safety rod insertion have been established, and all rod positions are monitored and controlled during power operation to ensure that the reactivity limits, ejected rod worth, and SDM limits are preserved. EDIT.

IN MODES 1 and 2, the

designated

The regulating groups are used for precise reactivity control of the reactor. The positions of the regulating groups are normally automatically controlled by the automatic control system, but they can also be manually controlled. They are capable of adding negative reactivity very quickly (compared to borating). The regulating groups must be maintained above designed insertion limits and are typically near the fully withdrawn position during normal operations. Hence, they are not capable of adding a large amount of positive reactivity. Boration or dilution of the Reactor Coolant System (RCS) compensates for the reactivity changes associated with large changes in RCS temperature and fuel burnup. EDIT. EDIT.

Prior to entry into MODE 2 from MODE 3,

must be

The safety groups can be fully withdrawn without the core going critical. This provides available negative reactivity in the event of boration errors. The safety groups are controlled manually by the control room operator. During ~~normal full power operation,~~ the safety groups are fully withdrawn. ~~The safety groups~~ must be completely withdrawn from the core prior to withdrawing any regulating groups EDIT.

(continued)

BASES

BACKGROUND (continued)

during an approach to criticality. The safety groups remain in the fully withdrawn position until the reactor is shut down. They add negative reactivity to shut down the reactor upon receipt of a reactor trip signal.

APPLICABLE SAFETY ANALYSES

CONTROL RODS,

On a reactor trip, all ~~rods (safety groups and regulating groups)~~, except the most reactive rod, are assumed to insert into the core. The safety groups shall be at their fully withdrawn limits and available to insert the maximum amount of negative reactivity on a reactor trip signal. The regulating groups may be partially inserted in the core as allowed by LCO 3.2.1, "Regulating Rod Insertion Limits." The safety group and regulating rod group insertion limits are established to ensure that a sufficient amount of negative reactivity is available to shut down the reactor ~~and maintain the required SDM~~ (see LCO 3.1.1, "SHUTDOWN MARGIN (SDM)") following a reactor trip from RTP ~~full power~~. The combination of regulating groups and safety groups (less the most reactive rod, which is assumed to be fully withdrawn) is sufficient to take the reactor from full power conditions at rated temperature to zero power and to achieve ~~maintain~~ the required SDM at rated no load temperature (Ref. 3). ~~The safety group insertion limit also limits the reactivity worth of an ejected safety rod.~~

EDIT.

EDIT.

EDIT.

edit

Although the SAR does not state this as an acceptance criteria for the main steam line break event, B&W has placed a design objective on this event that the core remains subcritical throughout the event (Ref. 4).

The acceptance criteria for addressing safety and regulating rod group insertion limits and inoperability or misalignment are that:

a. There shall be no violations of:

   1. specified acceptable fuel design limits, or
   2. RCS pressure boundary integrity; and

b. The core must remain subcritical after an abnormality. ~~accident transients.~~

44

In MODES 1 and 2 while critical, the safety rod insertion limits satisfy Criteria 2 and 3 of 10CFR50.36 (Ref. 5). ~~the NRC Policy Statement.~~

18

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

(continued)

BASES (continued)

LCO

The safety groups must be fully withdrawn any time the reactor is ~~critical or approaching criticality~~ in MODE 1 or 2. This LCO in combination with LCO 3.2.1 ensures that a sufficient amount of negative reactivity is available to shut down the reactor and ~~maintain~~ achieve the required SDM following a reactor trip. EDIT. EDIT.

APPLICABILITY

MOVE

The safety groups must be within their insertion limits with the reactor in MODES 1 and 2. This LCO in combination with LCO 3.2.1 ensures that a sufficient amount of negative reactivity is available to shut down the reactor and ~~maintain~~ achieve the required SDM following a reactor trip. Refer to LCO 3.1.1 for SDM requirements in MODES 3, 4, and 5. LCO 3.9.1, "Boron Concentration," ensures adequate SDM in MODE 6. EDIT.

This LCO has been modified by a Note indicating the LCO requirement is suspended ~~during~~ for those safety rods which are inserted solely due to testing in accordance with SR 3.1.4.2. This SR verifies the freedom of the rods to move, and requires the safety group to move below the LCO limits, which would normally violate the LCO. 14

ACTIONS

A.1.1, A.1.2, and A.2
~~A.1, A.2.1.1, A.2.1.2, and A.2.2~~ 40

~~When one safety rod is not fully withdrawn, 1 hour is allowed to fully withdraw the rod. This is necessary because the available SDM may be reduced with one of the safety rods not within insertion limits.~~

~~Alternatively,~~ The rod ~~may~~ must be declared inoperable within ~~the same~~ a 1 hour time frame. This requires entry into LCO 3.1.4, "CONTROL ROD Group Alignment Limits." In addition, since the safety rod may be inserted farther than the group average insertion for a long time, SDM must be evaluated. Ensuring the SDM meets the minimum requirement within 1 hour is adequate to determine that further degradation of the SDM is not occurring.

Restoration of the required SDM, if necessary, requires increasing the boron concentration, since the ~~CONTROL ROD~~ safety rod may remain misaligned and not be providing its normal negative reactivity on tripping. ~~RCS boration must occur as described in Bases Section 3.1.1.~~ The required Completion Time of 1 hour for initiating boration is reasonable, based EDIT. 40 EDIT.

(continued)

BASES

A.1.1, A.1.2 and A.2

ACTIONS ~~A.1, A.2.1.1, A.2.1.2, and A.2.2~~ (continued) 40

on the time required for potential xenon redistribution, the low probability of an accident occurring, and the steps required to complete the action. This allows the operator sufficient time for aligning the required valves and starting the boric acid pumps. Boration will continue until the required SDM is restored.

The allowed Completion Time of 1 hour provides an acceptable time for evaluating and repairing minor problems without allowing the unit ~~plant~~ to remain in an unacceptable condition for an extended period of time. EDIT.

B.1.1 and B.1.2 EDIT.

When more than one safety rod is not fully withdrawn, ~~inoperable~~, there is a possibility that the required SDM may be adversely affected. Under these conditions, it is important to determine the SDM, and if it is less than the required value, initiate boration until the required SDM is recovered. The Completion Time of 1 hour is adequate for determining SDM and, if necessary, for initiating emergency boration to restore SDM.

In this situation, SDM verification must include the worth of any rod not capable of being fully inserted ~~the untrippable rod~~ as well as the CONTROL ROD ~~rod~~ of maximum worth. EDIT. 5

B.2

If more than one safety rod is not fully withdrawn, ~~inoperable~~ the unit must be brought to a MODE where the LCO is not applicable. The allowed Completion Time of 6 hours is reasonable, based on operating experience, for reaching the required MODE from RTP ~~full power conditions~~ in an orderly manner and without challenging unit ~~plant~~ systems. EDIT. EDIT. EDIT.

SURVEILLANCE REQUIREMENTS

SR 3.1.5.1

Verification that each safety rod is fully withdrawn ensures the safety rods are available to provide reactor shutdown capability. EDIT.

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.1.5.1 (continued)

Verification that individual safety rod positions are fully withdrawn at a 12 hour Frequency allows the operator to detect a safety rod beginning to deviate from its expected position. Also, the 12 hour Frequency takes into account other information available in the control room for the purpose of monitoring the status of the safety rods. EDIT.

REFERENCES

1. SAR, Section 1.4, ~~10 CFR 50, Appendix A,~~ GDC 10, ~~and~~ GDC 26, and GDC 28. 38
2. 10 CFR 50.46.
3. ~~F~~SAR, ~~Section~~ Chapters ~~[ ]~~ 3 and 4. EDIT.
4. BAW-10179P-A, "Safety Criteria and Methodology for Acceptable Cycle Reload Analyses," Rev. 2. 44
5. 10 CFR 50.36. 18

B 3.1 REACTIVITY CONTROL SYSTEMS

B 3.1.6 AXIAL POWER SHAPING ROD (APSR) Alignment Limits

BASES

BACKGROUND

The OPERABILITY of the APSRs and APSR ~~rod mis~~alignment are initial condition assumptions in the safety analysis that directly affect core power distributions. The applicable criteria for these power distribution design requirements are ~~10 CFR 50, Appendix A,~~ GDC 10, "Reactor Design," and GDC ~~26~~ 28, "Reactivity Limits" (Ref. 1), and 10 CFR 50.46, "Acceptance Criteria for Emergency Core Cooling Systems for Light Water Nuclear Power Reactors" (Ref. 2). EDIT. 38 EDIT.

Mechanical or electrical failures may cause an APSR to become inoperable or to become misaligned from its group. APSR inoperability or misalignment may cause increased power peaking, due to the asymmetric reactivity distribution. Therefore, APSR alignment and OPERABILITY are related to core operation within design power peaking limits.

Limits on APSR alignment and OPERABILITY have been established, and all APSR and CONTROL ROD ~~rod~~ positions are monitored and controlled during power operation to ensure that the power distribution limits defined by the design peaking limits are preserved. EDIT.

~~CONTROL RODS and~~ APSRs are moved by their CONTROL ROD drive mechanisms (CRDMs). Each CRDM moves its rod ¾ inch for one revolution of the leadscrew, but at varying rates depending on the signal output ~~from the~~ Control Rod Drive Control System (CRDCS). EDIT. EDIT. EDIT.

The APSRs are ~~arranged into~~ ~~rod~~ groups that are radially symmetric. Therefore, movement of the APSRs does not introduce radial asymmetries in the core power distribution. The APSRs, which are used to assist in control of the axial power distribution, are positioned manually and do not trip. ED IT. EDIT.

LCO 3.1.6 is conservatively based on use of black (Ag-In-Cd) APSRs and bounds use of gray (Inconel) APSRs. The reactivity worth of black APSRs is greater than that of gray APSRs; thus the impact of black APSR misalignment on the core power distribution is greater.

(continued)

BASES (continued)

APPLICABLE SAFETY ANALYSES

<INSERT B3.1-34A>

~~APSR misalignment and inoperability are analyzed in the safety analysis (Ref. 3). The acceptance criteria for addressing APSR inoperability or misalignment are that there shall be no violations of:~~

~~a. Specified acceptable fuel design limits, and~~

~~b. Reactor Coolant System (RCS) pressure boundary integrity.~~

34

~~Two types of misalignment or inoperability are distinguished.~~ 36 During movement of an APSR group, one rod may stop moving while the other rods in the group continue. This condition may cause excessive power peaking. ~~The second type of misalignment occurs when one rod drops partially or fully into the reactor core. This event causes an initial power reduction, followed by a return towards the original power, due to positive reactivity feedback from the negative moderator temperature coefficient. Increased peaking during the power increase may result in excessive local linear heat rates (LHRs).~~ 36 ~~The accident analysis and reload safety evaluations define APSR insertion limits that ensure that if an APSR is stuck in or dropped in, the increase in local LHR is within the design limits. The Required Action statement in the LCO provides a conservative approach to ensure that continued operation remains within the bounds of the safety analysis (Ref. 4).~~ 34

move

Continued operation of the reactor with a misaligned APSR is allowed if ~~AXIAL POWER IMBALANCE limits~~ Section 3.2, "Power Distribution Limits," are preserved. edit

34

Because ANO-1 uses gray APSRs

The APSR alignment limits satisfy Criterion ~~2~~ 4 of ~~the NRC Policy Statement.~~ 10CFR50.36, (Ref 3). 18

LCO

The limits on CONTROL ROD group alignment, safety rod ~~insertion~~ withdrawal, and APSR alignment, together with the limits on regulating rod insertion, APSR insertion, AXIAL POWER IMBALANCE, and QPT, ensure the reactor will operate within the fuel design criteria. The Required Action in this LCO ensures deviations from the alignment limits will be adjusted so that excessive local LHRs will not occur.

The limit for individual APSR misalignment is ~~6.5%~~ 6.5% approximately (~~9~~ inches) deviation from the group average position. This edit

(continued)

<INSERT B3.1-34A>

There are no explicit safety analyses associated with misaligned APSRs. However, alignment of the APSRs is required to prevent inducing a QUADRANT POWER TILT. The LCOs governing APSR alignment are provided because the power distribution analysis supporting LCO 3.2.1, LCO 3.2.3 and LCO 3.2.4 assumes the APSRs are aligned.

BASES

LCO (continued)

value is established based on the distance between reed switches, with additional allowances for uncertainty in the absolute position indicator amplifiers, group ~~maximum or minimum synthesizer~~ average position calculator, and asymmetric alarm or fault detector outputs. Therefore, no additional uncertainties are required to be incorporated in the implementing procedures. The position of an inoperable ~~rod~~ APSR is not included in the calculation of the ~~rod~~ APSR group's average position. EDIT. EDIT. EDIT.

Failure to meet the requirements of this LCO may produce unacceptable ~~power peaking factors, and~~ LHRs, which may constitute initial conditions inconsistent with the safety analysis. 43 7

APPLICABILITY

The requirements on APSR OPERABILITY and alignment are applicable in MODES 1 and 2, ~~when the APSRs are not fully withdrawn~~ because these are the only MODES in which significant neutron ~~(or fission)~~ power is generated, and the OPERABILITY and alignment of ~~rods~~ APSRs have the potential to affect the safety of the ~~plant~~ unit. ~~OPERABILITY and alignment of the APSRs are not required when they are fully withdrawn because they do not influence core power peaking.~~ In MODES 3, 4, 5, and 6, the alignment limits do not apply because the reactor is shut down ~~and not producing fission power~~ and excessive local LHRs cannot occur from APSR misalignment. 7 EDIT. EDIT. EDIT. 7 EDIT.

ACTIONS

The ACTIONS described below are required if one APSR is inoperable. The ~~plant~~ unit is not allowed to operate with more than one inoperable APSR. This would require the reactor to be ~~shut down~~ placed in MODE 3, in accordance with LCO 3.0.3. EDIT. EDIT.

A.1 ~~and A.2~~ 41

An alternate to realigning a single misaligned APSR to the group average position is to align the remainder of the APSR group to the position of the misaligned or inoperable APSR, while maintaining APSR insertion, in accordance with the limits in the COLR. This restores the alignment requirements. Deviations up to 2 hours will not cause significant xenon redistribution to occur. ~~Required Action A.1~~ This alternative assumes the APSR group movement does not cause the limits of LCO 3.2.2, "AXIAL POWER SHAPING ROD (APSR) Insertion Limits," to be exceeded. For this reason, 41

3.1-10

(continued)

BASES

ACTIONS

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

APSR group movement

~~Required Action A.1~~ is only practical for instances where small movements of the APSR group are sufficient to re-establish APSR alignment.

The reactor may continue in operation with the APSR misaligned if ~~further movement of the APSR group is prohibited, so that the misalignment does not increase and cause the limits on AXIAL POWER IMBALANCE to be exceeded.~~ The required Completion Time of ~~up~~ to 2 hours will not cause significant xenon redistribution ~~to occur~~.

<INSERT B3.1-36A>

41

B.1

The ~~plant~~ unit must be brought to a MODE in which the LCO does not apply if the Required Actions and associated Completion Times cannot be met. To achieve this status, the ~~plant~~ unit must be brought to at least MODE 3 within 6 hours. The Completion Time of 6 hours is reasonable, based on operating experience, for reaching MODE 3 from RTP in an orderly manner and without challenging ~~plant~~ unit systems. In MODE 3, APSR group alignment limits are not required because the reactor is not generating significant THERMAL POWER and excessive local LHRs cannot occur from APSR misalignment.

edit
edit
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edit

3.1-10

SURVEILLANCE REQUIREMENTS

SR 3.1.6.1

Verification at a 12 hour Frequency that individual APSR positions are within ~~[~~6.5~~]~~% of the group average height limits allows the operator to detect an APSR beginning to deviate from its expected position. ~~If the asymmetric CONTROL ROD alarm is inoperable, a 4 hour Frequency is reasonable to prevent large deviations in APSR alignment from occurring without detection.~~ In addition, APSR position is continuously available to the operator in the control room so that during actual ~~rod~~ APSR motion, deviations can immediately be detected.

12

edit

(continued)

<INSERT B 3.1-36A>

3.1-10 power peaking are surveilled within 2 hours to determine if the power peaking is still within limits. Also, since any additional movement of the APSRs may result in additional imbalance, Required Action A.1 also requires the power peaking surveillance to be performed again within 2 hours after each APSR movement.

BASES (continued)

REFERENCES

1. SAR, Section 1.4, ~~10 CFR 50, Appendix A,~~ GDC 10 and GDC ~~26~~ 28. 38
2. 10 CFR 50.46.
3. ~~FSAR, Section [ ].~~
4. ~~FSAR, Section [ ].~~

3. 10CFR 50.36. 18

B 3.1 REACTIVITY CONTROL

B 3.1.7 Position Indicator Channels

BASES

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the SAR discussion of | adequate | and controls are provided

BACKGROUND

within prescribed ranges for normal operation and monitor accident conditions as appropriate to assure adequate safety

According to GDC 13 (Ref. 1), instrumentation to monitor variables and systems over their operating ranges during normal operation, anticipated operational occurrences, and accident conditions must be OPERABLE. LCO 3.1.7 is required to ensure OPERABILITY of the CONTROL ROD and APSR position indicators, and thereby ensure compliance with the CONTROL ROD and APSR alignment and insertion limits. maintain operating 38

CONTROL RODS

The OPERABILITY, including position indication, of the ~~safety and regulating rods~~ is an initial condition assumption in all safety analyses that assume rod insertion upon reactor trip. Maximum rod misalignment for the safety ~~rods, regulating rods,~~ and APSRs is assumed in the safety analysis, which directly affect core power distributions and assumptions of available SDM. EDIT. EDIT.

CONTROL RODS

Local linear heat rates (LHRs)

Mechanical or electrical failures may cause a CONTROL ROD or APSR to become misaligned from its group. CONTROL ROD or APSR misalignment may cause increased power peaking, due to the asymmetric reactivity distribution and a reduction in the total available rod worth for reactor shutdown. Therefore, CONTROL ROD and APSR alignment are related to core operation within design power peaking limits and the core design requirement of a minimum SDM. Rod position indication is needed to assess rod OPERABILITY and alignment. EDIT. EDIT. EDIT. EDIT.

CONTROL ROD | LHR | CONTROL ROD and APSR

group | and

Limits on CONTROL ROD ~~alignment,~~ APSR alignment, and safety rod position 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 preserved. EDIT EDIT.

CONTROL ROD and APSR | LHR

Three

Two methods of CONTROL ROD and APSR position indication are provided in the CONTROL ROD Drive Control System. The two means are by absolute position indicator and relative position indicator transducers. The absolute position indicator transducer consists of a series of magnetically operated reed switches mounted in a tube parallel to the CONTROL ROD drive mechanism (CRDM) motor tube extension. three 16 EDIT.

and zone reference indicators.

(continued)

BASES

BACKGROUND (continued)

or APSR

absolute

This series of seven indicators are

Switch contacts close when a permanent magnet mounted on the upper end of the CONTROL ROD assembly (CRA) leadscrew extension comes near. As the leadscrew and ~~CRA~~ move, the switches operate sequentially, producing an analog voltage proportional to position. Other reed switches included in the same tube with the position indicator matrix provide full in and full out limit indications, and ~~absolute~~ position indications at 0%, 25%, 50%, 75%, and 100% travel. ~~(called zone reference indicators)~~. The relative position indicator transducer is a potentiometer, driven by a step motor that produces a signal proportional to CONTROL ROD or APSR position, based on the electrical pulse steps that drive the CRDM.

CONTROL ROD or APSR

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~~Two absolute position indicator channel designs may be used in the unit: type A absolute position indicators and type A-R4C absolute position indicators. The type A absolute position indicator transducer is a voltage divider circuit made up of 48 resistors of equal value connected in series. One end of 48 reed switches is connected at a junction between each of the resistors, so that as the magnet mounted on the leadscrew moves, either one or two reed switches are closed in the vicinity of the magnet. The type A-R4C (redundant four channel) absolute position indicator transducer has two parallel sets of voltage divider circuits made up of 36 resistors each, connected in series (channels A and B). One end of 36 reed switches is connected at a junction between each of the resistors of the two parallel circuits. The reed switches making up each circuit are offset, such that the switches for channel A are staggered with the switches for channel B. The type A-R4C is designed such that either two or three reed switches are closed in the vicinity of the magnet. By its design, the type A-R4C absolute position indicator provides redundancy, with the two three sequence of pickup and drop out of reed switches to enable a continuity of position signal when a single reed switch fails to close.~~

edit

and APSR

rod

individual position indication

CONTROL ROD or APSR

CONTROL ROD position indicating readout devices located in the control room consist of single ~~CRA~~ position meters on a ~~wall mounted~~ position indication panel and ~~four~~ group average position meters. ~~on the console.~~ A selector switch permits either relative or absolute position indication to be displayed on all of the ~~single rod~~ meters. Indicator lights are provided on the ~~single CRA meter~~ panel to indicate when each ~~CRA~~ is fully withdrawn, fully inserted,

edit
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edit
edit

(continued)

BASES

EDIT.

BACKGROUND (continued)

enabled, or transferred, and whether a ~~CRA~~ position asymmetry alarm condition is present. Indicators ~~on the console~~ show full insertion, full withdrawal, and enabled for motion for each CONTROL ROD group. ~~Identical instrumentation and devices exist for the APSR group.~~ The consequence of continued operation with an inoperable absolute position indicator or relative position indicator channel is a decreased reliability in determining CONTROL ROD position. Therefore, the potential for operation in violation of design ~~peaking factors~~ or SDM is increased.

lc rod Additional EDIT. EDIT. and APSR EDIT. LHR Limits EDIT.

APPLICABLE SAFETY ANALYSES

CONTROL ROD and APSR position accuracy is essential during power operation. ~~Power peaking~~, ejected rod worth, or SDM limits may be violated in the event of a Design Basis Accident (Ref. 2) with CONTROL RODS or APSRs operating outside their limits undetected. ~~Regulating rod, safety rod~~, and APSR positions must be known in order to verify the core is operating within the group sequence, overlap, design ~~peaking limits~~, ejected rod worth, and with minimum SDM (LCO 3.1.5, "Safety Rod Insertion Limits"; LCO 3.2.1, "Regulating Rod Insertion Limits"; and LCO 3.2.2, "AXIAL POWER SHAPING ROD (APSR) Insertion Limits"). The ~~rod~~ positions must also be known in order to verify the alignment limits are preserved (LCO 3.1.4, "CONTROL ROD Group Alignment Limits," and LCO 3.1.6, "AXIAL POWER SHAPING ROD (APSR) Alignment Limits"). CONTROL ROD and APSR positions are continuously monitored to provide operators with information that ensures the ~~plant~~ is operating within the bounds of the accident analysis assumptions. ~~The~~ CONTROL ROD position indicator channels satisfy Criterion 2 of ~~the NRC Policy Statement~~. ~~The CONTROL ROD position indicators monitor CONTROL ROD position, which is an accident initial condition.~~

EDIT. LHR EDIT. CONTROL ROD LHRs EDIT. CONTROL ROD and APSR EDIT. unit

In MODES 1 and 2 while critical, the

and APSR

10CFR50.36 (Ref. 3).

18

LCO

LCO 3.1.7 specifies that one ~~absolute~~ position indicator channel ~~and one relative position indicator channel~~ be OPERABLE for each CONTROL ROD and APSR.

16

~~The agreement between the relative position indicator channel and the absolute position indicator channel, within the limit given in the COLR, indicates that relative position indicators are adequately calibrated and can be~~

16

In MODE 2 while subcritical, the CONTROL ROD and APSR position indicator channels satisfy Criterion 4 of 10CFR50.36.

(continued)

BASES

LCO (continued)

used for indication of the measurement of CONTROL ROD group position. A deviation of less than the allowable limit, given in the COLR, in position indication for a single CONTROL ROD or APSR, ensures confidence that the position uncertainty of the corresponding CONTROL ROD group or APSR group is within the assumed values used in the analysis that specifies CONTROL ROD group and APSR insertion limits. 16

This

and APSR

~~These~~ requirements ensures that CONTROL ROD position indication during ~~power operation~~ and PHYSICS TESTS is accurate, and that design assumptions are not challenged. OPERABILITY of the position indicator channels ensures that inoperable, misaligned, or mispositioned CONTROL RODS or APSRs can be detected. Therefore, power peaking and SDM can be controlled within acceptable limits.

MODES 1 and 2

LHR

EDIT. EDIT. 16 EDIT

APPLICABILITY

the

In MODES 1 and 2, OPERABILITY of position indicator channels is required, since the reactor is, or is capable of, generating THERMAL POWER in these MODES. In MODES 3, 4, 5, and 6, Applicability is not required because the reactor is shut down with the required minimum SDM and is not generating THERMAL POWER.

Significant

16 EDIT.

ACTIONS

A.1

If the relative position indicator channel is inoperable for one or more rods, the position of the rod(s) is still monitored by the absolute position indicator channel for each affected rod. The absolute position indicator channel may be used if it is determined to be OPERABLE. The required Completion Time of 8 hours is reasonable to provide adequate time for the operator to determine position indicator channel status. Continuing the verification every 8 hours thereafter in the applicable condition is acceptable, based on the fact that during normal power operation excessive movement of the groups is not required. Also, if the rod is out of position during this 8 hour period, the simultaneous occurrence of an event sensitive to the rod position has a small probability. 16

(continued)

BASES

ACTIONS (continued)

B.1.1

If the absolute position indicator channel is inoperable for one or more rods, the position of the rod(s) is monitored by the relative position indicator channel for each affected rod. However, the relative position indicator channel is not as reliable a method of monitoring rod position as the absolute position indicator because it counts electrical pulse steps driving the CRDM motor rather than actuating a switch located at a known elevation. Therefore, the affected rod's position can be determined with more certainty by actuating one of its zone reference indicator switches located at discrete elevations. The required Completion Time of 8 hours provides the operator adequate time for adjusting the affected rod's position to an appropriate zone reference indicator location. If the rod is out of position during this 8 hour period, the simultaneous occurrence of an event sensitive to the rod position has a small probability.

B.1.2

To allow continued operation, the rods with inoperable absolute position indicator channels are maintained at the zone reference indicator position. In addition, the affected rods are maintained within the limits of LCO 3.1.5 (when the affected rod is a safety rod); LCO 3.2.1 (when the affected rod is a regulating rod); or LCO 3.2.2 (when the affected rod is an APSR). This Required Action ensures safety rods remain fully withdrawn, and that regulating rods and APSRs remain aligned within their insertion limits. The required Completion Time of 8 hours is reasonable for allowing the operator adequate time to determine the affected rods are in compliance with these LCOs. Continuing to verify the rod positions every 8 hours thereafter is reasonable for ensuring that rod alignment and insertion are not changing, and provides the operator adequate time to correct any deviation that may occur. Continuing the verification every 8 hours thereafter in the applicable condition is acceptable, based on the fact that during normal power operation excessive movement of the groups is not required. Also, if the rod is out of position during this 8 hour period, the simultaneous occurrence of an event sensitive to the rod position has a small probability.

16

(continued)

BASES

ACTIONS (continued)

B.2.1

If the absolute position indicator is inoperable for one or more rods, the position of the rod is monitored by the relative position indicator channel for each affected rod. However, the relative position indicator channel is not as reliable a method of monitoring rod position as the absolute position indicator because it counts electrical pulse steps. The fixed incore system can be used to indirectly determine the absolute position of the affected rod. The fixed incore instrumentation can provide a continual update of CONTROL ROD position, therefore this method can be used to allow continued operation of the reactor with a manual CONTROL ROD movement, while maintaining verification of CONTROL ROD insertion and alignment. Required Action B.2.1. restricts rod motion by placing the groups with nonindicating rods in manual control; thus, even if the rod fails to move in alignment with the group, misalignment is limited. The required Completion Time of 8 hours provides the operator adequate time for placing the rods in manual control., and is consistent with the required Completion Time for Required Action B.1.1. If the rod is out of position during this 8 hour period, the simultaneous occurrence of an event sensitive to the rod position has a small probability.

B.2.2

Continuing to verify the rod positions every 8 hours is reasonable for ensuring that rod alignment and insertion are not changing, and provides the operator adequate time to correct any deviation that may occur. The additional Completion Time of 1 hour after motion of nonindicating rods, which exceeds 15 inches in one direction since the last determination of the rod's position, ensures that the rod with inoperable position indication will not be misaligned for a significant period of time, in the event the rod is moved. The specified Completion Times are acceptable because the simultaneous occurrence of a mispositioned rod and an event sensitive to the rod position has a small probability.

16

(continued)

BASES

ACTIONS (continued)

A.1

~~C.1~~

required

is

CONTROL ROD or APSR

If ~~both~~ the ~~absolute~~ position indicator channel ~~and relative position indicator channel are~~ inoperable for one or more rods, ~~or if the Required Actions and associated Completion Times are not met,~~ the position of the ~~rod(s)~~ is not known with certainty. Therefore, each affected ~~rod~~ must be declared inoperable, and the limits of LCO 3.1.4 or LCO 3.1.6 apply. The required Completion Time for declaring the rod(s) inoperable is immediately. Therefore LCO 3.1.4 or LCO 3.1.6 is entered immediately, and the required Completion Times for the appropriate Required Actions in those LCOs apply without delay.

16

edit.
edit.

SURVEILLANCE REQUIREMENTS

SR 3.1.7.1 <INSERT B 3.1-44 A> 16

~~Verification is required that the Absolute Position Indicator channels and Relative Position Indicator channels agree within the limit given in the COLR. This verification ensures that the Relative Position Indicator channels, which are regarded as the potentially less reliable means of position indication, remain OPERABLE and accurate.~~ 16 The required Frequency of 12 hours is adequate for verifying that no degradation in system OPERABILITY has occurred. ~~If the asymmetric CONTROL ROD alarm is inoperable, then the Surveillance is performed every 4 hours. This required Frequency is adequate for ensuring that the CONTROL RODS and APSRs do not exceed their alignment limits.~~ 12

<INSERT B3.1-44 B> 16

REFERENCES

1. SAR, Section 1.4, ~~10 CFR 50, Appendix A,~~ GDC 13. Chapter 14. 38 EDIT.

2. ~~FSAR, Section [14.1.2.2], Section [14.1.2.3], Section [14.1.2.6], Section [14.1.2.7], Section [14.2.2.4], and Section [14.2.2.5].~~

3. 10 CFR 50.36. 18

<INSERT B3.1-44A>

A CHANNEL CHECK of the required position indication channel ensures that position indication for each CONTROL ROD and APSR remains OPERABLE and accurate. A CHANNEL CHECK is normally a comparison of the parameter indicated on one channel to a similar parameter on other channels. However, this CHANNEL CHECK will be used to detect gross channel failure; therefore, it is key in verifying that the instrumentation continues to operate properly between each CHANNEL CALIBRATION.

When compared to other channels, the agreement criteria between the channels is determined by the unit staff. If the channels are within the criteria, it is an indication that the channels are OPERABLE.

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 position indicator channel.

<INSERT B3.1-44B>

SR 3.1.7.2

A CHANNEL CALIBRATION of the required position indication channel verifies that the channel responds within the necessary range and accuracy.

The Frequency of 18 months is based on operating experience and consistency with the typical industry refueling cycle.

B 3.1 REACTIVITY CONTROL SYSTEMS

B 3.1.8 PHYSICS TESTS Exceptions Systems—MODE 1

BASES

BACKGROUND

The purpose of this MODE 1 LCO is to permit PHYSICS TESTS to be conducted by providing exemptions from the requirements of other LCOs. Establishment of a test program to verify that structures, systems, and components will perform satisfactorily in service is required by Section XI of 10 CFR 50, Appendix B (Ref. 1). Testing is required as an integral part of the design, fabrication, construction, and operation of the ~~power plant~~ unit. All functions necessary to ensure that specified design conditions are not violated during normal operation and ~~anticipated operational occurrences~~ abnormalities must be tested. Requirements for notification of the NRC, for the purpose of conducting tests and experiments, are specified in 10 CFR 50.59 (Ref. 2). EDIT. EDIT. EDIT.

The key objectives of a test program are to (Ref. 3): 15

a. Ensure that the facility has been adequately designed;

b. Validate the analytical models used in the design and analysis;

c. Verify the assumptions used to predict unit response;

d. Ensure that installation of equipment in the facility has been accomplished in accordance with the design; and

e. Verify that the operating and emergency procedures are adequate.

To accomplish these objectives, testing is performed prior to initial criticality; during startup, low power operations, and power ascension; at high powers; and after each re fueling. The PHYSICS TESTS requirements for reload fuel cycles ensure that the operating characteristics of the core are consistent with the design predictions, and that the core can be operated as designed (Ref. ~~4~~ 3). EDIT. 15

<INSERT B3.1-45A>

PHYSICS TESTS procedures are written and approved in accordance with established guidelines. The procedures include all information necessary to permit a detailed

(continued)

<INSERT B 3.1-45A>

The inclusion of this PHYSICS TESTS Exception LCO is acceptable based on the use of approved written procedures, administrative controls, the requirements of 10CFR50.59, and the LCO 3.1.8 provisions in effect during the conduct of PHYSICS TESTS.

BASES

BACKGROUND (continued)

execution of testing required to ensure the design intent is met. PHYSICS TESTS are performed in accordance with these procedures, and test results are approved prior to continued power escalation and long term power operation. Examples of PHYSICS TESTS include determination of critical boron concentration, CONTROL ROD group worths, reactivity coefficients, flux symmetry, and core power distribution.

APPLICABLE SAFETY ANALYSES

It is acceptable to suspend certain LCOs for PHYSICS TESTS because reactor protection criteria are preserved by the LCOs still in effect and by the SRs. Even if an accident occurs during PHYSICS TESTS with one or more LCOs suspended, fuel damage criteria are preserved because the limits on linear heat rate (LHR), ~~nuclear hot channel factors,~~ ejected rod worth, and shutdown capability are maintained during the PHYSICS TESTS. 20 EDIT.

Reference ~~8~~ 4 describes the ~~defines requirements for~~ initial testing of the facility, including PHYSICS TESTS. Tables ~~13.2 and 13.4~~ 13-2 (Ref. ~~8~~ 4) summarizes the post-criticality ~~zero, low power, and power~~ tests. EDIT. EDIT.
Requirements for reload fuel cycle PHYSICS TESTS are given in SAR Section 3A.9 ~~Table 1 ANSI/ANS-19.6.1-1985~~ (Ref. ~~9~~ 5 3). Although these PHYSICS TESTS are generally accomplished within the limits of all LCOs, one or more LCOs must sometimes be suspended to make completion of PHYSICS TESTS possible or practical. 15

This is acceptable as long as the fuel design criteria are not violated. When one or more of the limits specified in:

LCO 3.1.4, "CONTROL ROD Group Alignment Limits";
LCO 3.1.5, "Safety Rod Insertion Limits";
LCO 3.1.6, "AXIAL POWER SHAPING ROD (APSR) Alignment Limits";
LCO 3.2.1, "Regulating Rod Insertion Limits," for the restricted operation region only;
LCO 3.2.2, "AXIAL POWER SHAPING ROD (APSR) Insertion Limits";
LCO 3.2.3, "AXIAL POWER IMBALANCE Operating Limits"; ~~or~~ and EDIT.
LCO 3.2.4, "QUADRANT POWER TILT (QPT)" 15

are suspended for PHYSICS TESTS, the fuel design criteria are preserved by maintaining the LHR ~~nuclear hot channel factors~~ (in MODE 1 PHYSICS TESTS) within ~~their~~ limits, maintaining ejected rod worth within limits by restricting regulating rod insertion to within the acceptable operating region or the restricted operating region, by limiting maximum THERMAL POWER and by maintaining SDM ~~≥ 1.0% Δk/k~~ within the limit provided in the COLR. Therefore, 20 1

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

surveillance of the ~~$F_Q(Z)$, the $F^N_{\Delta H}$~~ LHR and SDM is required to verify that their limits are not exceeded. The limits for the ~~nuclear hot channel factors~~ are specified in the COLR. Refer to the Bases for LCO 3.2.5 for a complete discussion of ~~$F_Q(Z)$ and $F^N_{\Delta H}$~~ LHR. During PHYSICS TESTS, one or more of the LCOs that normally preserve the ~~$F_Q(Z)$ and $F^N_{\Delta H}$~~ LHR limits may be suspended. However, the results of the safety analysis are not adversely impacted if verification that ~~$F_Q(Z)$ and $F^N_{\Delta H}$~~ core LHRs are within their limits is obtained, while one or more of the LCOs is suspended. Therefore, SRs are placed on ~~$F_Q(Z)$ and $F^N_{\Delta H}$~~ LHR during MODE 1 PHYSICS TESTS when THERMAL POWER exceeds 20% RTP to verify that ~~these factors~~ core LHRs remain within their limits. Periodic verification of ~~these factors~~ allows PHYSICS TESTS to be conducted while continuing to maintain the design criteria. (20)

PHYSICS TESTS include measurement of core nuclear parameters or exercise of control components that affect process variables. Among the process variables involved are AXIAL POWER IMBALANCE and QPT, which represent initial condition input (power peaking) for the accident analysis. Also involved are the movable control components, i.e., the regulating rods and the APSRs, which affect power peaking. ~~and are required for shutdown of the reactor.~~ The limits for these variables are specified for each fuel cycle in the COLR. EDIT.

~~PHYSICS TESTS satisfy Criteria 1, 2, and 3 of the NRC Policy Statement.~~ (13)

<INSERT B3.1-47A>

LCO

This LCO permits individual CONTROL RODS and APSR's to be positioned outside of their specified group alignment and withdrawal limits and to be assigned to other than specified CONTROL ROD groups, and permits AXIAL POWER IMBALANCE and QPT limits to be exceeded during the performance of PHYSICS TESTS. In addition, this LCO permits verification of the fundamental core characteristics and nuclear instrumentation operation. (15)

The requirements of LCO 3.1.4, LCO 3.1.5, LCO 3.1.6, LCO 3.2.1 (for the restricted operation region only), LCO 3.2.2, LCO 3.2.3, and LCO 3.2.4 may be suspended during the performance of PHYSICS TESTS provided: (15)

a. THERMAL POWER is maintained ≤ 85% RTP;

(continued)

<INSERT B 3.1-47A>

As described in LCO 3.0.7, compliance with Test Exception LCOs is optional, and therefore no criteria of 10CFR50.36 (Ref. 6) apply. Test Exception LCOs provide flexibility to perform certain operations by appropriately modifying requirements of other LCOs. A discussion of the criteria for the other LCOs is provided in their respective Bases.

BASES

LCO (continued)

b. Nuclear overpower trip setpoint is ≤ 10% RTP higher than the THERMAL POWER at which the test is performed, with a maximum setting of 90% RTP;

LHR is

c. ~~$F_Q(Z)$ and $F^N_{\Delta H}$ are~~ maintained within limits specified in the COLR; and 20

while operating at greater than 20% RTP

d. SDM is ~~maintained ≥ 1.0% Δk/k~~. verified to be within the limit provided in the COLR 1

Operation with THERMAL POWER ≤ 85% RTP during PHYSICS TESTS provides an acceptable thermal margin when one or more of the applicable LCOs is out of specification. Eighty-five percent RTP is consistent with the maximum power level for conducting the intermediate core power distribution test specified in Reference ~~4~~ 3. The nuclear overpower trip setpoint is reduced so that a similar margin exists between the steady state condition and trip setpoint as exists during normal operation at RTP. edit

<INSERT B3.1-48A> 20

APPLICABILITY

This LCO is applicable in MODE 1, when the reactor has completed low power testing and is in power ascension, or during power operation with THERMAL POWER > 5% RTP but ≤ 85% RTP. This LCO is applicable for power ascension testing, as ~~defined by Regulatory Guide 1.68~~ described in SAR Section 3A.9 (Ref. 3). In MODE 2, Applicability of this LCO is not required because LCO 3.1.9, "PHYSICS TESTS Exceptions—MODE 2," addresses PHYSICS TESTS exceptions initiated in MODE 2. In MODES 3, 4, 5, and 6, Applicability is not required because PHYSICS TESTS are not performed in these MODES. 15 24

ACTIONS

A.1 and A.2

If the SDM requirements are not met, boration must be initiated promptly. A Completion Time of 15 minutes is adequate for an operator to correctly align and start the required systems and components. The operator should begin boration with the best source available for the ~~plant~~ unit conditions. Boration will be continued until SDM is within limit. In the determination of the required combination of boration flow rate and boron concentration, there is no unique requirement that must be satisfied. EDIT

(continued)

<INSERT B 3.1-48A>

LCO provision c is modified by a Note that requires the adherence to LHR requirements only when THERMAL POWER is greater than 20% RTP. This establishes an LCO provision that is consistent with the Applicability of LCO 3.2.5, "Power Peaking."

BASES

ACTIONS

A.1 and A.2 (continued)

Suspension of PHYSICS TESTS exceptions requires restoration of each of the applicable LCOs to within specification. A Completion Time of one hour is provided for the operator to restore compliance with the excepted LCOs.

EDIT.

B.1

If THERMAL POWER exceeds 85% RTP, then 1 hour is allowed for the operator to reduce THERMAL POWER to within limits or to complete an orderly suspension of PHYSICS TESTS exceptions. Suspension of PHYSICS TESTS exceptions requires restoration of each of the applicable individual LCOs to within specification. This required Completion Time is consistent with, or more conservative than, those specified for the individual LCO, addressed by PHYSICS TESTS exceptions.

If the nuclear overpower trip setpoint is not within the specified limits, then 1 hour is allowed for the operator to restore the nuclear overpower trip setpoint within limits or to complete an orderly suspension of PHYSICS TESTS exceptions. Suspension of PHYSICS TESTS exceptions requires restoration of each of the applicable individual LCOs to within specification. This required Completion Time is consistent with, or more conservative than, those specified for the individual LCO, addressed by these PHYSICS TESTS exceptions.

If the results of the incore flux map indicate that ~~either $F_Q(Z)$ or $F_{\Delta H}^N$~~ LHR has exceeded its limit, then PHYSICS TESTS are suspended. This action is required because of direct indication that the core ~~peaking factors~~ LHR, which ~~are~~ is a fundamental initial condition~~s~~ for the safety analysis, ~~are~~ is excessive. Suspension of PHYSICS TESTS exceptions requires restoration of each of the applicable LCOs to within specification. (20)

<INSERT B3.1-49A>

SURVEILLANCE REQUIREMENTS

SR 3.1.8.1

Verification that THERMAL POWER is ≤ 85% RTP ensures that the required additional thermal margin has been established prior to and during PHYSICS TESTS. The required Frequency of once per hour allows the operator adequate time to

(continued)

<INSERT B 3.1-49A>

This Condition is modified by a Note that requires performance of the Required Action only when THERMAL POWER is greater than 20% RTP. This establishes an ACTIONS entry Condition that is consistent with LCO provision c and the Applicability of LCO 3.2.5, "Power Peaking."

BASES

SURVEILLANCE REQUIREMENTS

SR 3.1.8.1 (continued)

determine any degradation of the established thermal margin during PHYSICS TESTS.

SR 3.1.8.2

Verification that ~~$F_Q(Z)$ and $F^N_{\Delta H}$~~ core LHRs are within their limits ensures that core ~~local linear heat rate~~ LHR and departure from nucleate boiling ratio will remain within their limits, while one or more of the LCOs that normally control these design limits are out of specification. The required Frequency of 2 hours allows the operator adequate time for collecting a flux map and for performing the ~~hot channel factor~~ LHR verification~~s~~, based on operating experience. If SR 3.2.5.1 is not met, PHYSICS TESTS are suspended and LCO 3.2.5 applies. This Frequency is more conservative than the Completion Time for restoration of the individual LCOs that preserve the ~~$F_Q(Z)$ and $F^N_{\Delta H}$~~ LHR limits. (20)

<INSERT B3.1-50A>

SR 3.1.8.3

Verification that the nuclear overpower trip setpoint is within the limit specified for each PHYSICS TEST ensures that core protection at the reduced power level is established ~~and will remain in place~~ within during the PHYSICS TESTS. Performing the verification once ~~every 8 hours~~ prior to the performance of PHYSICS TESTS at each testing plateau allows the operator adequate time for ~~determining any degradation of~~ verifying the established trip setpoint ~~margin~~ before ~~and during~~ initiating PHYSICS TESTS~~, and for adjusting the nuclear overpower trip setpoint~~. (2)

3.1-12

SR 3.1.8.4

The SDM is verified by performing a reactivity balance calculation, considering the following reactivity effects:

a. Reactor Coolant System (RCS) boron concentration;

b. CONTROL ROD position;

c. ~~RCS average temperature;~~ Doppler defect; (39)

(continued)

<INSERT B 3.1-50A>

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

BASES

SURVEILLANCE REQUIREMENTS

SR 3.1.8.4 (continued) EDIT.

d. Fuel burnup based on gross thermal energy generation;

e. Samarium concentration;

f. ~~e.~~ ~~Xenon concentration; and~~

Moderator defect

g. ~~f.~~ ~~Isothermal temperature coefficient (ITC).~~ 39

~~Using the ITC accounts for Doppler reactivity in this calculation because the reactor is subcritical and the fuel temperature will be changing at the same rate as the RCS.~~

The Frequency of 24 hours is based on the generally slow change in required boron concentration and on the low probability of an accident occurring without the required SDM.

REFERENCES

1. 10 CFR 50, Appendix B, Section XI.
2. 10 CFR 50.59. EDIT.

~~3. Regulatory Guide 1.68, Revision 2, August 1978.~~ 15

~~4. ANSI/ANS-19.6.1-1985, December 13, 1985.~~

4. ~~5.~~ ~~F~~SAR, Section ~~[13.4.8]~~ 13.3, 13.4 and 13.6. EDIT.

5. ~~6.~~ ~~F~~SAR, Section 13.4, Table 13-2 ~~[13.4.8], [Tables 13-2 and 13-4, Am. 49, September 30, 1976]~~. EDIT.

6. 10 CFR 50.36. 18

3. SAR, Section 3A.9 15

B 3.1 REACTIVITY CONTROL SYSTEMS

B 3.1.9 PHYSICS TESTS Exceptions—MODE 2

edit

BASES

BACKGROUND

The purpose of this MODE 2 LCO is to permit PHYSICS TESTS to be conducted by providing exemptions from the requirements of other LCOs. Establishment of a test program to verify that structures, systems, and components will perform satisfactorily in service is required by Section XI of 10 CFR 50, Appendix B (Ref. 1). Testing is required as an integral part of the design, fabrication, construction, and operation of the unit ~~power plant~~. All functions necessary to ensure that specified design conditions are not violated during normal operation and abnormalities ~~anticipated operational occurrences~~ must be tested. Requirements for notification of the NRC, for the purpose of conducting tests and experiments, are specified in 10 CFR 50.59 (Ref. 2).

edit

edit

edit

The key objectives of a test program are to ~~(Ref. 3)~~:

edit

a. Ensure that the facility has been adequately designed;

b. Validate the analytical models used in the design and analysis;

c. Verify the assumptions used to predict unit response;

d. Ensure that installation of equipment in the facility has been accomplished in accordance with the design; and

e. Verify that the operating and emergency procedures are adequate.

To accomplish these objectives, testing is performed prior to initial criticality; during startup, low power operations, and power ascension; at high powers; and after each refueling. The PHYSICS TESTS requirements for reload fuel cycles ensure that the operating characteristics of the core are consistent with the design predictions, and that the core can be operated as designed (Ref. ~~4~~ 3).

<Insert B 3.1-52A>

PHYSICS TESTS procedures are written and approved in accordance with established guidelines. The procedures include all information necessary to permit a detailed

15

(continued)

<INSERT B 3.1-52A>

The inclusion of this PHYSICS TESTS Exception LCO is acceptable based on the use of approved written procedures, administrative controls, the requirements of 10CFR50.59, and the LCO 3.1.9 provisions in effect during the conduct of PHYSICS TESTS.

BASES

BACKGROUND (continued)

execution of testing required to ensure that the design intent is met. PHYSICS TESTS are performed in accordance with these procedures, and test results are approved prior to continued power escalation and long term power operation.

Examples of MODE 2 PHYSICS TESTS include determination of critical boron concentration, CONTROL ROD group worth, and reactivity coefficients.

APPLICABLE SAFETY ANALYSES

Reference ~~5~~ 4 ~~defines requirements for~~ describes the initial testing of the facility, including PHYSICS TESTS. Tables ~~13-3 and 13-4~~ 13-2 (Ref. ~~6~~ 5) summarize ~~s~~ the ~~zero, low power, and power~~ post-criticality tests. Requirements for reload fuel cycle PHYSICS TESTS are given in ~~Table 1 of ANSI/ANS-19.6.1-1985~~ SAR Section 3A.9 (Ref. ~~4~~ 3). Although these PHYSICS TESTS are generally accomplished within the limits of all LCOs, conditions may occur when one or more of the LCOs must be suspended to make completion of PHYSICS TESTS possible or practical. EDIT. EDIT. 15

It is acceptable to suspend the following LCOs for PHYSICS TESTS because reactor protection criteria are preserved by the LCOs still maintained and by the SRs:

LCO 3.1.3, "Moderator Temperature Coefficient (MTC)";
LCO 3.1.4, "CONTROL ROD Group Alignment Limits";
LCO 3.1.5, "Safety Rod Insertion Limits";
LCO 3.1.6, "AXIAL POWER SHAPING ROD (APSR) Alignment Limits";
LCO 3.2.1, "Regulating Rod Insertion Limits" ~~for the restricted operation region only~~; ~~and~~ 42
LCO 3.2.2, "AXIAL POWER SHAPING ROD (APSR) Insertion Limits"; and
LCO 3.4.2, "RCS Minimum Temperature for Criticality." 15

Even if an accident occurs during PHYSICS TESTS with one or more LCOs suspended, fuel damage criteria are preserved because the limits on THERMAL POWER and shutdown capability are maintained during the PHYSICS TESTS.

Shutdown capability is preserved by limiting ~~maximum obtainable~~ THERMAL POWER and maintaining adequate SDM, when in MODE 2 PHYSICS TESTS. In MODE 2, the Reactor Coolant System (RCS) temperature must be within the narrow range instrumentation for ~~plant~~ unit control. The narrow range temperature instrumentation goes on scale at 520°F. Therefore, it is considered safe to allow the minimum RCS EDIT. EDIT.

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

temperature to decrease to 520°F during MODE 2 PHYSICS TESTS, based on the low probability of an accident occurring and on prior operating experience.

PHYSICS TESTS include measurement of core nuclear parameters or exercise of control components that affect process variables.

~~PHYSICS TESTS satisfy Criteria 1, 2, and 3 of the NRC Policy Statement.~~ 13

<INSERT B 3.1-54B>>

LCO

This LCO permits individual CONTROL RODS and APSRs to be positioned outside of their specified group alignment and withdrawal limits and to be assigned to other than specified CONTROL ROD groups during the performance of PHYSICS TESTS. In addition, this LCO permits verification of the fundamental core characteristics. EDIT.

This LCO also allows suspension of LCO 3.1.3, LCO 3.1.4, LCO 3.1.5, LCO 3.1.6, LCO 3.2.1, LCO 3.2.2, ~~and LCO 3.4.2~~, provided: 15

a. THERMAL POWER is ≤ 5% RTP;

b. Nuclear overpower trip setpoints on the OPERABLE nuclear power range channels are set to ≤ ~~25%~~ 5% RTP; 23

c. Nuclear instrumentation ~~source range and intermediate range~~ high startup rate CONTROL ROD withdrawal inhibit ~~are~~ 15 OPERABLE; and 25

d. SDM is maintained ~~≥ [1.0]% Δk/k~~ within the limit provided in the COLR. 1

The limits of LCO 3.2.3 and LCO 3.2.4 are not exempted by this specification because they do not apply in MODE 2. Inhibiting CONTROL ROD withdrawal, based on startup rate, also limits local linear heat rate (LHR), departure from nucleate boiling ratio (DNBR), and peak RCS pressure during accidents initiated from low power. EDIT.

APPLICABILITY

This LCO is applicable ~~in MODE 2~~ when the reactor is either subcritical or ~~not~~ critical with ~~or when~~ THERMAL POWER ~~is~~ ≤ 5% RTP. This LCO is applicable for initial criticality or low power testing, as ~~defined by Regulatory Guide 1.68~~ described in SAR, Section 3A.9 (Ref. 3). In MODE 1, 24 15

<INSERT B 3.1-54A>

(continued)

<INSERT B 3.1-54A>

The Applicability is stated as "during PHYSICS TESTS initiated in MODE 2" to ensure that the 5% RTP maximum power level is not exceeded. Should the THERMAL POWER exceed 5% RTP, and consequently the unit enter MODE 1, this Applicability statement prevents exiting this Specification and its Required Actions.

<INSERT B 3.1-54B>

As described in LCO 3.0.7, compliance with Test Exception LCOs is optional, and therefore no criteria of 10CFR50.36 (Ref. 6) apply. Test Exception LCOs provide flexibility to perform certain operations by appropriately modifying requirements of other LCOs. A discussion of the criteria for the other LCOs is provided in their respective Bases.

BASES

APPLICABILITY (continued)

Applicability of this LCO is not required because LCO 3.1.8, "PHYSICS TESTS Exceptions," addresses PHYSICS TESTS exceptions in MODE 1. In MODES 3, 4, 5, and 6, a test exception LCO ~~Applicability~~ is not required because ~~physics testing is not performed in these MODES~~ the excepted LCOs do not apply in these MODES. EDIT. EDIT.

ACTIONS

A.1

If THERMAL POWER exceeds 5% RTP, a positive reactivity addition could be occurring, and a nuclear excursion could result. To ensure that local LHR, DNBR, and RCS pressure limits are not violated, the reactor is immediately tripped. The necessary prompt action requires manual operator action to open the CONTROL ROD drive trip breakers without attempts to reduce THERMAL POWER by actuating the control system (i.e., CONTROL ROD insertion or RCS boration). EDIT. EDIT.

B.1 and B.2

If the SDM requirements are not met, boration must be initiated promptly. A Completion Time of 15 minutes is adequate for an operator to correctly align and start the required systems and components. The operator should begin boration with the best source available for the ~~plant~~ unit conditions. Boration will be continued until SDM is within limit. In the determination of the required combination of boration flow rate and boron concentration, there is no unique requirement that must be satisfied. EDIT.

Suspension of PHYSICS TESTS exceptions requires restoration ~~of each of the applicable LCOs to within specification.~~ a Completion Time of one hour is provided for the operator to restore compliance with the excepted LCOs. EDIT.

C.1

If the nuclear overpower trip setpoint is > ~~25%~~ 5% RTP, then 1 hour is allowed for the operator to restore the nuclear overpower trip setpoint within limits or to complete an orderly suspension of PHYSICS TESTS exceptions. Suspension of PHYSICS TESTS exceptions requires restoration of each of the applicable individual LCOs to within specification, in order to ensure that continuity of reactor operation is within initial condition limits. This required Completion 23

(continued)

BASES

ACTIONS

C.1 (continued)

Time is consistent with, or more conservative than, those specified for the individual LCOs addressed by PHYSICS TESTS exceptions.

If the nuclear instrumentation ~~source and intermediate range~~ high startup rate CONTROL ROD withdrawal inhibit function~~s~~ ~~are~~ 15 inoperable, then 1 hour is allowed for the operator to restore the functions to OPERABLE status or to complete an orderly suspension of PHYSICS TESTS exceptions. Suspension of PHYSICS TESTS exceptions requires restoration of each of the applicable individual LCOs to within specification. This required Completion Time is consistent with, or more conservative than, those specified for the individual LCOs addressed by PHYSICS TESTS exceptions. 25

<INSERT B3.1-56 A> 25

SURVEILLANCE REQUIREMENTS

~~SR 3.1.9.1~~

~~Performing a CHANNEL FUNCTIONAL TEST on each nuclear instrumentation source and intermediate range high startup rate CONTROL ROD withdrawal inhibit and nuclear overpower channel, ensures that the instrumentation required to detect a deviation from THERMAL POWER or to detect a high startup rate is OPERABLE. Performing the test once within 24 hours, prior to initiating PHYSICS TESTS, ensures that the instrumentation is OPERABLE shortly before PHYSICS TESTS begin and allows the operator to correct any instrumentation problems.~~ 26

SR 3.1.9.~~2~~ 1 26

Verification that THERMAL POWER is ≤ 5% RTP ensures that ~~an adequate margin is maintained between the THERMAL POWER level and the nuclear overpower trip setpoint.~~ local LHR, DNBR, and RCS pressure limits are not violated and that entry into Actions Condition A is performed promptly. Hourly verification is adequate for the operator to determine any change in core conditions, such as xenon redistribution occurring after a THERMAL POWER reduction, that could cause THERMAL POWER to exceed the specified limit. 23

(continued)

<INSERT B 3.1-56A>

The nuclear instrumentation high startup rate CONTROL ROD withdrawal inhibit function is not required when the reactor power level is above the operating range of the instrumentation channel. For example, if the reactor power level is above the source range channel operating range, then only the intermediate range high startup rate CONTROL ROD withdrawal inhibit is required to be functional.

BASES

SURVEILLANCE REQUIREMENTS (continued)

3/1-12

SR 3.1.9.~~3~~ 2 | 26

Verification that the nuclear overpower trip setpoint is within the limit specified for PHYSICS TESTS ensures that core protection at the reduced power level is established ~~and will remain in place~~ during within PHYSICS TESTS. Performing the verification once ~~per~~ within 8 hours prior to the performance of PHYSICS TESTS allows the operator adequate time for ~~determining any degradation of~~ verifying the established trip setpoint ~~margin~~ before ~~and during~~ Initiating PHYSICS TESTS. ~~and for adjusting the nuclear overpower trip setpoint.~~ | 2

SR 3.1.9.~~4~~ 3 | 26

The SDM is verified by performing a reactivity balance calculation, considering the following reactivity effects:

a. RCS boron concentration;

b. CONTROL ROD position;

c. RCS average temperature;

~~d. Fuel burnup based on gross thermal energy generation;~~

e. Samarium concentration;

~~e.~~ f. Xenon concentration; ~~and~~

~~f.~~ g. Isothermal temperature coefficient (ITC), when below the point of adding heat (POAH);

h. Moderator defect, when above the POAH; and

i. Doppler defect, when above the POAH.

Using the ITC accounts for Doppler reactivity in this calculation ~~because~~ when the reactor is subcritical, or critical but below the POAH, and the fuel temperature will be changing at the same rate as the RCS. | 39

The Frequency of 24 hours is based on the generally slow change in required boron concentration and on the low probability of an accident occurring without the required SDM.

REFERENCES

1. 10 CFR 50, Appendix B, Section XI.

2. 10 CFR 50.59.

~~3. Regulatory Guide 1.68, Revision 2, August 1978.~~ | 15

3. SAR, Section 3A.9.

(continued)

BASES

REFERENCES (continued)

~~4.~~ ANSI/ANS-~~19.6.1-1985, December 13, 1985~~ (15)

4. ~~5.~~ FSAR, Section ~~[13.4.8].~~ 13.3, 13.4 and 13.6. EDIT.

5. ~~6.~~ FSAR, Section ~~[13.4.8], [Table 13-3 and Table 13-4].~~ 13.4, Table 13-2. EDIT.

6. 10CFR50.36. (18)

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
SR 3.2.1.3	Verify SDM $\geq$ 1% $\Delta k/k$.	Within 4 hours prior to achieving criticality

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.	Once per 2 hours
	<u>OR</u>	
	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 <u>OR</u> 2 hours after last performance of SR 3.2.5.1
	<u>AND</u>	
	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.	10 hours <u>OR</u> 10 hours after last performance of SR 3.2.5.1
	<u>AND</u>	

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 <u>AND</u> 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.

<u>A.1</u>

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.

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.

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.

<u>A.2</u>

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.

<u>B.1</u>

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

<u>SR 3.2.3.1</u>

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.

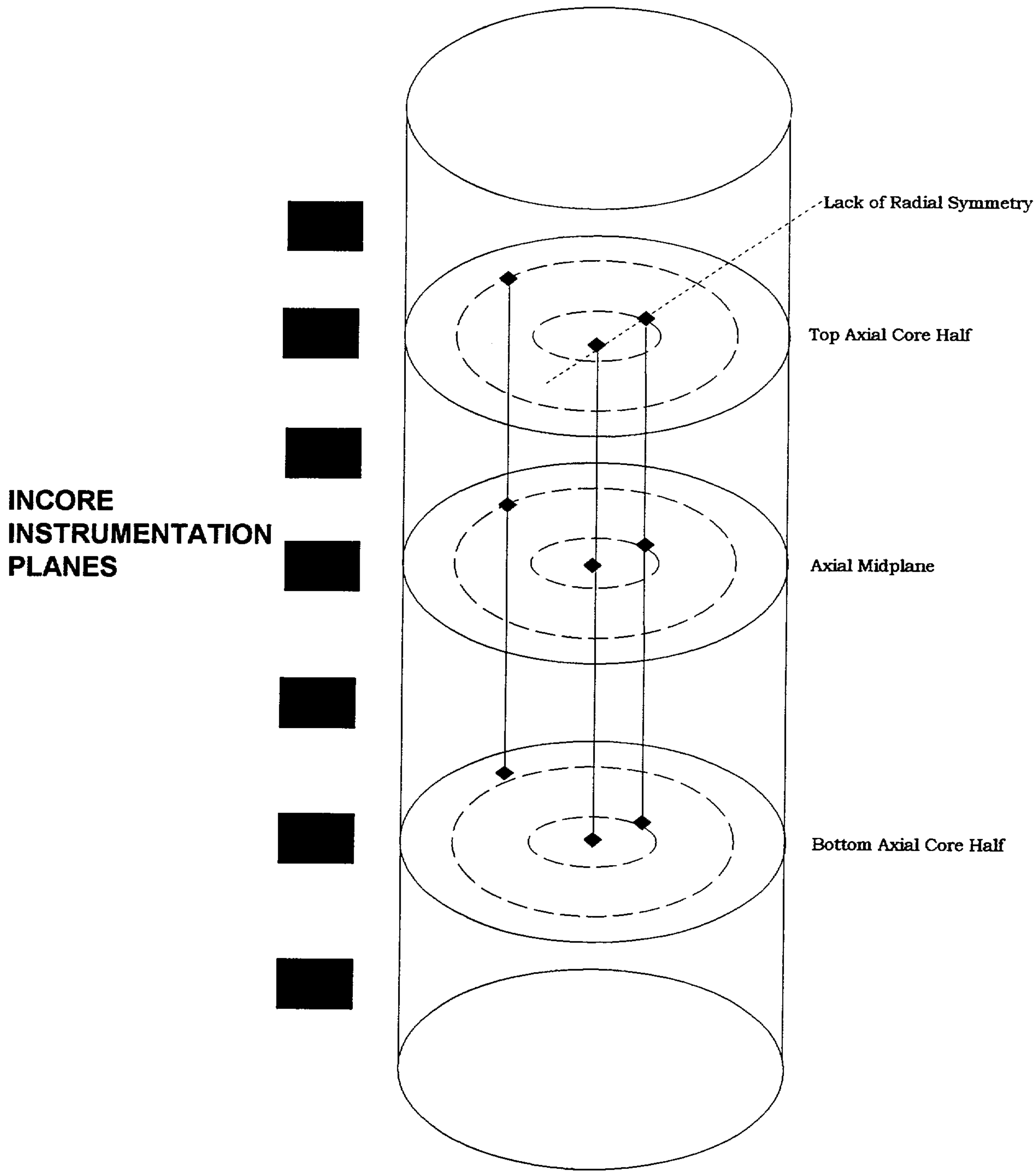


Figure B 3.2.3-1 (page 1 of 1)
Minimum Incore System for AXIAL POWER IMBALANCE Measurement

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

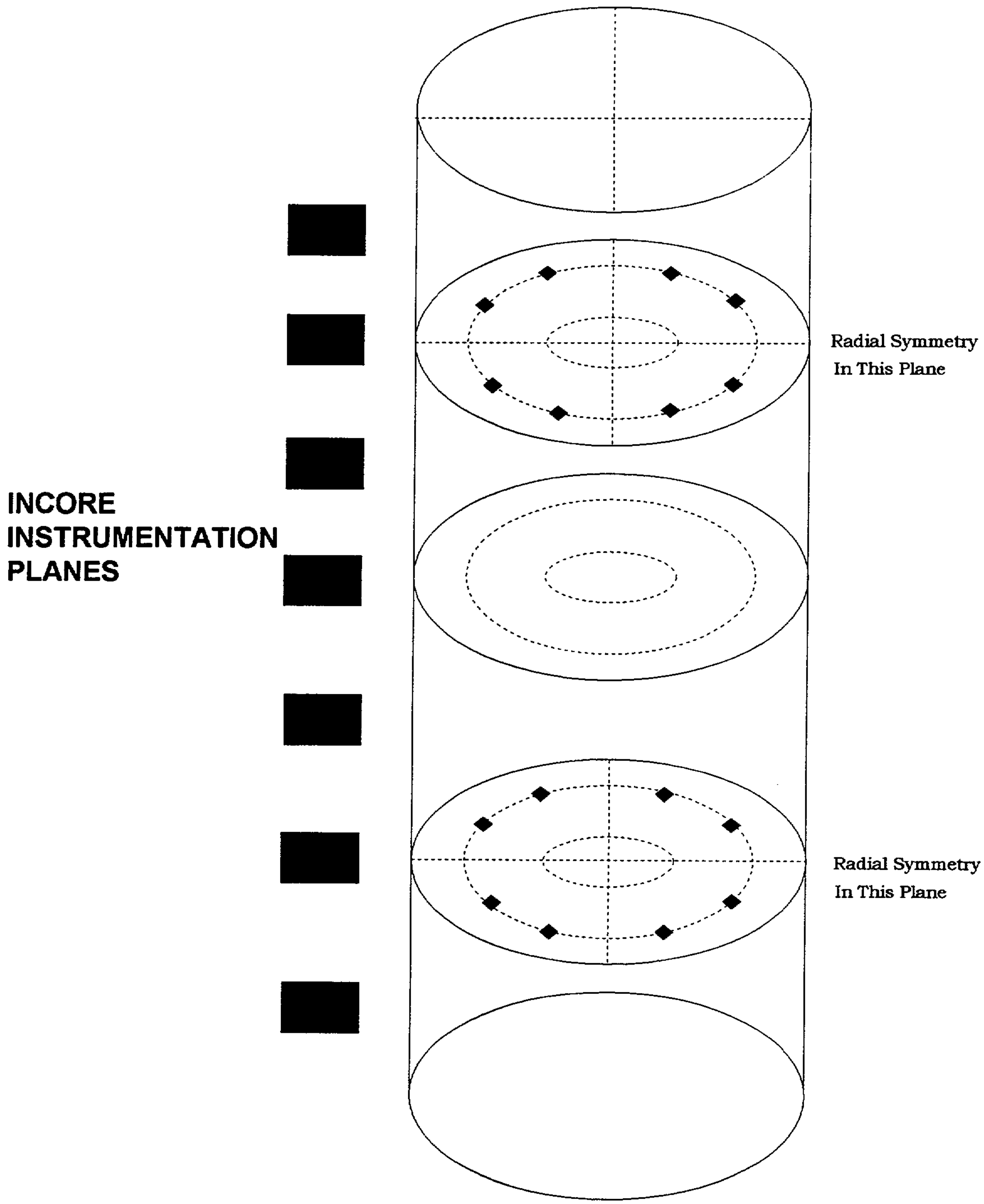


Figure B 3.2.4-1 (page 1 of 1)
Minimum Incore System for QUADRANT POWER Tilt Measurement

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.

CTS DISCUSSION OF CHANGES

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.

CTS DISCUSSION OF CHANGES

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.

M19 The CTS markup shows the adoption of ITS SR 3.2.1.3. This SR requires a verification that SDM is ≥ 1%Δk/k within 4 hours prior to achieving criticality. The SR verifies that there is sufficient SDM with the control rods at the estimated critical position if it is necessary to shutdown or trip the reactor following criticality. The adoption of this SR is more restrictive because the CTS had no similar SR requirement. This change is consistent with NUREG-1430.

3.2-02

CTS DISCUSSION OF CHANGES

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.

CTS DISCUSSION OF CHANGES

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.

CTS DISCUSSION OF CHANGES

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)
3.2-04	3.5.4 Specification (23 detectors)	TRM
	3.5.4.1	Bases (3.2.3)
	3.5.4.2	Bases (3.2.4)
	Table 4.1-1, Item 39	TRM

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

Specification

<LATER> (3.4A) 3.1.3.1 The reactor coolant temperature shall be above 525F — LATER except for portions of low power physics testing when the requirements of Specification 3.1.8 shall apply. — LATER <LATER> (3.1)

<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 $\Delta k/k$ 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) 3.1.3.5 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

3.2.1 Appl LATER (3.1)

<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 — A2

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

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

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

During physics tests, special operating precautions will be taken. In addition, the strong negative Doppler coefficient (1) and the small integrated $\Delta k/k$ 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

& LATER

3.2.1 Appl
3.2.2 Appl

Applicability

This specification applies to power distribution and operation of control rods during power operation. MODES 1 & 2 — M3

(LATER) (3.1)

~~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.~~ — LATER

Specification

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

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. — L8 — & LATER

(LATER) (3.1)

~~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.~~ — LATER

~~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 3.2.1.3> — M19

3.1-02

<Add SR 3.2.4.1 Frequency> M16

<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

3.2.4

<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.~~

3.5.2.4 Quadrant Power Tilt (QPT) A1

<LATER> (3.1) — LATER

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 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~~ ≤20% RTP within 4 hours. ~~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 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. ~~Technical 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

within the limits specified in the COLR

<LATER> (3.1)

<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

3.2.1 RA D.2.1 & D.2.2

RA A.2

RA B.1

RA C.1

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

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 ~~2~~ 24 hours. L4

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

SR 3.2.3.1

3.2.3 APPL.

1. AXIAL ~~Reactor~~ 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⟩ (3.1)

3.2.3 LCO

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

3.2.3 RA A.2

3. If the AXIAL ~~reactor~~ 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 AXIAL ~~reactor~~ power imbalance is not achieved within ~~4~~ 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~~

~~REFERENCES~~

~~(1) FSAR, Section 3.2.2.1.2~~

~~(2) FSAR, Section 14.2.2.2~~

A2

3.2.3
3.2.4

3.2-04

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

LA1 TRM

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

3.2.3
3.2.4

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**~~

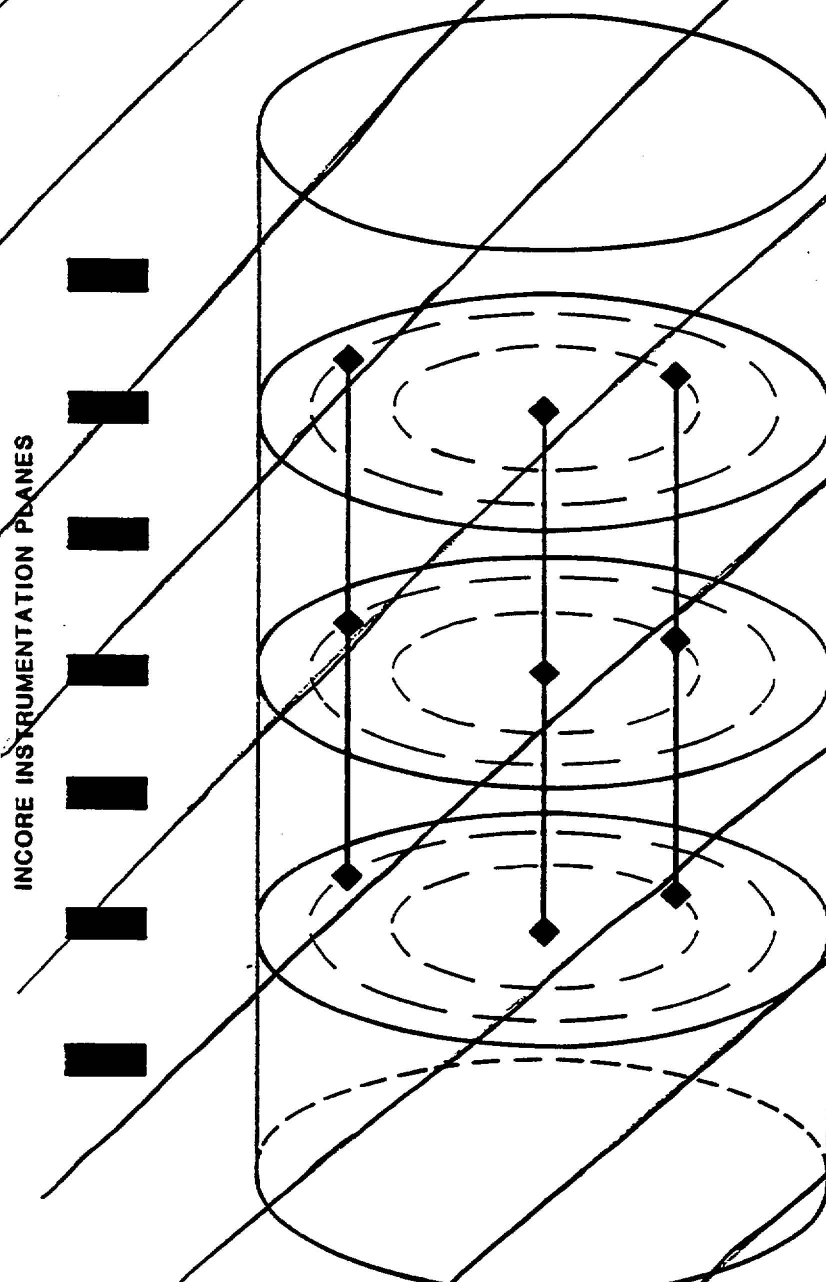
~~(1) FSAR, Section 4.1.1.3~~

A2

3.2.3
3.2.4

INCORE INSTRUMENTATION SPECIFICATION
AXIAL IMBALANCE INDICATION

Figure No. 3.5.4-1

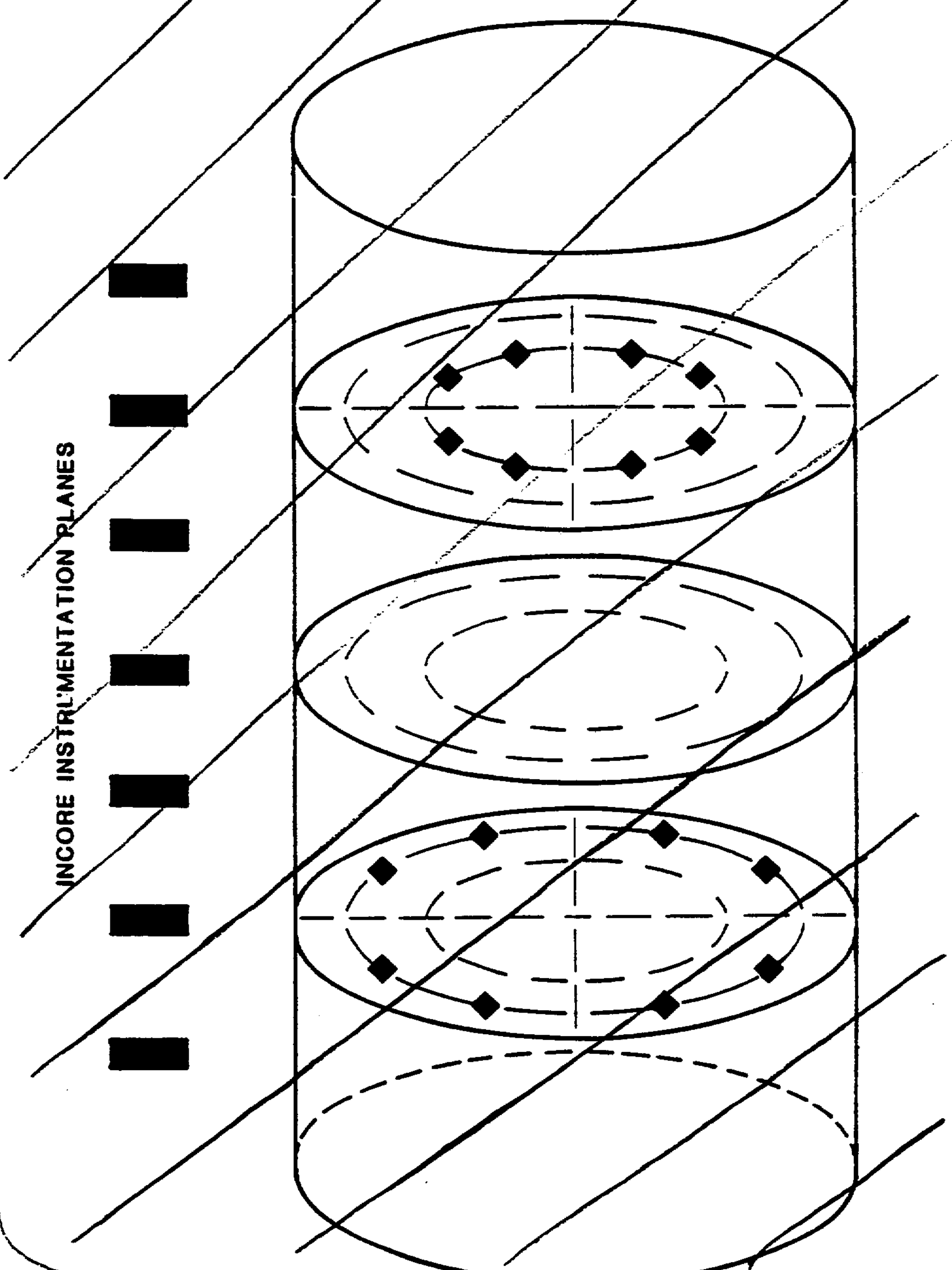


A2

3.2.3
3.2.4

INCORE INSTRUMENTATION SPECIFICAITON
RADIAL FLUX TILT INDICATION

Figure No. 3.5.4-2

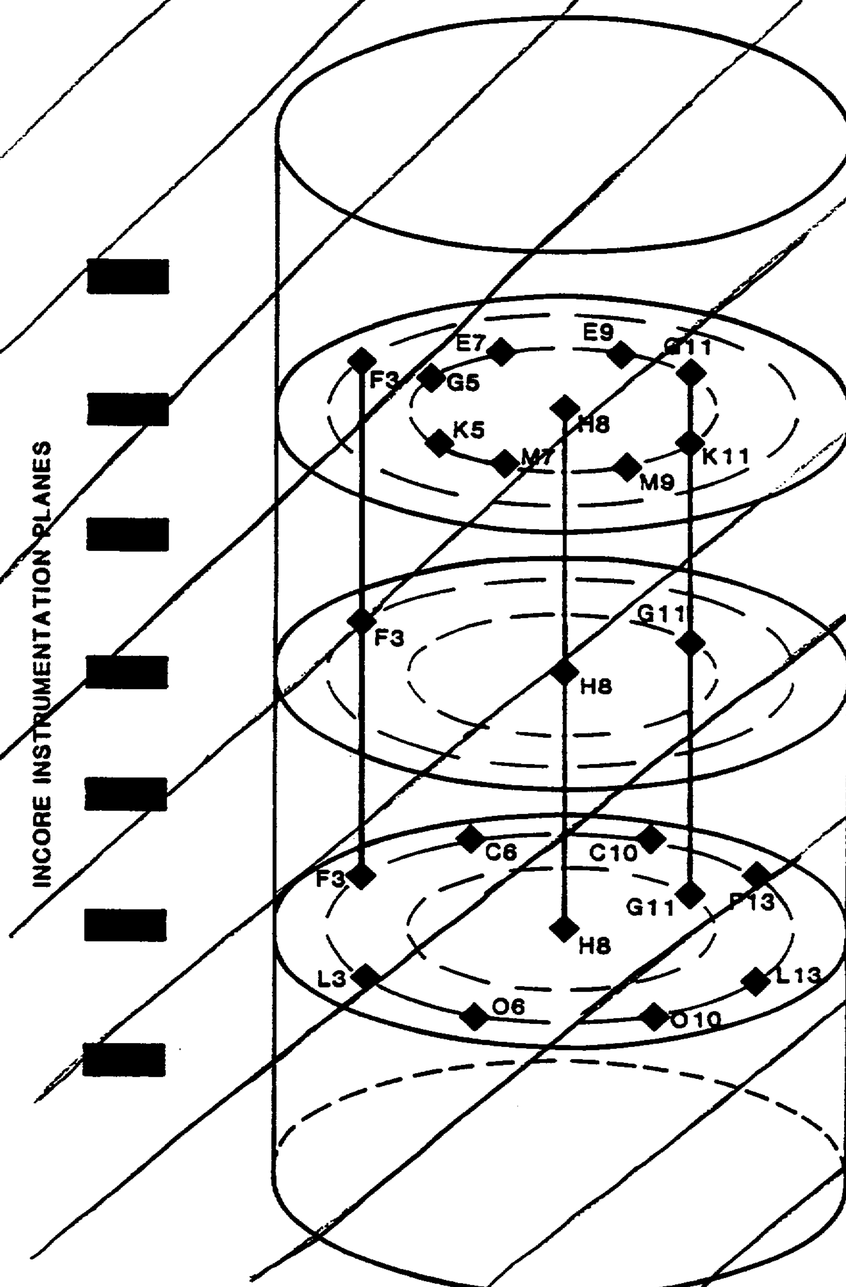


A2

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 ~~map shall be made to verify the expected~~ power distribution (as specified by applicable LCO(s)) ~~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	W	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
GENERIC EVALUATIONS

<u>"R" - Relocation of requirements:</u>

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

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

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

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

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

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

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

NO SIGNIFICANT HAZARDS CONSIDERATIONS
GENERIC EVALUATIONS

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

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

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

   The proposed change relocates requirements and surveillances for structures, systems, components or variables which did not meet the criteria for inclusion in Technical Specifications as identified in the Application of Selection Criteria to the ANO-1 Technical Specifications. The affected structures, systems, components or variables are not assumed to be initiators of analyzed events and are not assumed to mitigate accident or transient events. The requirements and surveillances for these affected structures, systems, components or variables will be relocated from the Technical Specifications to an appropriate administratively controlled license basis document and maintained pursuant to the applicable regulatory requirements. Therefore, this change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

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

   The proposed change does not necessitate a physical alteration of the plant (no new or different type of equipment will be installed) or change in parameters governing normal plant operation. The proposed change will not impose any different requirements and adequate control of information will be maintained. Thus, this change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

   The proposed change will not reduce a margin of safety because it has no impact on any safety analysis assumptions. In addition, the affected requirement will be relocated to an owner controlled license basis document for which future changes will be evaluated pursuant to the requirements of the applicable regulatory requirements. Therefore, this change does not involve a significant reduction in a margin of safety.

NO SIGNIFICANT HAZARDS CONSIDERATIONS
GENERIC EVALUATIONS

"A" - Administrative changes to requirements:

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

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

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

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

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

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

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

   The proposed change will not significantly reduce the margin of safety because it has no impact on any safety analysis assumptions. This change is administrative in nature. As such, there is no technical change to the requirements and therefore, there is no significant reduction in the margin of safety.

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

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

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

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

   The proposed change relocates requirements from the Technical Specifications to other license basis documents which are under licensee control. The documents containing the relocated requirements will be maintained using the provisions of applicable regulatory requirements. Therefore, this change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

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

   The proposed change does not necessitate a physical alteration of the plant (no new or different type of equipment will be installed) or changes in parameters governing normal plant operation. The proposed change will not impose any different requirements and adequate control of the information will be maintained. Thus, this change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

The proposed change will not reduce a margin of safety because it has no impact on any safety analysis assumptions. In addition, the requirements to be transposed from the Technical Specifications to other license basis documents, which are under licensee control, are the same as the existing Technical Specifications. The documents containing the relocated requirements will be maintained using the provisions of applicable regulatory requirements. Therefore, this change does not involve a significant reduction in a margin of safety.

"M" - More restrictive changes to requirements:

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

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

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

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

   The proposed change provides more stringent requirements for the ANO-1 Technical Specifications. These more stringent requirements are not assumed to be initiators of analyzed events and will not alter assumptions relative to mitigation of accident or transient events. The change has been confirmed to ensure no previously evaluated accident has been adversely affected. The more stringent requirements are imposed to ensure process variables, structures, systems and components are maintained consistent with the safety analyses and licensing basis. Therefore, this change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

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

   The proposed change does not necessitate a physical alteration of the plant (no new or different type of equipment will be installed) or changes in parameters governing normal plant operation. The proposed change does impose different requirements. However, these changes do not impact the safety analysis and licensing basis. Thus, this change does not create the possibility of a new or different kind of accident from any accident previously evaluated for ANO-1.

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

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

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

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

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.

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.

<u>NSHC 3.2 L7</u> 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.

NO SIGNIFICANT HAZARDS CONSIDERATIONS STATEMENTS

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.

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.

NO SIGNIFICANT HAZARDS CONSIDERATIONS STATEMENTS

<u>NSHC 3.2 L11</u>

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.

3.2-02 4. Not used.

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.

3.2 POWER DISTRIBUTION LIMITS

CTS

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
				N/A
A. Regulating rod groups inserted in restricted operational regions, ~~or sequence or overlap, or any combination, not met.~~	A.1	Perform SR 3.2.5.1.	Once per 2 hours	1
	AND			
	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	
D. ~~C.~~ Regulating rod groups inserted in unacceptable ~~operational~~ operation 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
E. ~~D.~~ Required Action(s) and associated Completion Time(s) of Condition(s) 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	N/A

3.2-02

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	
				6
A. APSRs not within limits.	A.1	Perform SR 3.2.5.1.	Once per 2 hours	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. __AND__ A.2 Reduce AXIAL POWER IMBALANCE within limits. To	Once per 2 hours 24 hours	N/A 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.	~~4~~ 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)%~~ 20% RTP. 3.5.2.4.4

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME	
			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	3.5.2.4.1 3.5.2.4.2
		OR	
		2 hours after last performance of SR ~~3.2.1~~ 3.2.5.1	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	3.5.2.4.2.a N/A
		OR 10 hours after last performance of SR 3.2.5.1	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	
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	9 N/A
	AND			
	B ~~C~~.2	Reduce nuclear overpower trip setpoint to ≤ 65.5% of the ALLOWABLE THERMAL POWER.	10 hours	9

(continued)

CTS

ACTIONS (continued)

CONDITION	REQUIRED ACTION		COMPLETION TIME	
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~~ 20% RTP.	4 ~~2~~ hours	7 N/A
D. ~~F.~~ QPT greater than the maximum limit ~~.~~ (specified in the COLR.	D ~~F~~.1	Reduce THERMAL POWER to ≤ ~~20~~ 20% RTP.	4 ~~2~~ hours	9 3.5.2.4.3

CTS

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY	CTS
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

 (31)

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~~	to Restore LHR ~~$F_Q(Z)$~~ to within limit. (1c) (S)	~~24 hours~~

move

N/A

(31)

(continued)

CTS

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
B. $F^N_{\Delta H}$ not within limit.	B.1 Reduce THERMAL POWER ≥ RH(%) RTP (specified in the COLR) for each 1% that $F^N_{\Delta H}$ 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 $F^N_{\Delta H}$ exceeds limit.	8 hours
	AND	
	B.3 Restore $F^N_{\Delta H}$ 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

NA

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
LHR is	Verify ~~$F_Q(Z)$ and $F^N_{\Delta H}$ 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 assumptions of ~~available~~ SDM, and the initial reactivity insertion rate. edit

The applicable criteria for these reactivity and power distribution design requirements are described in ~~10 CFR 50 Appendix A~~ 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 ~~Plants~~ Reactors" (Ref. 2). edit edit edit

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~~ rapid reactivity changes ~~quickly~~ compared with borating or diluting the Reactor Coolant System ~~(RCS)~~. 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 linear heat rate ~~$F_Q(Z)$ 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) and ~~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~~ support the minimum required SDM in MODES 1 and 2. edit

31

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 abnormalities ~~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: 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)

edit

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)~~.

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 ~~and insertion positions~~ 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 ~~3.1.1~~ 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 violate~~s~~ 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

edit
edit
(11)
edit
(1)
edit

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 the unacceptable region are described under Condition B.~~ D

regions

Setpoint

Operation

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

Setpoint

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 ~~($F_Q(Z)$ 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.~~ (11) (1) (31) (1)

LHRs

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.

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.

<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

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1
1

B.1

positioned

operation region

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~~

Setpoints

unit

operation

regulating rod position

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11
edit
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(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)

~~B.1 (continued)~~

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

<INSERT B 3.2-7A>> 1

1

D.1 ~~C.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~~ 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 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
operation

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. H11 H11 H3

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.

3.2-02

REFERENCES

SAR, Section 1.4

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

(continued)

BASES

REFERENCES (continued)

3. (U)FSAR, Chapter ~~Section [ ].~~ (3) edit

4. (U)FSAR, Chapter ~~Section [ ].~~ (14) edit

~~5. FSAR, Section [ ].~~

~~6. FSAR, Section [ ].~~ edit

~~7. FSAR, Section [ ].~~

~~8. FSAR, Section [ ].~~ (1)

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~~ $F^N_{\Delta H}$ 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). In MODE 2 while subcritical, the APSR insertion limits satisfy Criterion 4 of 10CFR50.36. 10

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 MODE~~S~~ 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 ed.

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.

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edit

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<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~~ [redacted] 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 ~~limits~~ 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

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edit

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(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

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

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SURVEILLANCE REQUIREMENTS

SR 3.2.2.1

End of Cycle

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.~~ Verification that the APSRs are within their insertion ~~limits~~ Setpoints at a 12 hour Frequency is sufficient to ensure that the APSR insertion ~~limits~~ are preserved ~~and the computer alarm remains OPERABLE~~. 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~~ The Frequency takes into account other information available in the control room for monitoring the axial power distribution in the reactor core.

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CAP

(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. 10 CFR 50.36. 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) and ~~. Operation within the $F_{\Delta H}$ limits given in the COLR~~ 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

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APPLICABLE SAFETY ANALYSES

The fuel cladding must not sustain damage as a result of normal operation ~~(Condition 1)~~ and ~~anticipated operational occurrences (Condition 2)~~ [illegible]. 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 LHR ~~$F_Q(Z)$ or $F_{\Delta H}$ peaking factors~~ 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. 11

Setpoints

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 (~~$F^N_{\Delta H}$ limits~~) or both. For verification that ~~$F_Q(Z)$ and $F^N_{\Delta H}$~~ 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 $F^N_{\Delta H}$~~ 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

core local LHRs

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. 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

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(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

are not

If the Required Actions 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 ~~$F_Q(Z)$ and $F^N_{\Delta H}$~~ LHR initial condition limits assumed in the accident analyses. The required Completion Time of 2 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. 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 ~~alarm setpoints~~ 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 edit 11

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BASES

SR 3.2.3.1 (Continued)

edit

SURVEILLANCE REQUIREMENTS (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

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edit

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.

REFERENCES

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

10 CFR 50.36.

10

edit

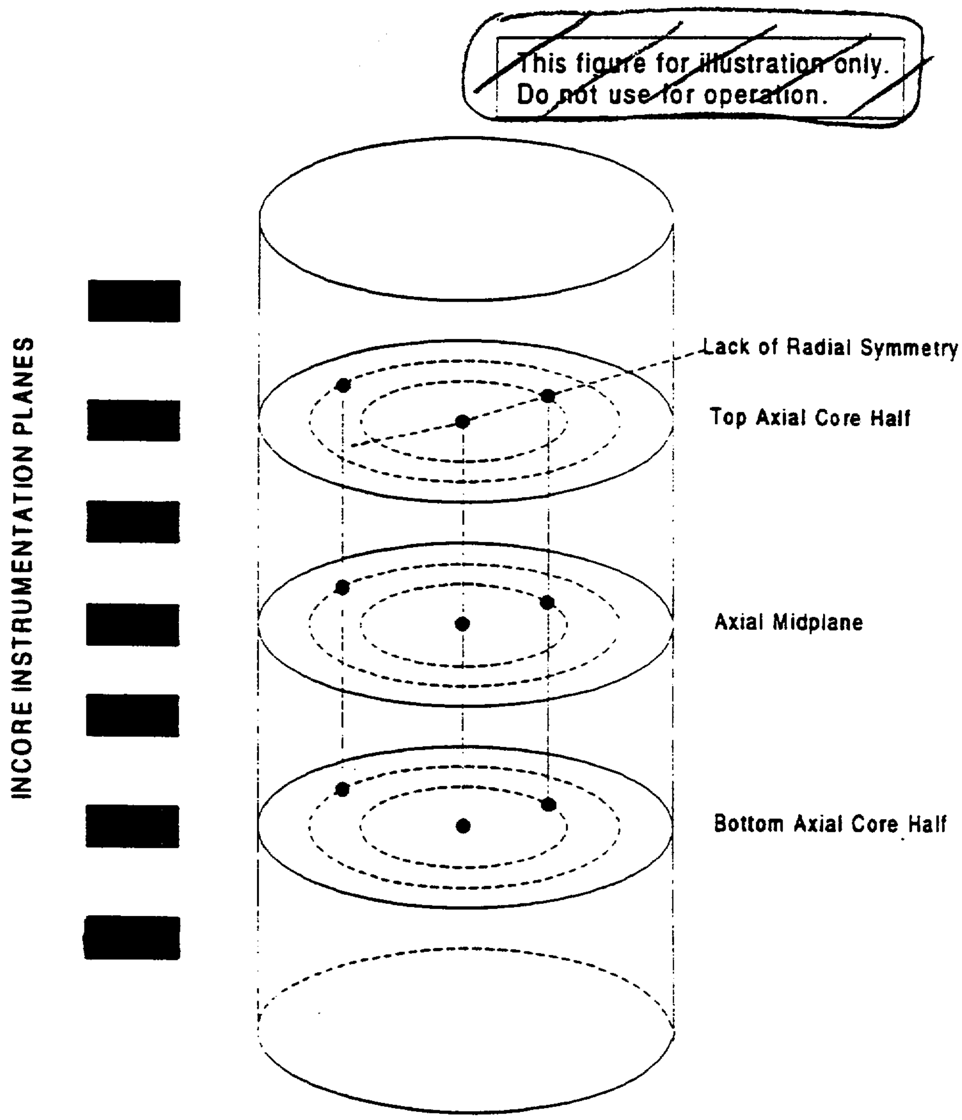

Figure B 3.2.3-1 (page 1 of 1)
Minimum Incore System for AXIAL POWER IMBALANCE Measurement

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). 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.

edit
edit

(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 analytically determined ~~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. ~~3~~ 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

The ~~maximum~~ allowable setpoints for steady state, ~~transient, and maximum~~ ~~limits~~ Setpoints 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 also given in the COLR to allow for system observability and instrumentation errors. 9 11

~~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 reason~~ based on engineering judgment.

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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.

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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~~ 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 ~~($F_{\Delta H}$ limits)~~, or both. For verification that ~~$F_Q(Z)$ and $F_{\Delta H}$~~ 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 $F_{\Delta H}$~~ 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~~. 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~~ is obtained as core conditions that influence QPT change.

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(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~~ 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 margin~~s~~ 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 5 and associated Completion Time 5 of Condition A ~~or B~~ are not met, a further power reduction is required. Power reduction to < 60% of ALLOWABLE THERMAL POWER ~~RTP~~ 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 ~~C~~.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. 9

~~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

ACTIONS C.1 ~~E.1~~ (continued) 9

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

D.1 ~~E.2~~ 9

~~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.~~ 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. 9 11

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. 7 edit

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 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.

Figure B 3.2.4-~~2~~ 1 (Minimum Incore Detector System for QPT Measurement) depicts an example of this configuration. The full symmetric 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.

edit

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

22

(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~~ Setpoint, operation at ≥ 95% RTP may proceed provided the QPT is determined to remain within the ~~steady state~~ ~~limit~~ Setpoint at the increased THERMAL POWER level. In case QPT exceeds the ~~steady state~~ ~~limit~~ Setpoint 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~~ Setpoint again. 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

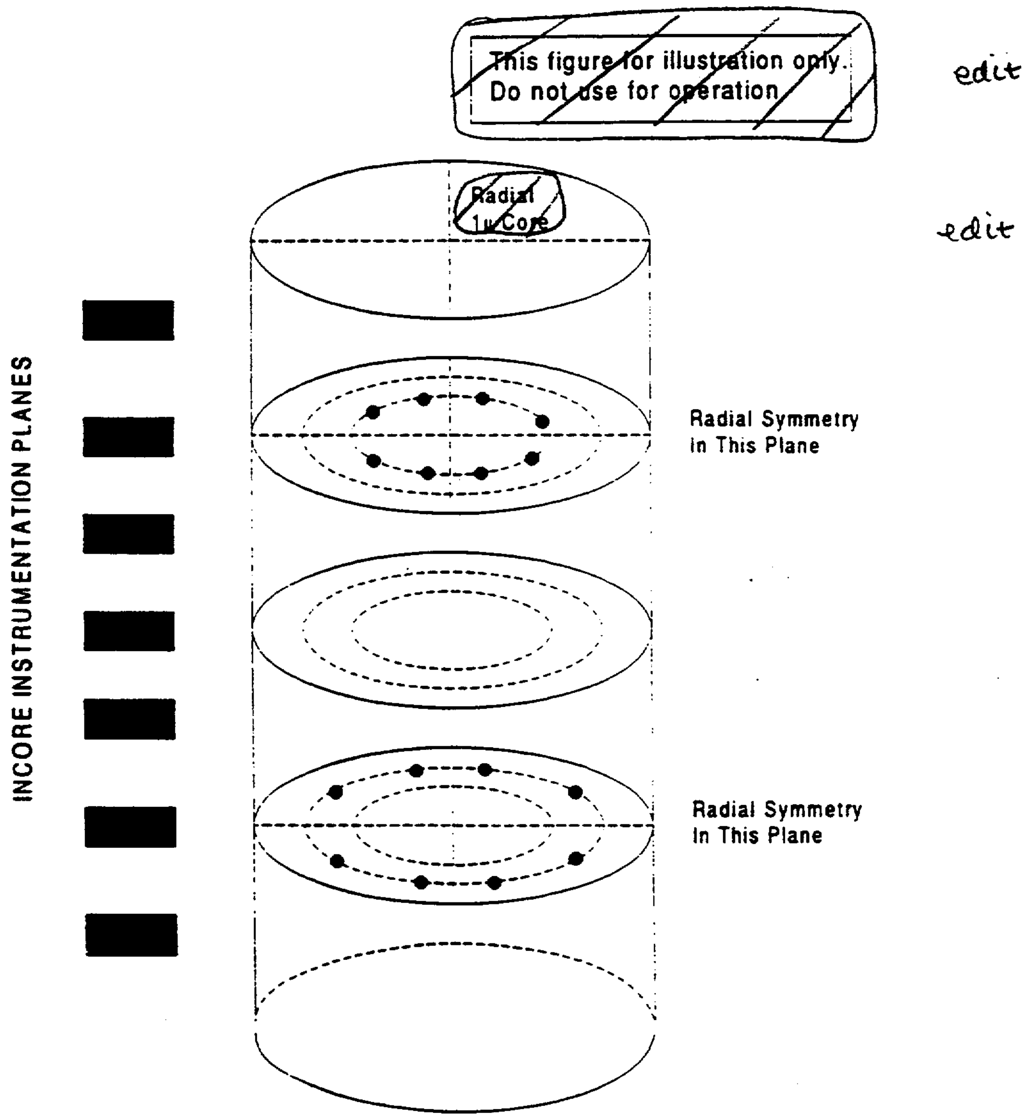

Figure B 3.2.4-1 (page 1 of 1)
Minimum Incore System for QUADRANT POWER TILT Measurement

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. (23) (31)

abnormalities and

<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~~. (31)

<INSERT B 3.2-38B>

by the LOCA LHR figure

generation rates

-limited

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 ~~$F^N_{\Delta H}$ 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)

APPLICABLE SAFETY ANALYSES

LOCA-limited LHR limits

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 $F_{\Delta H}^N$~~ 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~~ that would cause DNB at a particular core location to the actual ~~cladding surface~~ 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

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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).

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

edit 31 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."

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.

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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.~~ (31)

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.~~ (31)

(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

4

unit

31

23

7 edit

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

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.

distribution

LHR

31

(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 ~~peaking factors~~ Core distribution) 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 ~~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~~ are within 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)