Entergy

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U. S. Nuclear Regulatory Commission
Document Control Desk
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Subject: Arkansas Nuclear One - Unit 1
Docket No. 50-313
License No. DPR-51
Arkansas Nuclear One - Unit 1 - Reply To Request For Additional Information (RAI) RE: Improved Technical Specification Section 1.0, "Use and Application," 2.0, "Safety Limits," 3.1, "Reactivity Control Systems," and 3.2, "Power Distribution Limits" (TAC No. MA8082)

Gentlemen:

By letter dated January 28, 2000 (1CAN010007), Entergy Operations submitted a license amendment request to convert the Arkansas Nuclear One - Unit 1 (ANO-1) current Technical Specifications (CTS) to an improved Technical Specification (ITS) format similar to NUREG-1430, "Standard Technical Specifications - Babcock & Wilcox Plants," Revision 1, dated April 1995. During meetings on December 18, 2001, and December 19, 2001, members of the ANO staff and the NRC Technical Specifications Branch discussed the NRC comments on ITS Sections 1.0, "Use and Application," 2.0, "Safety Limits," 3.1, "Reactivity Control Systems," and 3.2, "Power Distribution Limits," and the ANO resolutions of these comments.

This submittal contains the Entergy Operations responses to the RAIs discussed at the December 18, 2001, and December 19, 2001, meetings. The contents are arranged as follows:

Attachment 1 contains a description of the contents and format of the supplement package,

Attachments 2 and 3 delineate those comments received from the NRC Staff and ANO personnel, respectively, and the associated resolutions of those comments for Section 1.0,

Attachment 4 delineates those comments received from the NRC Staff and the associated resolutions of those comments for Section 2.0,

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Attachment 5 delineates those comments received from the NRC Staff and the associated resolutions of those comments for Section 3.1, and

Attachment 6 delineates those comments received from the NRC Staff and the associated resolutions of those comments for Section 3.2.

I declare under penalty of perjury that the foregoing is true and correct.

Executed on February 6, 2001.

Very truly yours,

CGA/cws
Attachments

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**Format of Supplement Package**

The improved Technical Specification (ITS) supplement package is organized as described below:

TAB ITS

Contains the proposed ITS Limiting Conditions for Operation (LCOs).

TAB ITS Bases

Contains the proposed ITS Bases

TAB Current Technical Specification (CTS) Markup

Contains annotated copies of the CTS pages which show the disposition of existing requirements into the proposed ITS. The pages are arranged in ITS order. The upper right hand corner of the CTS page is annotated with the ITS Specification number to which the CTS page applies. Items on the CTS page that are addressed in other proposed ITS Sections (or Specifications within the Section) are annotated with the appropriate location.

Where a proposed ITS requirement differs from a CTS requirement, individual details of the CTS revision are annotated with alpha-numeric designators which relate to the appropriate Discussion of Change (DOC). The DOC provides a concise justification for the change. The DOCs are located directly preceding the CTS Markup in each Section or sub-Section. The alpha-numeric designators also relate to the evaluations supporting a finding of No Significant Hazard Consideration (NSHC).

The CTS pages in the Section packages reflect License Amendments issued as of the date of the submittal letter, and License Amendment Requests described in Attachment 2 to the submittal letter.

The DOCs are numbered sequentially within each letter category for each ITS Section or sub-Section. The proposed changes for each CTS requirement are separated into the following categories:

| Designator | Category |
|---|---|
| A | ADMINISTRATIVE - changes to the CTS that result in no additional or reduced restrictions or flexibility. These changes are supported in aggregate by a single NSHC. |

M TECHNICAL CHANGES - MORE RESTRICTIVE - changes to the CTS that result in added restrictions or reduced flexibility. These changes are supported in aggregate by a single NSHC.

L TECHNICAL CHANGES - LESS RESTRICTIVE - changes to the CTS that result in reduced restrictions or added flexibility. Each corresponding evaluation is supported by a corresponding evaluation supporting a finding of NSHC.

LA TECHNICAL CHANGES - REMOVAL OF DETAIL - changes to the CTS that eliminate detail and relocate the detail to a licensee controlled document. Typically, this involves details of system design and function, or procedural detail on methods of conducting a surveillance. These changes are supported in aggregate by a single NSHC.

R RELOCATED SPECIFICATIONS - changes to the CTS that encompass the requirements that do not meet the selection criteria of 10 CFR 50.36(c)(2)(ii). These changes are supported in aggregate by a single NSHC.

The CTS Bases pages are replaced in their entirety. A single DOC justifies the replacement.

TAB NSHC

Contains evaluations required by 10 CFR 50.91(a) supporting a finding of No Significant Hazard Consideration (NSHC). Generic evaluations for a finding of NSHC have been written for each category of changes except Category "L." The evaluations supporting a finding of NSHC are ordered as follows: A, M, LA, R, and L. Each evaluation is annotated to correspond to the DOC discussed in the NSHC. The generic NSHC evaluations for Category A, M, and R changes are located in the Split Report section.

TAB NUREG Markup

Contains annotated copies of the applicable NUREG-1430, Revision 1, LCOs which show how the proposed ITS LCO differs from the NUREG LCO. Where a proposed ITS LCO differs from the NUREG LCO, individual details of the change are annotated with numeric designators which relate to the appropriate Discussion of Difference (DOD). The DOD provides a concise justification for the change. The LCO DODs are located directly preceding the associated markup for each Section or sub-Section.

TAB Bases Markup

Contains annotated copies of the applicable NUREG-1430, Revision 1, Bases which show how the proposed ITS Bases differ from the NUREG Bases. Where a proposed ITS Bases requirement differs from the NUREG Bases, individual details of the change are annotated with numeric designators which relate to the appropriate DOD. The DOD provides a justification for the change. The DODs are located directly preceding the associated markup of the NUREG Limiting Conditions for Operation for each Section or sub-Section.

**Existing ANO-1 License Amendment Requests (LARs) Incorporated in this supplement**

There are no new LARs incorporated in this supplement. Our letter dated January 28, 2000, showed an LAR dated July 14, 1999 as affecting CTS page 126. This CTS page appears among the CTS markup pages for Section 2.0. This LAR was approved as Amendment 208 to the ANO-1 CTS.

**Disposition of Generic Changes**

In addition to those generic changes shown as incorporated in our letter dated January 28, 2000, several additional generic changes have been incorporated in this supplement.

| Section | TSTF Number | Title | Discussion |
|---|---|---|---|
| 1.0 | ANO-1-062 | Correct Definition of APSRs to Restrict to Control Components With Part Length Absorbers Only. | 1.0DOD-11 |
| 3.1 | TSTF-220 | Revise Actions for inoperable, misaligned APSR | 3.1DOD-41 |
| 3.1 | ANO-1-063 | Change Required Action for Action A of Axial Power shaping Rod Alignment Limits (3.1.6) | 3.1DOD-41 |

**List of Beyond Scope Items**

No additional Beyond Scope Items, beyond those addressed in our January 28, 2000, submittal are contained in this supplement.

**Resolution of NRC Comments and ANO-1 Initiated Changes**

Attachment 2 provides a listing of all comments on ITS Section 1.0 received as a result of NRC review and the ANO resolutions of these comments. Attachment 3 provides a list of changes to ITS Section 1.0 as a result of the incorporation of comments received from the ANO staff. Attachment 4 provides a listing of all comments on ITS Section 2.0 received as a result of NRC review and the ANO resolutions of these comments. Attachment 5 provides a listing of all comments on ITS Section 3.1 received as a result of NRC review and the ANO resolutions of these comments. Attachment 6 provides a listing of all comments on ITS Section 3.2 received as a result of NRC review and the ANO resolutions of these comments.

In each ITS Section, each comment is assigned a unique identifying number such as 3.6.1-1, for an NRC generated comment, or ANO-71, for an ANO generated comment. This identifying number also appears in the left hand margin on each page of the submittal package that was revised as a result of the comment. Each comment response details the location of the necessary changes.

**NRC Comment Resolution**
**ITS Section 1.0: "Use and Application"**

**Comment 1. 0-01**

CTS 1.2.1, Cold Shutdown; 1.2.2, Hot Shutdown; 1.2.4, Hot Standby; 1.2.5, Power Operation;
1.2.6, Refueling Shutdown; 1.2.7, Refueling Operation; 1.2.8, Startup; and 1.3, Operable - Operability
DOCs A4, A6, A7, A8, A13, and A14

Changes to the listed CTS definitions are not considered administrative changes. Administrative changes are those that are purely editorial in nature, involve the movement or reformatting of requirements within the technical specifications without affecting the technical content. The DOCs need to be further discussed to address them as either M or L changes, as appropriate. Comment: Revise the DOC to reflect proper categorization.

**Response** Based on discussions with the reviewer, no response is required and no revision to the ITS submittal is necessary.

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**Comment 1. 0-01a**

The licensee's proposal to modify the CTS definition of OPERABILITY is not acceptable. The proposed revision would result in allowing plant operations in MODES 5 and 6 with only offsite power or diesel generators OPERABLE. This constitutes a less restrictive change which has not been justified. The licensee should provide a detailed discussion of why this change is acceptable, or retain the CTS. See also RAI 3.8.1-01

**Response** Revised 1.0DOC A8 to provide a suitable discussion based on the incorporation of S/D electrical specifications in the ANO-1 ITS.

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Note: Justification for Difference (JFD) and Discussion of Difference (DOD) are used interchangeably.

**Comment 1. 0-02**

DOCs A1, A11
The licensee is proposing to add the following definitions that are currently not in CTS to the
ITS:

| | |
|---|---|
| Core Alteration | Axial Power Shaping Rods |
| Modes | Physics Tests |
| Actions | Thermal Power |
| Leakage | Allowable Thermal Power |
| Control Rods | Shutdown Margin |

While adding these definitions to the ITS is acceptable, however, these changes are not considered administrative. Administrative changes are those that are purely editorial in nature, involve the movement or reformatting of requirements within the technical specifications without affecting the technical content. Comment: Revise the DOC to provide proper categorization for these changes.

**Response** Based on discussions with the reviewer, no response is required and no revision to the ITS submittal is necessary.

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**Comment 1. 0-03**

ITS definition of Axial Power Shaping Rods
JFD 11

The licensee is proposing to modify the STS definition by specifying that these rods are the part-length control components. This change is not acceptable without an approved traveler. Comment: Either submit a traveler for this change or leave the definition as it is in STS.

**Response**

1) A generic change, currently designated as ANO-1-062, has been submitted to the BWOG for processing. This generic change incorporates the wording for the APSR definition that had previously been approved for the Oconee Nuclear Station in their ITS conversion.
2) NUREG-1430 markup page 1.1-1 has been revised to incorporate the definition wording that was approved for Oconee, consistent with the draft generic change.
3) 1.0DOD-11 has been revised to reflect the Oconee wording and the draft generic change.
4) The proposed ITS definition of APSRs has been revised to reflect the NUREG markup.

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Note: Justification for Difference (JFD) and Discussion of Difference (DOD) are used interchangeably.

**Comment 1. 0-04**

ITS definition of Leakage
JFD 13

The licensee is proposing to delete the STS' reference to injection in the ITS' definition. This change is not acceptable without an approved traveler. Comment: Either submit a traveler for this change or leave the definition as it is in STS.

**Response**
1) Revised NUREG markup pages 1.1-4 and 1.1-5 to retain "injection" in the definitions of Identified and Unidentified LEAKAGE, consistent with the NUREG and TSTF-40.
2) Revised 1.0DOD-13 to show this DOD as "Not used."
3) Revised 1.0DOD-14 to delete reference to DOD-13.
4) Revised Proposed ITS page 1.1-3 to incorporate changes described on NUREG markup pages.

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**Comment 1. 0-05**

ITS 1.3, Completion Times, Example 1.3-6, Required Action A.2
JFD 19

The ITS replaces the STS' "Reduce Thermal Power to </= 50% RTP" with "Place the channel in bypass." This change is not acceptable without an approved traveler. Comment: Either submit a traveler for this change or leave the definition as it is in STS.

**Response**
1) Revised NUREG-1430 markup page 1.3-10 to retain the NUREG Required Action A.2 in Example 1.3-6.
2) Revised 1.0DOD-19 to show "Not used."
3) Revised proposed ITS page 1.3-10 to reflect the change in the markup.

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Note: Justification for Difference (JFD) and Discussion of Difference (DOD) are used interchangeably.

**ANO Comment Resolution**
**ITS Section 1.0: "Use and Application"**

**Comment ANO-242**

Generic Change TSTF-284, Revision 3 has been approved by the NRC. This change enhances the discussions of "met" and "performed" with respect to the performance of Surveillances. Incorporate this generic change.

**Response**
1. 1.0DOD-22 created to show incorporation of TSTF-284, Rev 3.
2. NUREG-1430 markup page 1.4-1 revised to show deletion of third paragraph of Description and the incorporation of Insert 1.4-1, in accordance with TSTF-284, Rev 3.
3. NUREG-1430 markup page 1.4-4 revised to show incorporation of Inserts 1.4-4A, 1.4-4B, and 1.4-4C, in accordance with TSTF-284, Rev 3.
4. Proposed ITS revised to incorporate changes as reflected in the markups for these pages.

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**Comment ANO-250**

CTS 3.1.6.3.a contains a description of leakage through a non-isolable fault in the reactor coolant system strength boundary and lists examples of this leakage. ITS 3.4.13 uses the terminology of pressure boundary leakage. DOC LA1 (Bases) which is identified for this change, states this information is relocated to the ITS Bases 3.4.13. ITS Bases 3.4.13 does not contain the details included CTS 3.1.6.3.a for non-isolable reactor coolant boundary leakage. This information is contained in the ITS definition for LEAKAGE. Revise Section 1.0 to include this information in the CTS and ITS markups. Refer to NRC Comment 3.4B-24.

**Response**
1. Provided a markup of CTS page 27 showing a portion of CTS 3.1.6.3.a as applicable to the ITS definition of Pressure Boundary LEAKAGE.
2. Revised NUREG-1430 markup page 1.1-5 to show a reference to CTS 3.1.6.3 as applicable to the definition of Pressure Boundary LEAKAGE.

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**NRC Comment Resolution**
**ITS Section 2.0: "Safety Limits"**

**Comment** **2. 0-01**

CTS 6.6 and 6.7, page 126
DOC A4

The CTS markup submittal, CTS page 126, is dependent upon NRC approval of ANO-1 LAR of July 14, 1999 (Ref. 0CAN079901). Comment: Has the NRC approved the ANO-1 LAR of July 14, 1999?

**Response** Amendment 208 was approved by the NRC on August 17, 2000.

1) Revised 2.0DOC A4 to show "Not used."
2) Revised CTS markup page 126 to show page as approved following incorporation of Amendment 208.

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**Comment** **2. 0-02**

ITS 2.1.1.3, RCS Variable Pressure-Temperature Limits
CTS 2.1.3, RCS Variable Pressure-Temperature Limits
DOD-19

The CTS and ITS reference RCS Variable Pressure-Temperature Limits that are located in the Core Operating Limits Report (COLR). Comment: Safety Limits must appear in the Technical Specifications. To ensure that there is no confusion over the limits that are located in the COLR, it is recommended that the following (or similar) phrase be added to the 2.1.1.3 sentence: ", so that the safety limits are met."

**Response**

1) NUREG-1430 markup page 2.0-1 has been revised to add the phrase ", so that the safety limits are met" to the end of ITS 2.1.1.3.
2) 2.0DOD-19 has been revised to discuss this editorial change.
3) Proposed ITS page 2.0-1 has been revised to incorporate the markup change.

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**Comment 2. 0-03**

ITS Bases B2.1.2, Applicable Safety Analysis
DOD-18

Reference is made to "The startup event ...". Comment: The word "analysis" appears to be missing after the words, "The startup event."

**Response**
1) NUREG-1430 markup page B 2.0-7 has been revised to include "analysis" in the text revised by 2.0DOD-18.
2) Proposed ITS page B 2.1.2-1 has been revised to reflect the change to the markup.

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**NRC Comment Resolution**
**ITS Section 3.1: "Reactivity Control Systems"**

**Comment 3. 1-01**

ITS Bases, TRM, SAR, COLR
CTS 3.1.7.2, 3.1.9.1, 3.1.9.2, 3.1.9.3, Figure 3.1.9-1, Table 4.1-3 Item 1.d, Table 4.1-3 Note 7, 4.7.1.1, 4.7.1.2, and 4.7.1.3
DOC LA1

Some details not necessary to convey a regulatory requirement are moved to licensee controlled documents; the ITS Bases, the Technical Requirements Manual (TRM), the Core Operating Limits Report (COLR), and the Safety Analysis Report (SAR). DOC LA1 states that, "Changes to the SAR, TRM and COLR will be controlled by 10 CFR 50.59." Comment: Details that are moved to licensee controlled documents must have approved change control processes to ensure that safety margins are not significantly reduced. The ITS Bases are controlled by the ITS Bases Control Program in Section 5 of the ITS. The SAR, which includes the TRM either by complete inclusion or by reference, is controlled by the 10 CFR 50.59 process. The COLR is not always incorporated into the SAR and controlled by 10 CFR 50.59; it is frequently controlled by requirements in Section 5 of the ITS. Is the ANO-1 COLR included in the SAR and to be controlled by 10 CFR 50.59?

**Response** 3.1DOC-LA1 has been revised to state that the TRM and Core Operating Limits Report are considered to be part of the SAR.

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**Comment 3. 1-02**

TRM
CTS 3.1.9.3 Dissolved Gases Concentration
DOC L14 and DOC LA1

The CTS dissolved gases concentration requirements are moved to the TRM. The CTS required actions, if the gas concentrations are not restored to within limits within 24 hours, are to be in Hot Shutdown within the next 6 hours and to be in Cold Shutdown within the next 30 hours. These CTS required actions are not moved to the TRM because a shutdown is required to perform the remaining required action of checking the vessel level instrument vent for the accumulation of undissolved gases. Comment: By deleting the shutdown required actions results in there being no time frame in which to perform the vent check. What does ANO-1 perceive the appropriate actions and time frames to be? What process did ANO-1 use to determine that these actions were to be deleted?

**Response**

1) 3.1DOC-L14 has been revised to provide additional details concerning the proposed TRM Required Actions and Completions Times in the event dissolved gas concentration is not within limits.
2) CTS markup page 32 has been revised to show that the action requiring restoration of parameters to within limits within 24 hours has been relocated to the TRM as discussed in 3.1DOC-L14 and 3.1DOC-LA1.

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**Comment 3. 1-03**

ITS 3.1.1 SDM LCO Statement
DOD-1

In adopting TSTF-9, ANO-1 changed the LCO wording to the "limit provided in the COLR," rather than "limit specified in the COLR." Comment: Unfortunately, TSTF-9 is inconsistent in the wording used, however, the TSTF OG consensus is that using "specified" is the preferred way wording the LCO statement. Recommend changing "provided" to "specified," since the TS do specify limits and not merely provide them.

**Response**

1) NUREG-1430 markup page 3.1-1 has been revised to refer to the "limit specified in the Core Operating Limits Report" in the LCO statement.
2) Proposed ITS page 3.1.1-1 has been revised to reflect this change in the markup.

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**Comment 3. 1-04**

Bases B3.1.1 Required Action A.1
DOD-30

The ITS Bases section addressing Required Action A.1 has the paragraph on boration flow rate deleted. The justification for this deletion is that the example is very much based upon plant conditions and time in core life. Comment: The deleted paragraph is merely an example, it is stated as such, and it is intended to remind the operator what is involved in the calculation; it is an aid. It is worth noting that the old B&W Standard TS had the boration flow specified in an Action statement. Recommend retaining the example in the Bases.

**Response** Borations are performed in accordance with approved implementing procedures. Having this example in the Bases is not considered to be a clarification of the specification for the operators. All necessary guidance is provided in the implementing procedures.

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**Comment 3. 1-05**

ITS 3.1.4 Control Rod Alignment Limits
Bases B3.1.4 LCO and Background sections
CTS 3.5.2 and CTS 4.7.1
DOD-5 and DOC A11

DOC A11 appears to state that the ITS definition of control rod operability will not include the CTS alignment aspect of operability as defined in CTS 4.7.1.2. The ITS Bases should include a definition of system operability. In retaining the CTS definition of control operability the STS definition of control rod operability has been removed. Comment: The CTS definition has not been included in the ITS 3.1.4 Bases. Recommend including a definition of control rod operability in the ITS Bases, as defined in CTS 4.7.1.1, 4.7.1.2, and 4.7.1.3. Fully discuss the reasons for any differences in the DOCs.

**Response**

1) Revised NUREG-1430 markup page B 3.1-20 to provide a discussion of control rod operability in the LCO Bases.
2) Revised 3.1DOD-05 to address the change to the LCO Bases and provide a discussion of Control Rod Operability.
3) Revised proposed ITS page B 3.1.4-3 in accordance with the markup change.

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**Comment 3. 1-06**

ITS Bases B3.1.4, Control Rod Alignment Limits
Bases SR 3.1.4.2
DOD-5

The ITS retains the CTS definition of control rod operability, which includes movability as an element of operability. In the last two sentences of the Bases for SR 3.1.4.2, the ITS permits a control rod to be considered operable even after it is found to be immovable and not necessarily even trippable. Comment: The ITS is adopting an aspect of the STS version of control rod operability, that does not apply to the CTS definition. The last two sentences of the Bases for SR 3.1.4.2 do not apply to the CTS/ITS and should be deleted.

**Response** Based on discussions with the reviewer, no response is required and no revision to the ITS submittal is necessary.

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**Comment 3. 1-07**

ITS 3.1.4 Control Rod Alignment Limits
CTS 4.7.1.2
DOC M18

The CTS requirement to first evaluate the control rod with the greatest deviation from the group average position is not included in the ITS. Comment: It is not clear why removing this requirement is more restrictive and acceptable.

**Response**
1) Revised 3.1DOC-M18 to show that it is "Not used."
2) Developed 3.1DOC-A13 to discuss the deletion of this allowance as administrative in nature due to the requirements of CTS 3.5.2.2.1 which would prohibit operation with more than one control rod misaligned by more than nine inches from the group average.
3) Revised CTS markup page 102 to show that the change to CTS 4.7.1.2 is addressed by 3.1DOC-A13, not 3.1DOC-M18.

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**Comment 3. 1-08**

ITS 3.1.4 Control Rod Alignment Limits
ITS Required Action A.2.1, Completion Time
CTS 3.5.5.2.5
DOD-11

The ITS changes the STS Completion Time for restoring control rod alignment from 1 to 2 hours. The corresponding CTS Required Action does not have a Completion Time, and in reorganizing the STS Required Actions it appears logical to allow 2 hours since it is grouped with an alternative action to reduce power that has a Completion Time of 2 hours. Comment: By extending the STS Completion by 1 hour, the ITS is also relaxing when it is necessary to perform the subsequent Required Actions of verifying the safety basis is met and that LHR is within limits. Recommend retaining the 1 hour Completion Time for restoring control rod alignment.

**Response** Based on discussions with the reviewer, no response is required and no revision to the ITS submittal is necessary.

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**Comment 3. 1-09**

ITS 3.1.4 Control Rod Alignment Limits
ITS SR 3.1.4.2
DOD-8

ITS SR 3.1.4.2 deletes the STS words that explicitly state that control rod freedom of movement should be verified "by moving" the control rods. The Bases do state that the control rods should be moved "enough to verify freedom of movement." Comment: The obvious question is "how much is enough?" Are there any circumstances where an alternative means of verification would be considered acceptable and no movement would be "enough?"

**Response**

1) Revised NUREG-1430 markup page B 3.1-26 to incorporate appropriate acceptance criteria of approximately 1.5% (approximately 2 inches) in the SR 3.1.4.2 Bases, and added a statement that this parameter does not require additional allowances for instrument uncertainty to be incorporated in the implementing procedures.
2) Revised 3.1DOD-08 to discuss the addition of this acceptance criteria.
3) Revised 3.1DOD-43 to discuss the addition of the information concerning instrument uncertainties in the SR 3.1.4.2 Bases.
4) Revised proposed ITS page B 3.1.4-7 to incorporate changes on markup.

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**Comment 3. 1-10**

ITS 3.1.6 Axial Power Shaping (APSR) Alignment Limits
ITS 3.1.6 Required Actions
DOD-7 and DOD-41

The ITS revises the STS 3.1.6 Required Actions. The ITS does not include the STS Required Action B to be in Mode 3 if the Required Actions and associated Completion Times for Condition A are not met. Also, the ITS does not include the requirement, in the most recent version of STS 3.1.6, to perform SR 3.2.3.1 to verify Axial Power Balance is within limits.

Comment: The STS Condition B and associated Required Actions preclude entry into LCO 3.0.3 when Required Actions and associated Completion Times for Condition A are not met. The STS Required Action to perform SR 3.2.3.1 ensures that core power distribution is within limits. Recommend adopting these STS Required Actions.

**Response**

1) Revised NUREG-1430 markup page 3.1-12 to incorporate TSTF-220 and ANO-1-063 in the Condition A Required Action and to retain the NUREG Condition B and associated Required Actions and Completion Times.
2) Revised NUREG-1430 markup pages B 3.1-35 and B 3.1-36 to incorporate TSTF-220 and ANO-1-063 and page B 3.1-36 to retain the discussion of Action B.1.
3) Proposed ITS page 3.1.6-1 revised to reflect incorporation of changes in the markup.
4) Proposed ITS page B 3.1.6-3 revised to reflect incorporation of changes in the Bases markup.
5) CTS markup page 47 revised to show incorporation of NUREG-1430 LCO 3.1.6 Required Action A.1 and Condition B.
6) Drafted 3.1DOC-M26 to discuss incorporation of NUREG-1430 LCO 3.1.6 Condition B.
7) Drafted 3.1DOC-M27 to discuss incorporation of NUREG 3.1.6 Required Action A.1.
8) Revised 3.1DOD-41 to discuss incorporation of TSTF-220 and proposed generic change ANO-1-063.

**Comment 3. 1-11**

ITS 3.1.8 and ITS 3.1.9 Physics Test Exceptions
STS 3.1.8 and STS 3.1.9 Physics Test Exceptions
ITS 3.1.8 and ITS 3.1.9 LCO Statements
DOD-15 and DOC L10

The ITS 3.1.8 and ITS 3.1.9 Physics Test Exceptions LCO Statements include test exceptions for ITS 3.2.2 on APSR Insertion Limits. The STS 3.1.8 and STS 3.1.9 Physics Test Exceptions do not include these test exceptions. Comment: It is not apparent that the test exceptions for ITS 3.2.2 on APSR Insertion Limits is part of the current licensing basis as described. The NRC staff is reviewing this change.

**Response** Based on discussions with the reviewer, no response is required and no revision to the ITS submittal is necessary.

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**Comment 3. 1-12**

ITS 3.1.8 and ITS 3.1.9 Physics Test Exceptions
STS SR 3.1.8.3 and STS SR 3.1.9.2
ITS SR 3.1.8.2 and ITS 3.1.9.2
DOD-2

The ITS changes the STS SR 3.1.8.3 and STS SR 3.1.9.2 frequencies from "8 hours" to "Prior to performance of Physics Test." This is based upon TSTF-344. Comment: TSTF-344 has not been approved, and is not expected to be approved. Recommend adopting the STS SR frequencies.

**Response**

1) Revised NUREG-1430 markup page 3.1-19 to revise SR 3.1.8.3 Frequency to read "Within 8 hours prior to performance of PHYSICS TESTS at each testing plateau."
2) Revised NUREG-1430 markup page 3.1-22 to revise SR 3.1.9.2 Frequency to read "Within 8 hours prior to performance of PHYSICS TESTS."
3) Revised proposed ITS page 3.1.8-2 to reflect change in SR 3.1.8.3 Frequency.
4) Revised proposed ITS page 3.1.9-2 to reflect change in SR 3.1.9.2 Frequency.
5) Revised NUREG-1430 Bases markup page B 3.1-50 to reflect the change in SR 3.1.8.3 Frequency.
6) Revised NUREG-1430 Bases markup page B 3.1-57 to reflect the change in SR 3.1.9.2 Frequency.
7) Revised proposed ITS Bases page B 3.1.8-5 to reflect change in SR 3.1.8.3 Bases markup.
8) Revised proposed ITS Bases page B 3.1.9-5 to reflect change in SR 3.1.9.2 Bases markup.
9) Revised 3.1DOD-02 to discuss the change in SR 3.1.8.3 and SR 3.1.9.2 Frequencies.

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**Comment 3. 1-13**

ITS 3.1.9 Physics Test Exception
ITS 3.1.9 and STS 3.1.9 LCO Statements
DOD-42

The ITS does not adopt the STS LCO Statement regarding suspending LCO 3.2.1, on "Regulating Rod Insertion Limits," that limits suspension to the "restricted operation region only." Comment: It is not apparent that the test exception limitation for LCO 3.2.1 on Regulating Rod Insertion Limits is part of the current licensing basis as described. The NRC staff is reviewing this change.

**Response** Based on discussions with the reviewer, no response is required and no revision to the ITS submittal is necessary.

## NRC Comment Resolution
## ITS Section 3.2: "Power Distribution Limits"

**Comment 3. 2-01**

ITS 3.2.1 Regulating Rod Insertion Limits
STS 3.2.1 Regulating Rod Insertion Limits
CTS 3.5.2.5.3
DOD-1(2) and DOC L6

The CTS provides one Required Action on exceeding control rod position setpoints that addresses both insertion limits, and sequence and overlap requirements, with the completion times of initiating corrective action immediately and restoring to within limits of 4 hours. The STS, with TSTF-345 incorporated, provides 24 hours to restore regulating rods to within insertion limits, and 2 hours to restore sequence and overlap limits. The ITS proposes to adopt 24 hours to restore regulating rods to within insertion limits and 4 hours to restore sequence and overlap limits. Comment: The ITS ignores the CTS explicit requirement to initiate action immediately while adopting the least conservative completion times of the CTS and STS. Unless there is a plant specific rationale for the 4 hour completion time to restore sequence and overlap limits, recommend adopting the STS completion times (STS with TSTF-345 incorporated).

**Response** Based on discussions with the reviewer, no response is required and no revision to the ITS submittal is necessary.

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**Comment 3. 2-02**

ITS 3.2.1 Regulating Rod Insertion Limits
STS 3.2.1 Regulating Rod Insertion Limits
STS SR 3.2.1.3
DOD-4

The STS includes a surveillance to verify SDM 4 hours prior to criticality. The STS Bases states that this is to ensure "that sufficient SDM capability exists with the control rods at the estimated critical position if necessary to shutdown or trip the reactor after criticality." The justification for not adopting this SR is that it is redundant to the 24 hour SR 3.1.1.1 and the CTS does not contain this requirement. Comment: As indicated in the STS Bases the SDM surveillances are not redundant in that they are determined for different plant conditions. Recommend adopting the STS SR 3.2.1.3.

**Response**

1 CTS markup page 46 revised to show the inclusion of NUREG-1430 SR 3.2.1.3.
2 Drafted 3.2DOC-M19 to discuss incorporation of NUREG-1430 SR 3.2.1.3.
3 Revised NUREG-1430 markup page 3.2-3 to show the incorporation of SR 3.2.1.3.
4 Revised NUREG-1430 markup page B 3.2-9 to show incorporation of SR 3.2.1.3.
5 Revised proposed ITS page 3.2.1-2 to show the incorporation of SR 3.2.1.3.
6 Revised proposed ITS page B 3.2.1-7 to show the incorporation of SR 3.2.1.3.
7 Revised 3.2DOD-04 to indicate that it is "Not used."

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**Comment** **3. 2-03**

ITS Bases 3.2.2 APSR Insertion Limits
ITS Bases 3.2.2 Background
STS Bases 3.2.2 Background
DOD-5

The ITS does not include the STS phrase, "are not required for reactivity insertion rate on trip or SDM and therefore, they" do not insert upon a reactor trip. The justification given is that it is potentially misleading because the "... APSRs are not designed to insert ... they were never credited in the analyses ...". Comment: The deleted phrase and DOD-5 appear to be consistent; it is not clear how the STS Bases is potentially misleading.

**Response** No changes have been made as a result of this comment. This is purely an editorial preference. The NUREG-1430 Bases state that "The APSRs are not required for reactivity insertion rate on trip or SDM and therefore, they do not trip upon a reactor trip." In actuality, the APSR design is such that they are incapable of tripping and are therefore not considered for reactivity rate insertion on a trip or SDM.

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**Comment** **3. 2-04**

ITS Bases 3.2.3 Axial Power Imbalance Operating Limits
ITS Bases 3.2.4 QPT
CTS 3.5.4 Incore Instrumentation
DOC LA1

The CTS detail, "... at least 23 individual incore detectors shall be operable to ...", is identified as being relocated to the ITS Bases. Comment: Where in the Bases is this specific information?

**Response**
1 Revised CTS markup page 51 to show the CTS 3.5.4 Specification for 23 detectors as relocated to the Technical Requirements Manual.
2 Revised 3.2DOC-LA1 to discuss this relocation.

---

**Comment 3. 2-05**

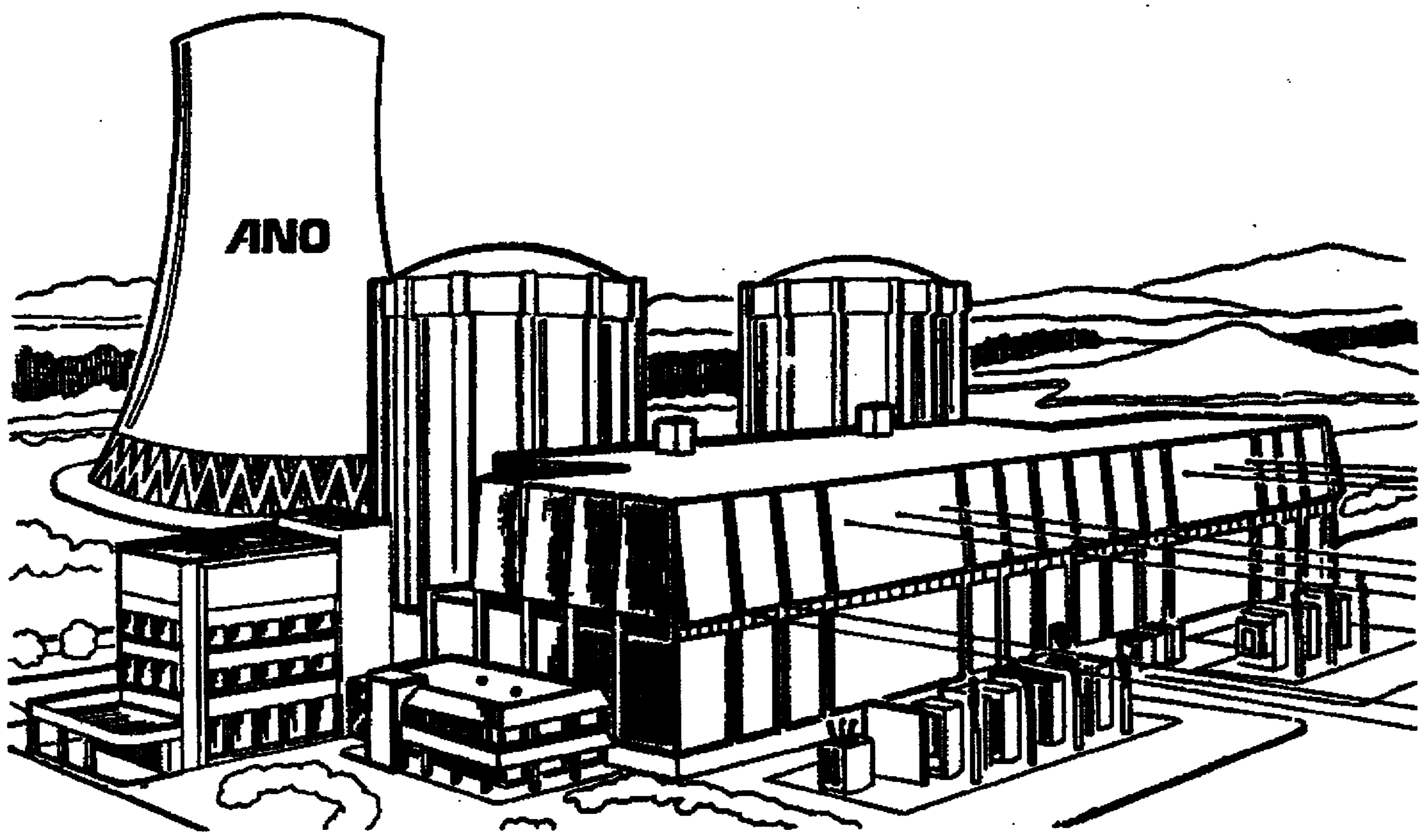
ITS 3.2.4 QPT
STS 3.2.4 QPT
DOD-17

Conditional Completion Times of "10 hours after last performance of SR 3.2.5.1" are added in the ITS to Required Actions that have a Completion Time of 10 hours. The justification for these additions is that SR 3.2.5.1 may be performed over an extended period of time. Comment: Required Action A.1.1 is to perform SR 3.2.5.1 once per two hours. How much longer than 2 hours can it take to perform SR 3.5.2.1 and why? Are the added conditional Completion Times necessary?

**Response** The change to the NUREG-1430 3.2.4 Required Actions was reviewed by the NRC Technical Staff as a beyond scope issue (TAC # MA8660) and found to be acceptable.

# ARKANSAS NUCLEAR ONE - UNIT 1

# IMPROVED TECHNICAL SPECIFICATIONS SUBMITTAL




## 02/06/01 Supplement

(Sections 1.0, 2.0, 3.1, and 3.2)

Entergy

February 6, 2001

# 1.0 USE AND APPLICATION

## 1.1 Definitions

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-----NOTE-----

The defined terms of this section appear in capitalized type and are applicable throughout these Technical Specifications and Bases.

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| Term | Definition |
|---|---|
| ACTIONS | ACTIONS shall be that part of a Specification that prescribes Required Actions to be taken under designated Conditions within specified Completion Times. |
| ALLOWABLE THERMAL POWER | ALLOWABLE THERMAL POWER shall be the maximum steady state reactor core heat transfer rate to the reactor coolant permitted by consideration of the number and configuration of reactor coolant pumps (RCPs) in operation. |
| AXIAL POWER IMBALANCE | AXIAL POWER IMBALANCE shall be the power in the top half of the core, expressed as a percentage of RATED THERMAL POWER (RTP), minus the power in the bottom half of the core, expressed as a percentage of RTP. |
| AXIAL POWER SHAPING RODS (APSRs) | APSRs shall be the control components with part length absorbers used to control the axial power distribution of the reactor core. The APSRs are positioned manually by the operator and are not trippable. |
| CHANNEL CALIBRATION | A CHANNEL CALIBRATION shall be the adjustment, as necessary, of the channel output such that it responds within the necessary range and accuracy to known values of the parameter that the channel monitors. The CHANNEL CALIBRATION shall encompass all devices in the channel required for channel OPERABILITY and the CHANNEL FUNCTIONAL TEST. Calibration of instrument channels with resistance temperature detector (RTD) or thermocouple sensors may consist of an inplace qualitative assessment of sensor behavior and normal calibration of the remaining adjustable devices in the channel. |

1.1 Definition

| | |
|---|---|
| CHANNEL CALIBRATION (continued) | The CHANNEL CALIBRATION may be performed by means of any series of sequential, overlapping, or total channel steps. |
| CHANNEL CHECK | A CHANNEL CHECK shall be the qualitative assessment, by observation, of channel behavior during operation. This determination shall include, where possible, comparison of the channel indication and status to other indications or status derived from independent instrument channels measuring the same parameter. |
| CHANNEL FUNCTIONAL TEST | A CHANNEL FUNCTIONAL TEST shall be the injection of a simulated or actual signal into the channel as close to the sensor as practicable to verify OPERABILITY of all devices in the channel required for channel OPERABILITY. The CHANNEL FUNCTIONAL TEST may be performed by means of any series of sequential, overlapping, or total steps. |
| CONTROL RODS | CONTROL RODS shall be all full length safety and regulating rods that are used to shutdown the reactor and control power level during maneuvering operations. |
| CORE ALTERATION | CORE ALTERATION shall be the movement of any fuel, sources, or reactivity control components, within the reactor vessel with the vessel head removed and fuel in the vessel. Suspension of CORE ALTERATIONS shall not preclude completion of movement of a component to a safe position. |
| CORE OPERATING LIMITS REPORT (COLR) | The COLR is the ANO-1 specific document that provides cycle specific parameter limits for the current reload cycle. These cycle specific parameter limits shall be determined for each reload cycle in accordance with Specification 5.6.5. Plant operation within these limits is addressed in individual Specifications. |
| DOSE EQUIVALENT I-131 | DOSE EQUIVALENT I-131 shall be that concentration of I-131 (microcuries/gram) that alone would produce the same thyroid dose as the quantity and isotopic mixture of I-131, I-132, I-133, I-134, and I-135 actually present. The thyroid dose conversion factors used for this calculation shall be those listed in Table III of TID-14844, AEC, 1962, "Calculation of Distance Factors for Power and Test Reactor Sites." |

1.1 Definition (continued)

$\bar{E}$-AVERAGE DISINTEGRATION ENERGY

$\bar{E}$ shall be the average (weighted in proportion to the concentration of each radionuclide in the reactor coolant at the time of sampling) of the sum of the average beta and gamma energies per disintegration (in MeV) for isotopes, other than iodines, with half lives > 15 minutes, making up at least 95% of the total noniodine activity in the coolant.

LEAKAGE

LEAKAGE shall be:

a. Identified LEAKAGE

1. LEAKAGE, such as that from pump seals or valve packing (except RCP seal water injection or leakoff), that is captured and conducted to collection systems or a sump or collecting tank;

2. LEAKAGE into the containment atmosphere from sources that are both specifically located and known either not to interfere with the operation of leakage detection systems or not to be pressure boundary LEAKAGE; or

3. Reactor Coolant System (RCS) LEAKAGE through a steam generator (SG) to the Secondary System;

b. Unidentified LEAKAGE

All LEAKAGE (except RCP seal water injection and leakoff) that is not identified LEAKAGE;

c. Pressure Boundary LEAKAGE

LEAKAGE (except SG LEAKAGE) through a nonisolable fault in an RCS component body, pipe wall, or vessel wall.

MODE

A MODE shall correspond to any one inclusive combination of core reactivity condition, power level, average reactor coolant temperature, and reactor vessel head closure bolt tensioning specified in Table 1.1-1 with fuel in the reactor vessel.

1.1 Definition (continued)

| | |
|---|---|
| OPERABLE-OPERABILITY | A system, subsystem, train, component, or device shall be OPERABLE or have OPERABILITY when it is capable of performing its specified safety function(s) and when all necessary attendant instrumentation, controls, normal or emergency electrical power, cooling and seal water, lubrication, and other auxiliary equipment that are required for the system, subsystem, train, component, or device to perform its specified safety function(s) are also capable of performing their related support function(s). |
| PHYSICS TESTS | PHYSICS TESTS shall be those tests performed to measure the fundamental nuclear characteristics of the reactor core and related instrumentation.<br><br>These tests are:<br><br>a. Described in the SAR;<br><br>b. Authorized under the provisions of 10 CFR 50.59; or<br><br>c. Otherwise approved by the Nuclear Regulatory Commission. |
| QUADRANT POWER TILT (QPT) | QPT shall be defined by the following equation and is expressed as a percentage.<br><br>$QPT = 100\left(\frac{\text{Power in any Core Quadrant}}{\text{Average Power in all Quadrants}} - 1\right)$ |
| RATED THERMAL POWER (RTP) | RTP shall be a total steady state reactor core heat transfer rate to the reactor coolant of 2568 MWt. |

1.1 Definition (continued)

| | |
|---|---|
| SHUTDOWN MARGIN (SDM) | SDM shall be the instantaneous amount of reactivity by which the reactor is subcritical or would be subcritical from its present condition assuming:<br><br>a. All full length CONTROL RODS (safety and regulating) are fully inserted except for the single CONTROL ROD of highest reactivity worth, which is assumed to be fully withdrawn. With any CONTROL ROD not capable of being fully inserted, the reactivity worth of these CONTROL RODS must be accounted for in the determination of SDM;<br><br>b. In MODES 1 and 2, the fuel and moderator temperatures are changed to the nominal zero power design level; and<br><br>c. There is no change in APSR position. |
| STAGGERED TEST BASIS | A STAGGERED TEST BASIS shall consist of the testing of one of the systems, subsystems, channels, or other designated components during the interval specified by the Surveillance Frequency, so that all systems, subsystems, channels, or other designated components are tested during *n* Surveillance Frequency intervals, where *n* is the total number of systems, subsystems, channels, or other designated components in the associated function. |
| THERMAL POWER | THERMAL POWER shall be the total reactor core heat transfer rate to the reactor coolant. |

Table 1.1-1 (page 1 of 1)
MODES

| MODE | TITLE | REACTIVITY CONDITION ($k_{eff}$) | % RATED THERMAL POWER [a] | AVERAGE REACTOR COOLANT TEMPERATURE (°F) |
|---|---|---|---|---|
| 1 | Power Operation | $\geq 0.99$ | $> 5$ | NA |
| 2 | Startup | $\geq 0.99$ | $\leq 5$ | NA |
| 3 | Hot Standby | $< 0.99$ | NA | $\geq 280$ |
| 4 | Hot Shutdown [b] | $< 0.99$ | NA | $280 > T_{avg} > 200$ |
| 5 | Cold Shutdown [b] | $< 0.99$ | NA | $\leq 200$ |
| 6 | Refueling [c] | NA | NA | NA |

(a) Excluding decay heat.

(b) All reactor vessel head closure bolts fully tensioned.

(c) One or more reactor vessel head closure bolts less than fully tensioned.

# 1.0 USE AND APPLICATION

## 1.2 Logical Connectors

PURPOSE

The purpose of this section is to explain the meaning of logical connectors.

Logical connectors are used in Technical Specifications (TS) to discriminate between, and yet connect, discrete Conditions, Required Actions, Completion Times, Surveillances, and Frequencies. The only logical connectors that appear in TS are <u>AND</u> and <u>OR</u>. The physical arrangement of these connectors constitutes logical conventions with specific meanings.

BACKGROUND

Several levels of logic may be used to state Required Actions. These levels are identified by the placement (or nesting) of the logical connectors and by the number assigned to each Required Action. The first level of logic is identified by the first digit of the number assigned to a Required Action and the placement of the logical connector in the first level of nesting (i.e., left justified with the number of the Required Action). The successive levels of logic are identified by additional digits of the Required Action number and by successive indentations of the logical connectors.

When logical connectors are used to state a Condition, Completion Time, Surveillance, or Frequency, only the first level of logic is used, and the logical connector is left justified with the statement of the Condition, Completion Time, Surveillance, or Frequency.

EXAMPLES

The following examples illustrate the use of logical connectors.

1.2 Logical Connectors

EXAMPLES (continued)

EXAMPLE 1.2-1

ACTIONS

| CONDITION | REQUIRED ACTION | COMPLETION TIME |
|---|---|---|
| A. LCO not met. | A.1 Verify . . .<br><br>AND<br><br>A.2 Restore . . . | |

In this example the logical connector AND is used to indicate that when in Condition A, both Required Actions A.1 and A.2 must be completed.

1.2 Logical Connectors

EXAMPLES (continued)

EXAMPLE 1.2-2

ACTIONS

| CONDITION | REQUIRED ACTION | COMPLETION TIME |
|---|---|---|
| A. LCO not met. | A.1 Trip . . .<br><br>OR<br><br>A.2.1 Verify . . .<br><br>AND<br><br>A.2.2.1 Reduce . . .<br><br>OR<br><br>A.2.2.2 Perform . . .<br><br>OR<br><br>A.3 Align . . . | |

This example represents a more complicated use of logical connectors. Required Actions A.1, A.2, and A.3 are alternative choices, only one of which must be performed as indicated by the use of the logical connector OR and the left justified placement. Any one of these three Actions may be chosen. If A.2 is chosen, then both A.2.1 and A.2.2 must be performed as indicated by the logical connector AND. Required Action A.2.2 is met by performing A.2.2.1 or A.2.2.2. The indented position of the logical connector OR indicates that A.2.2.1 and A.2.2.2 are alternative choices, only one of which must be performed.

# 1.0 USE AND APPLICATION

## 1.3 Completion Times

| | |
|---|---|
| PURPOSE | The purpose of this section is to establish the Completion Time convention and to provide guidance for its use. |
| BACKGROUND | Limiting Conditions for Operation (LCOs) specify minimum requirements for ensuring safe operation of the unit. The ACTIONS associated with an LCO state Conditions that typically describe the ways in which the requirements of the LCO can fail to be met. Specified with each stated Condition are Required Action(s) and Completion Time(s). |
| DESCRIPTION | The Completion Time is the amount of time allowed for completing a Required Action. It is referenced to the time of discovery of a situation (e.g., inoperable equipment or variable not within limits) that requires entering an ACTIONS Condition unless otherwise specified, providing the unit is in a MODE or specified condition stated in the Applicability of the LCO. Required Actions must be completed prior to the expiration of the specified Completion Time. An ACTIONS Condition remains in effect and the Required Actions apply until the Condition no longer exists or the unit is not within the LCO Applicability.<br><br>If situations are discovered that require entry into more than one Condition at a time within a single LCO (multiple Conditions), the Required Actions for each Condition must be performed within the associated Completion Time. When in multiple Conditions, separate Completion Times are tracked for each Condition starting from the time of discovery of the situation that required entry into the Condition.<br><br>Once a Condition has been entered, subsequent trains, subsystems, components, or variables expressed in the Condition, discovered to be inoperable or not within limits, will <u>not</u> result in separate entry into the Condition, unless specifically stated. The Required Actions of the Condition continue to apply to each additional failure, with Completion Times based on initial entry into the Condition. |

1.3 Completion Times

DESCRIPTION (continued)

DESCRIPTION (continued)

However, when a subsequent train, subsystem, component, or variable, expressed in the Condition, is discovered to be inoperable or not within limits, the Completion Time(s) may be extended. To apply this Completion Time extension, two criteria must first be met. The subsequent inoperability:

a. Must exist concurrent with the first inoperability; and

b. Must remain inoperable or not within limits after the first inoperability is resolved.

The total Completion Time allowed for completing a Required Action to address the subsequent inoperability shall be limited to the more restrictive of either:

a. The stated Completion Time, as measured from the initial entry into the Condition, plus an additional 24 hours; or

b. The stated Completion Time as measured from discovery of the subsequent inoperability.

The above Completion Time extensions do not apply to those Specifications that have exceptions that allow completely separate re-entry into the Condition (for each train, subsystem, component, or variable expressed in the Condition) and separate tracking of Completion Times based on this re-entry. These exceptions are stated in individual Specifications.

The above Completion Time extension does not apply to a Completion Time with a modified "time zero." This modified "time zero" may be expressed as a repetitive time (i.e., "once per 8 hours," where the Completion Time is referenced from a previous completion of the Required Action versus the time of Condition entry) or as a time modified by the phrase "from discovery . . ." Example 1.3-3 illustrates one use of this type of Completion Time. The 10 day Completion Time specified for Conditions A and B in Example 1.3-3 may not be extended.

EXAMPLES

The following examples illustrate the use of Completion Times with different types of Conditions and changing Conditions.

1.3 Completion Times (continued)

EXAMPLES (continued)

EXAMPLE 1.3-1

ACTIONS

| CONDITION | REQUIRED ACTION | COMPLETION TIME |
|---|---|---|
| B. Required Action and associated Completion Time not met. | B.1 Be in MODE 3.<br><br>AND<br><br>B.2 Be in MODE 5. | 6 hours<br><br><br><br>36 hours |

Condition B has two Required Actions. Each Required Action has its own separate Completion Time. Each Completion Time is referenced to the time that Condition B is entered.

The Required Actions of Condition B are to be in MODE 3 within 6 hours AND in MODE 5 within 36 hours. A total of 6 hours is allowed for reaching MODE 3 and a total of 36 hours (not 42 hours) is allowed for reaching MODE 5 from the time that Condition B was entered. If MODE 3 is reached within 3 hours, the time allowed for reaching MODE 5 is the next 33 hours because the total time allowed for reaching MODE 5 is 36 hours.

If Condition B is entered while in MODE 3, the time allowed for reaching MODE 5 is the next 36 hours.

## 1.3 Completion Times

EXAMPLES (continued)

EXAMPLE 1.3-2

ACTIONS

| CONDITION | REQUIRED ACTION | COMPLETION TIME |
|---|---|---|
| A. One pump inoperable. | A.1 Restore pump to OPERABLE status. | 7 days |
| B. Required Action and associated Completion Time not met. | B.1 Be in MODE 3.<br>AND<br>B.2 Be in MODE 5. | 6 hours<br><br>36 hours |

When a pump is declared inoperable, Condition A is entered. If the pump is not restored to OPERABLE status within 7 days, Condition B is also entered and the Completion Time clocks for Required Actions B.1 and B.2 start. If the inoperable pump is restored to OPERABLE status after Condition B is entered, Condition A and B are exited, and therefore, the Required Actions of Condition B may be terminated.

When a second pump is declared inoperable while the first pump is still inoperable, Condition A is not re-entered for the second pump. LCO 3.0.3 is entered, since the ACTIONS do not include a Condition for more than one inoperable pump. The Completion Time clock for Condition A does not stop after LCO 3.0.3 is entered, but continues to be tracked from the time Condition A was initially entered.

While in LCO 3.0.3, if one of the inoperable pumps is restored to OPERABLE status and the Completion Time for Condition A has not expired, LCO 3.0.3 may be exited and operation continued in accordance with Condition A.

While in LCO 3.0.3, if one of the inoperable pumps is restored to OPERABLE status and the Completion Time for Condition A has expired, LCO 3.0.3 may be exited and operation continued in accordance with Condition B. The Completion Time for Condition B is tracked from the time the Condition A Completion Time expired.

1.3 Completion Times

EXAMPLES (continued)

On restoring one of the pumps to OPERABLE status, the Condition A Completion Time is not reset, but continues from the time the first pump was declared inoperable. This Completion Time may be extended if the pump restored to OPERABLE status was the first inoperable pump. A 24 hour extension to the stated 7 days is allowed, provided this does not result in the second pump being inoperable for > 7 days.

1.3 Completion Times

EXAMPLES (continued)

EXAMPLE 1.3-3

ACTIONS

| CONDITION | REQUIRED ACTION | COMPLETION TIME |
|---|---|---|
| A. One Function X train inoperable. | A.1 Restore Function X train to OPERABLE status. | 7 days<br><br>AND<br><br>10 days from discovery of failure to meet the LCO |
| B. One Function Y train inoperable. | B.1 Restore Function Y train to OPERABLE status. | 72 hours<br><br>AND<br><br>10 days from discovery of failure to meet the LCO |
| C. One Function X train inoperable.<br><br>AND<br><br>One Function Y train inoperable. | C.1 Restore Function X train to OPERABLE status.<br><br>OR<br><br>C.2 Restore Function Y train to OPERABLE status. | 72 hours<br><br><br><br>72 hours |

1.3 Completion Times

EXAMPLES (continued)

When one Function X train and one Function Y train are inoperable, Condition A and Condition B are concurrently applicable. The Completion Times for Condition A and Condition B are tracked separately for each train starting from the time each train was declared inoperable and the Condition was entered. A separate Completion Time is established for Condition C and tracked from the time the second train was declared inoperable (i.e., the time the situation described in Condition C was discovered).

If Required Action C.2 is completed within the specified Completion Time, Conditions B and C are exited. If the Completion Time for Required Action A.1 has not expired, operation may continue in accordance with Condition A. The remaining Completion Time in Condition A is measured from the time the affected train was declared inoperable (i.e., initial entry into Condition A).

The Completion Times of Conditions A and B are modified by a logical connector, with a separate 10 day Completion Time measured from the time it was discovered the LCO was not met. In this example, without the separate Completion Time, it would be possible to alternate between Conditions A, B, and C in such a manner that operation could continue indefinitely without ever restoring systems to meet the LCO. The separate Completion Time modified by the phrase "from discovery of failure to meet the LCO" is designed to prevent indefinite continued operation while not meeting the LCO. This Completion Time allows for an exception to the normal "time zero" for beginning the Completion Time "clock." In this instance, the Completion Time "time zero" is specified as commencing at the time the LCO was initially not met, instead of at the time the associated Condition was entered.

1.3 Completion Times

EXAMPLES (continued)

EXAMPLE 1.3-4

ACTIONS

| CONDITION | REQUIRED ACTION | COMPLETION TIME |
|---|---|---|
| A. One or more valves inoperable. | A.1 Restore valve(s) to OPERABLE status. | 4 hours |
| B. Required Action and associated Completion Time not met. | B.1 Be in MODE 3.<br><br>AND<br><br>B.2 Be in MODE 4. | 6 hours<br><br><br><br>12 hours |

A single Completion Time is used for any number of valves inoperable at the same time. The Completion Time associated with Condition A is based on the initial entry into Condition A and is not tracked on a per valve basis. Declaring subsequent valves inoperable, while Condition A is still in effect, does not trigger the tracking of separate Completion Times.

Once one of the valves has been restored to OPERABLE status, the Condition A Completion Time is not reset, but continues from the time the first valve was declared inoperable. The Completion Time may be extended if the valve restored to OPERABLE status was the first inoperable valve. The Condition A Completion Time may be extended for up to 4 hours provided this does not result in any subsequent valve being inoperable for > 4 hours.

If the Completion Time of 4 hours (plus the extension) expires while one or more valves are still inoperable, Condition B is entered.

1.3 Completion Times

EXAMPLES (continued)

EXAMPLE 1.3-5

ACTIONS

---------------------------------------NOTE---------------------------------------
Separate Condition entry is allowed for each inoperable valve.
------------------------------------------------------------------------------------

| CONDITION | REQUIRED ACTION | COMPLETION TIME |
|---|---|---|
| A. One or more valves inoperable. | A.1 Restore valve to OPERABLE status. | 4 hours |
| B. Required Action and associated Completion Time not met. | B.1 Be in MODE 3.<br><br>AND<br><br>B.2 Be in MODE 4. | 6 hours<br><br><br><br>12 hours |

The Note above the ACTIONS Table is a method of modifying how the Completion Time is tracked. If this method of modifying how the Completion Time is tracked was applicable only to a specific Condition, the Note would appear in that Condition rather than at the top of the ACTIONS Table.

The Note allows Condition A to be entered separately for each inoperable valve, and Completion Times tracked on a per valve basis. When a valve is declared inoperable, Condition A is entered and its Completion Time starts. If subsequent valves are declared inoperable, Condition A is entered for each valve and separate Completion Times start and are tracked for each valve.

1.3 Completion Times

EXAMPLES (continued)

If the Completion Time associated with a valve in Condition A expires, Condition B is entered for that valve. If the Completion Times associated with subsequent valves in Condition A expire, Condition B is entered separately for each valve and separate Completion Times start and are tracked for each valve. If a valve that caused entry into Condition B is restored to OPERABLE status, Condition B is exited for that valve.

Since the Note in this example allows multiple Condition entry and tracking of separate Completion Times, Completion Time extensions do not apply.

EXAMPLE 1.3-6

ACTIONS

| CONDITION | REQUIRED ACTION | COMPLETION TIME |
|---|---|---|
| A. One channel inoperable. | A.1 Perform SR 3.x.x.x.<br><br>OR<br><br>A.2 Reduce THERMAL POWER to ≤ 50% RTP. | Once per 8 hours<br><br><br><br>8 hours |
| B. Required Action and associated Completion Time not met. | B.1 Be in MODE 3. | 6 hours |

1.3 Completion Times

EXAMPLES (continued)

Entry into Condition A offers a choice between Required Action A.1 or A.2. Required Action A.1 has a "once per" Completion Time, which qualifies for the 25% extension, per SR 3.0.2, to each performance after the initial performance. The initial 8 hour interval of Required Action A.1 begins when Condition A is entered and the initial performance of Required Action A.1 must be complete within the first 8 hour interval. If Required Action A.1 is followed and the Required Action is not met within the Completion Time (plus the extension allowed by SR 3.0.2), Condition B is entered. If Required Action A.2 is followed and the Completion Time of 8 hours is not met, Condition B is entered.

If after entry into Condition B, Required Action A.1 or A.2 is met, Condition B is exited and operation may then continue in Condition A.

1.3 Completion Times

EXAMPLES (continued)

ACTIONS

| CONDITION | REQUIRED ACTION | COMPLETION TIME |
|---|---|---|
| A. One subsystem inoperable. | A.1 Verify affected subsystem isolated. | 1 hour<br><br>AND<br><br>Once per 8 hours thereafter |
| | AND<br><br>A.2 Restore subsystem to OPERABLE status. | 72 hours |
| B. Required Action and associated Completion Time not met. | B.1 Be in MODE 3. | 6 hours |
| | AND<br><br>B.2 Be in MODE 5. | 36 hours |

Required Action A.1 has two Completion Times. The 1 hour Completion Time begins at the time the Condition is entered and each "Once per 8 hours thereafter" interval begins upon performance of Required Action A.1.

If after Condition A is entered, Required Action A.1 is not met within either the initial 1 hour or any subsequent 8 hour interval from the previous performance (plus the extension allowed by SR 3.0.2), Condition B is entered. The Completion Time clock for Condition A does not stop after Condition B is entered, but continues from the time Condition A was initially entered. If Required Action A.1 is met after Condition B is entered, Condition B is exited and operation may continue in accordance with Condition A, provided the Completion Time for Required Action A.2 has not expired.

1.3 Completion Times

| | |
|---|---|
| IMMEDIATE COMPLETION TIME | When "Immediately" is used as a Completion Time, the Required Action should be pursued without delay and in a controlled manner. |

# 1.0 USE AND APPLICATION

## 1.4 Frequency

PURPOSE

The purpose of this section is to define the proper use and application of Frequency requirements.

DESCRIPTION

Each Surveillance Requirement (SR) has a specified Frequency in which the Surveillance must be met in order to meet the associated LCO. An understanding of the correct application of the specified Frequency is necessary for compliance with the SR.

The "specified Frequency" is referred to throughout this section and each of the Specifications of Section 3.0, Surveillance Requirement (SR) Applicability. The "specified Frequency" consists of the requirements of the Frequency column of each SR, as well as certain Notes in the Surveillance column that modify performance requirements.

Sometimes special situations dictate when the requirements of a Surveillance are to be met. They are "otherwise stated" conditions allowed by SR 3.0.1. They may be stated as clarifying Notes in the Surveillance, as part of the Surveillances, or both.

Situations where a Surveillance could be required (i.e., its Frequency could expire), but where it is not possible or not desired that it be preformed until sometime after the associated LCO is within its Applicability, represent potential SR 3.0.4 conflicts. To avoid these conflicts, the SR (i.e., the Surveillance or the Frequency) is stated such that it is only "required" when it can be and should be performed. With an SR satisfied, SR 3.0.4 imposes no restriction.

The use of "met" or "performed" in these instances conveys specific meanings. A Surveillance is "met" only when the acceptance criteria are satisfied. Known failure of the requirements of a Surveillance, even without a Surveillance specifically being "performed," constitutes a Surveillance not "met." "Performance" refers only to the requirement to specifically determine the ability to meet the acceptance criteria.

Some Surveillances contain notes that modify the Frequency of performance or the conditions during which the acceptance criteria must be satisfied. For these Surveillances, the MODE-entry restrictions of SR 3.0.4 may not apply. Such a Surveillance is not required to be performed prior to entering a MODE or other specified condition in the Applicability of the associated LCO if any of the following three conditions are satisfied:

1.4 Frequency

DESCRIPTION (continued)

a. The Surveillance is not required to be met in the MODE or other specified condition to be entered: or

b. The Surveillance is required to be met in the MODE or other specified condition to be entered, but has been performed within the specified Frequency (i.e., it is current) and is known not to be failed; or

c. The Surveillance is required to be met, but not performed, in the MODE or other specified condition to be entered, and is known no to be failed.

Examples 1.4-3, 1.4-4, 1.4-5, and 1.4-6 discuss these special situations.

EXAMPLES

The following examples illustrate the various ways that Frequencies are specified. In these examples, the Applicability of the LCO (LCO not shown) is MODES 1, 2, and 3.

1.4 Frequency

EXAMPLES (continued)

EXAMPLE 1.4-1

SURVEILLANCE REQUIREMENTS

| SURVEILLANCE | FREQUENCY |
|---|---|
| Perform CHANNEL CHECK. | 12 hours |

Example 1.4-1 contains the type of SR most often encountered in the Technical Specifications (TS). The Frequency specifies an interval (12 hours) during which the associated Surveillance must be performed at least one time. Performance of the Surveillance initiates the subsequent interval. Although the Frequency is stated as 12 hours, an extension of the time interval to 1.25 times the stated Frequency is allowed by SR 3.0.2 for operational flexibility. The measurement of this interval continues at all times, even when the SR is not required to be met per SR 3.0.1 (such as when the equipment is inoperable, a variable is outside specified limits, or the unit is outside the Applicability of the LCO). If the interval specified by SR 3.0.2 is exceeded while the unit is in a MODE or other specified condition in the Applicability of the LCO, and the performance of the Surveillance is not otherwise modified (refer to Example 1.4-3), then SR 3.0.3 becomes applicable.

If the interval as specified by SR 3.0.2 is exceeded while the unit is not in a MODE or other specified condition in the Applicability of the LCO for which performance of the SR is required, the Surveillance must be performed within the Frequency requirements of SR 3.0.2 prior to entry into the MODE or other specified condition. Failure to do so would result in a violation of SR 3.0.4.

1.4 Frequency

EXAMPLES (continued)

EXAMPLE 1.4-2

SURVEILLANCE REQUIREMENTS

| SURVEILLANCE | FREQUENCY |
|---|---|
| Verify flow is within limits. | Once within 12 hours after ≥ 25% RTP<br><br>AND<br><br>24 hours thereafter |

Example 1.4-2 has two Frequencies. The first is a one time performance Frequency, and the second is of the type shown in Example 1.4-1. The logical connector "AND" indicates that both Frequency requirements must be met. Each time reactor power is increased from a power level < 25% RTP to ≥ 25% RTP, the Surveillance must be performed within 12 hours.

The use of "once" indicates a single performance will satisfy the specified Frequency (assuming no other Frequencies are connected by "AND"). This type of Frequency does not qualify for the extension allowed by SR 3.0.2. "Thereafter" indicates future performances must be established per SR 3.0.2, but only after a specified condition is first met (i.e., the "once" performance in this example). If reactor power decreases to < 25% RTP, the measurement of both intervals stops. New intervals start upon reactor power reaching 25% RTP.

1.4 Frequency

EXAMPLES (continued)

EXAMPLE 1.4-3

SURVEILLANCE REQUIREMENTS

| SURVEILLANCE | FREQUENCY |
|---|---|
| ----------------------------NOTE----------------------------<br>Not required to be performed until 12 hours after ≥ 25% RTP.<br>-------------------------------------------------------------<br><br>Perform channel adjustment. | 7 days |

The interval continues whether or not the unit operation is < 25% RTP between performances.

As the Note modifies the required performance of the Surveillance, it is construed to be part of the "specified Frequency." Should the 7 day interval be exceeded while operation is < 25% RTP, this Note allows 12 hours after power reaches ≥ 25% RTP to perform the Surveillance. The Surveillance is still considered to be performed within the "specified Frequency." Therefore, if the Surveillance were not performed within the 7 day (plus the extension allowed by SR 3.0.2) interval, but operation was < 25% RTP, it would not constitute a failure of the SR or failure to meet the LCO. Also, no violation of SR 3.0.4 occurs when changing MODES, even with the 7 day Frequency not met, provided operation does not exceed 12 hours with power ≥ 25% RTP.

Once the unit reaches 25% RTP, 12 hours would be allowed for completing the Surveillance. If the Surveillance were not performed within this 12 hour interval, there would then be a failure to perform a Surveillance within the specified Frequency, and the provisions of SR 3.0.3 would apply.

1.4 Frequency

EXAMPLES (continued)

EXAMPLE 1.4-4

SURVEILLANCE REQUIREMENTS

| SURVEILLANCE | FREQUENCY |
|---|---|
| ---------------------------NOTE--------------------------- Only required to be met in MODE 1. ------------------------------------------------------------ | |
| Verify leakage rates are within limits. | 24 hours |

Example 1.4-4 specifies that the requirements of this Surveillance do not have to be met until the unit is in MODE 1. The interval measurement for the Frequency of this Surveillance continues at all times, as described in Example 1.4-1. However, the Note constitutes an "otherwise stated" exception to the Applicability of this Surveillance. Therefore, if the Surveillance were not performed within the 24 hour interval (plus the extension allowed by SR 3.0.2), but the unit was not in MODE 1, there would be no failure of the SR nor failure to meet the LCO. Therefore, no violation of SR 3.0.4 occurs when changing MODES, even with the 24 hour Frequency exceeded, provided the MODE change was not made into MODE 1. Prior to entering MODE 1 (assuming again that the 24 hour Frequency were not met), SR 3.0.4 would require satisfying the SR.

1.4 Frequency

EXAMPLES (continued)

EXAMPLE 1.4-5

SURVEILLANCE REQUIREMENTS

| SURVEILLANCE | FREQUENCY |
|---|---|
| ---------------------------NOTE---------------------------<br>Only required to be performed in MODE 1.<br>------------------------------------------------------------ | |
| Perform complete cycle of the valve. | 7 days |

The interval continues, whether or not the unit operation is in MODE 1,2 or 3 (the assumed Applicability of the associated LCO) between performances.

As the Note modifies the required performance of the Surveillance, the Note is construed to be part of the "specified Frequency." Should the 7 day interval be exceeded while operation is not in MODE 1, this Note allows entry into and operation in MODES 2 and 3 to perform the Surveillance. The Surveillance is still considered to be performed within the "specified Frequency" if completed prior to entering MODE 1. Therefore, if the Surveillance were not performed within the 7 day (plus the extension allowed by SR 3.0.2) interval, but operation was not in MODE 1, it would not constitute a failure of the SR or failure to meet the LCO. Also, no violation of SR 3.0.4 occurs when changing MODES, even with the 7 day Frequency not met, provided operation does not result in entry into MODE 1.

Once the unit reaches MODE 1, the requirement for the Surveillance to be performed within its specified Frequency applies and would require that the Surveillance had been performed. If the Surveillance were not performed prior to entering MODE 1, there would then be a failure to perform a Surveillance within the specified Frequency, and the provisions of SR 3.0.3 would apply.

1.4 Frequency

EXAMPLES (continued)

EXAMPLE 1.4-6

SURVEILLANCE REQUIREMENTS

| SURVEILLANCE | FREQUENCY |
|---|---|
| -----------------------------NOTE----------------------------- Not required to be met in MODE 3. ------------------------------------------------------------------ | |
| Verify parameter is within limits. | 24 hours |

Example 1.4-6 specifies that the requirements of this Surveillance do not have to be met while the unit is in MODE 3 (the assumed Applicability of the associated LCO is MODES 1,2, and 3). The interval measurement for the Frequency of this Surveillance continues at all times, as described in Example 1.4-1. However, the Note constitutes an "otherwise stated" exception to the Applicability of this Surveillance. Therefore, if the Surveillance were not performed within the 24 hour interval (plus the extension allowed by SR 3.0.2), and the unit was in MODE 3, there would be no failure of the SR nor failure to meet the LCO. Therefore, no violation of SR 3.0.4 occurs when changing MODES to enter MODE 3, even with the 24 hour Frequency exceeded, provided the MODE change does not result in entry into MODE 2. Prior to entering MODE 2 (assuming again that the 24 hour Frequency were not met), SR 3.0.4 would require satisfying the SR.

# CTS DISCUSSION OF CHANGES

## ITS Section 1.0: Use and Application

**ADMINISTRATIVE**

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

A2 Not used.

A3 Not used.

A4 The RSTS establishes MODES of operation which are equivalent to the Reactor Operating Conditions defined in Section 1.2 of the CTS. The CTS presents individual definitions for each Reactor Operating Condition. The MODE equivalent of these Conditions will be defined by the combination of reactivity condition (Keff), % Rated Thermal Power, Average Reactor Coolant Temperature and bolting status of the reactor vessel head closure studs in the ITS (MODE definition and Table 1.1-1). The CTS defines the reactivity condition in terms of a subcritical condition (expressed in $\%\Delta k/k$). The RSTS defines the reactivity condition in terms of Keff. The ITS will adopt the Keff convention treating the small absolute difference between Shutdown Margin and Keff as a purely administrative change. In addition, the overlap of Cold Shutdown and Refueling is eliminated with the ITS definitions such that the unit is only in one of the defined MODES. The relocation of the CTS definitions for Reactor Operating Conditions into the ITS Table 1.1-1 is considered a purely administrative change. This change is consistent with the RSTS method of presentation of MODES. The applicability of the Reactor Operating Condition definition changes will be evaluated at each occurrence of the defined Reactor Operating Condition in the CTS. Changes to the CTS will be discussed on an individual basis with the Specification. Each change will be evaluated to determine if the change represents a more stringent or less stringent requirement with respect to the current license basis.

A5 The CTS 1.2.1 reference to pressure in defining a Reactor Operating Condition is redundant to the requirements of CTS 3.1.2 which defines the allowable combination of Reactor Coolant System pressure and temperature. In establishing operational MODES in the ITS, the removal of the reference to pressure in defining a Reactor Operating Condition is considered an administrative change.

A6 CTS 1.2.2 defines Hot Shutdown in terms of a subcritical condition (1% Δk/k shutdown) and an average reactor coolant temperature of greater than or equal to 525°F. This Hot Shutdown operating condition definition will be modified to correlate with the MODE 4 (Hot Shutdown) criteria established in RSTS Table 1.1-1. The RSTS MODE 4 criteria (per Table 1.1-1) imposes a maximum average reactor coolant temperature criteria of 280°F and a minimum average reactor coolant temperature of 200°F. The lower average reactor coolant temperature band could represent more restrictive requirements on the operation of the facility. Specifically, equipment that was previously required when average reactor coolant temperature exceeded 350°F may now be required when the average reactor coolant temperature exceeds 200°F. The applicability of this Reactor Operating Condition definition change will be evaluated at each occurrence of the defined Hot Shutdown Applicability in the CTS. Changes to the CTS will be discussed on an individual basis with the Specification. Each change will be evaluated to determine if the change represents a more stringent or less stringent requirement with respect to the current license basis.

A7 CTS 1.2.4 which defines Hot Standby presently correlates to the RSTS MODE 2 (Startup) criteria. The CTS Hot Standby definition will be revised to correlate with the RSTS MODE 3 (Hot Standby) criteria. By adopting the RSTS convention, the CTS Hot Standby definition could impose more stringent requirements on the facility if this definition were substituted for the CTS Hot Standby in the Specification Applicability statements without consideration for the intent of the Specification (i.e. action to reduce reactor power level vice actions to take the reactor subcritical). For example, ACTIONS in the CTS that presently direct the unit to Hot Standby (which would allow critical operation at a power level below 2%) will now require that the reactor be taken to a subcritical condition (Keff < 0.99). Similarly, during a plant heatup, the new MODE definition would require equipment to be placed into service at a lower operating temperature (280°F vice 350° or 525°F) than required by the CTS. The applicability of this Reactor Operating Condition definition change will be evaluated at each occurrence of the defined Hot Standby Applicability in the CTS. Changes to the CTS will be discussed on an individual basis with the Specification. Each change will be evaluated to determine if the change represents a more stringent or less stringent requirement with respect to the current license basis.

A8 The CTS 1.3 definition of OPERABLE-OPERABILITY requires the capability of "*necessary* ... normal [(offsite)] AND emergency [(DG)] electrical power sources ... that are required for the system ... to perform its function(s)" (emphasis added). However, in MODES 1, 2, 3, and 4, CTS LCO 3.0.5 allows the features to be considered OPERABLE provided at least one source of power is still available and their redundant features are OPERABLE. In the ITS, the definition has been modified to require "normal OR emergency electrical power." For MODES 1, 2, 3, and 4, the CTS LCO 3.0.5 requirements are incorporated into the improved Technical Specification LCO 3.8.1 ACTIONS for when an emergency diesel generator or an offsite power source is inoperable.

# CTS DISCUSSION OF CHANGES

1.0-01a

For other than MODES 1, 2, 3, and 4 (i.e., "cold shutdown conditions"), LCO 3.0.5 is not applicable. However, the incorporation of shutdown electrical specifications, as discussed in Section 3.8, provide actions to ensure that activities that require components with no offsite power source or no onsite power source are prohibited or that these components will be declared inoperable and the appropriate actions taken. Although this change to the definition of OPERABLE-OPERABILITY may appear to be a less restrictive change, when viewed as a whole with the retention of CTS LCO 3.0.5-like actions when above MODE 5, the incorporation of a Safety Function Determination Program, and with the incorporation of specific shutdown electrical technical specifications in the ITS, this change is considered to be administrative in nature.

A9 The CTS 1.5.1 and 1.5.2 definitions for Trip Test and Channel Test, respectively, when combined, are considered to be equivalent to the RSTS definition of CHANNEL FUNCTIONAL TEST. Therefore, the CHANNEL FUNCTIONAL TEST definition from the RSTS has been adopted in its entirety. In addition, the sentence "The CHANNEL FUNCTIONAL TEST may be performed by means of any series of sequential, overlapping, or total channel steps so that the entire channel is functionally tested" was added to provide clarification for how the test may be performed. The addition of this sentence represents the continuation of the current operating practice which would allow testing in this manner. Lastly, the addition of this sentence establishes consistency with the CHANNEL CALIBRATION definition given in the RSTS and adopted for use in the ITS.

A10 Selected definitions are deleted because the CTS that use these definitions are not retained in the ITS; or the equivalent ITS will not use the defined term. Discussions of the technical aspects of these changes are addressed in the discussion of change (DOC) for the individual specifications where the phrase is used in the CTS. The removal of a definition that is not used in the ITS is an administrative change because it has no impact on the implementation of any existing requirement not addressed in the ITS conversion. These deleted definitions are: CTS 1.2.3, 1.4, 1.5.5, 1.8, and 1.11 through 1.15.

A11 This administrative change adds definitions to the ITS that are established in the RSTS but which do not exist as definitions in the CTS. The addition of the definitions is made to make the ITS consistent with RSTS. The addition of the definitions by itself does not add limitations or requirements on the facility and is therefore considered to be an administrative change. These additional definitions are: MODES, ACTIONS, LEAKAGE, CONTROL RODS, AXIAL POWER SHAPING RODS, PHYSICS TESTS, THERMAL POWER, ALLOWABLE THERMAL POWER, and SHUTDOWN MARGIN.

A12 The CTS 1.2.5 definition for Power Operation makes specific reference to the power range channels (nuclear instruments) as representing the instrumentation used to determine the transition from CTS Reactor Operating Condition Hot Standby to Power Operation. The ITS will establish the transition from Startup (MODE 2) to Power Operation (MODE 1) as a function of percent RATED THERMAL POWER. This change in identification criteria is considered to be administrative because the nuclear instrumentation is calibrated to a heat balance which represents a measure of the thermal power of the reactor.

A13 CTS 1.2.5 establishes the transition power level between the Hot Standby and Power Operation Reactor Operating Conditions as 2% rated power as indicated on the power range channels (nuclear instrumentation). The ITS will establish the transition power level as 5% RATED THERMAL POWER in accordance with Table 1.1-1 of the RSTS. The 5% RTP MODE transition criteria is adopted for the purpose of maintaining consistency with the RSTS and with the ANO-2 Technical Specifications.

The different MODES are typically defined as transition points when more or less equipment is required to be operable. The accident analyses defined in the SAR are not impacted by this change in MODE transition. These accidents are based on worst case conditions and are not dependent on MODES, other than for the assumption of the equipment available to operate during an accident. NUREG-1430 has been reviewed for those instances in which additional equipment OPERABILITY is required as a result of entering MODE 1 from MODE 2 and MODE 2 from MODE 1. In the instance of the first MODE change, the following Specifications were found:

3.1.8 Physics Test Exceptions,

3.2.5 Power Peaking Factors, and

3.4.1 RCS Pressure, Temperature, and Flow DNB Limits.

In the instance of the second MODE change, the following Specifications were found:

3.3.9 Source Range Neutron Flux and

3.3.10 Intermediate Range Neutron Flux.

The CTS requirements for Physics Testing (3.1.8) are based on RCS pressure and not MODES. As stated in Item A4, these requirements will be evaluated to determine if the change represents a more stringent or less stringent requirement with respect to the current license basis. LCO 3.2.5 is incorporated in the ANO ITS as 3.2.5, "Power Peaking" with an Applicability of MODE 1 with reactor power ≥ 20% RTP, as discussed in package section 3.2. Therefore, this difference in MODE 1 definition has no bearing with respect to LCO 3.2.5. The requirements of LCO 3.4.1 are not specified in the CTS and the inclusion of these requirements is considered to be more restrictive in total and a difference in MODE 1 definition has no bearing with respect to current requirements. The CTS requires OPERABILITY of the source and intermediate range neutron instrumentation during "startup and operation" while NUREG-1430 requires these instruments to be OPERABLE during MODES 2, 3, 4, and 5 (for source range) and MODE 2, When any CRD trip breaker is in the closed position and the CRD system is capable of rod withdrawal (for the intermediate range). The impact of the difference between 2% and 5% RTP on LCO 3.3.9 and 3.3.10 requirements will be evaluated to determine if the change represents a more stringent or less stringent requirement with respect to the current license basis, as discussed in DOC A4.

A13 (continued)

NUREG-1430 was also reviewed for those instances in which a REQUIRED ACTION directs entry into MODE 2 from MODE 1. The following Specifications were found to apply:

3.2.5 Power Peaking Factors,

3.4.1 RCS Pressure Temperature, and Flow DNB Limits, and

3.7.2 MSIVs

As previously discussed for LCOs 3.2.5 and 3.4.1, the change from 2% to 5% RTP has no effect on the requirements. With respect to LCO 3.7.2, The CTS require the plant to be placed in Hot Shutdown in the event one MSIV is inoperable. For this same CONDITION, NUREG-1430 requires placing the plant in MODE 2. Again, the difference between 2% and 5% RTP has no bearing on the less restrictive nature of the MSIV requirements. The impact of the difference between 2% and 5% RTP on LCO 3.7.2 requirements will be evaluated to determine if the change represents a more stringent or less stringent requirement with respect to the current license basis, as discussed in DOC A4.

Note: DOC A12 addresses the equivalence between the CTS reference to power range (nuclear) instrumentation and the ITS reference to RATED THERMAL POWER.

A14 The modification of CTS 1.2.6, Refueling Shutdown, to the RSTS equivalent MODE 6, Refueling, results in the deletion of the requirement that the reactor must be maintained subcritical by 1% dk/k even with all control rods removed and the coolant temperature at the decay heat removal pump suction is at the refueling temperature (normally 140°F). These conditions differ significantly from the RSTS Bases for LCO 3.9.1, Boron Concentration during Refueling Operations. The Bases for ITS LCO 3.9.1 state that the procedures establish a boron concentration that will maintain an overall core reactivity of Keff < 0.95 during fuel handling, with the control rods and fuel assemblies assumed to be in the most adverse configuration (least negative reactivity) allowed by unit procedures.

The RSTS definition for MODE 6, Refueling, in RSTS Table 1.1-1 will be adopted in the ITS. The review of RSTS 3.9.1 and its Bases will evaluate the implications of this change in definition and will categorize the adoption of RSTS 3.9.1 and its Bases as more restrictive or less restrictive as appropriate.

A15 CTS 1.9 currently defines Staggered Test Basis. The adoption of the RSTS definition for STAGGERED TEST BASIS in the ITS is considered an administrative change in that the required interval at which a component is actually surveilled is not changed. The manner of presentation in the Surveillance Requirements portion of the ITS will change; however, to reflect the RSTS definition. Further, each CTS which references a Staggered Test Basis will have to be individually evaluated and modified to reflect the formatting and presentation requirements of the RSTS definition.

A16 The CTS 1.5.6 definition for Heat Balance Calibration constitutes a specific application of a CHANNEL CALIBRATION to the power range nuclear instrumentation. In conformance with the terminology and format of the RSTS, the duplication of the term calibration will be eliminated through the consideration of the Heat Balance Calibration to be a type of CHANNEL CALIBRATION. This eliminates the need to retain the Heat Balance Calibration definition. [Note: The second portion of the CTS definition dealt with the methodology for the Heat Balance Calibration. As signified by the LATER indication, this information will be relocated into the Bases of ITS 3.3.1.]

A17 Not used.

A18 The CTS is revised to include ITS 1.2 which establishes the usage and convention for Logical Connectors used throughout the ITS. In addition, ITS 1.2 demonstrates through example the usage of the Logical Connectors. The ITS will adopt this usage and convention. This is an administrative change made to make the CTS conform to the NUREG-1430 convention.

A19 The CTS is revised to include ITS 1.3 which establishes the use and convention for Completion Times associated with the LCOs throughout the ITS. In addition, ITS 1.3 demonstrates through example the correct interpretation and usage of the Completion Times. The ITS will adopt this usage and convention. This is an administrative change made to make the CTS conform to the NUREG-1430 convention.

A20 The CTS is revised to include ITS 1.4 which establishes the use and convention for Frequency requirements associated with the Surveillance Requirements throughout the ITS. In addition, ITS 1.4 demonstrates through example the correct interpretation and usage of the Frequency requirements. The ITS will adopt this usage and convention. This is an administrative change made to make the CTS conform to the NUREG-1430 convention.

# CTS DISCUSSION OF CHANGES

## TECHNICAL CHANGE -- MORE RESTRICTIVE

M None

## TECHNICAL CHANGE -- LESS RESTRICTIVE

L None

## LESS RESTRICTIVE -- ADMINISTRATIVE DELETION OF REQUIREMENTS

LA1 None

# 1.0 USE AND APPLICATION

## 1.1 Definitions

Note

The defined terms of this section appear in capitalized type and are applicable throughout these Technical Specifications and Bases. (A1)

1. Definitions

~~The following terms are defined for uniform interpretation of these specifications.~~ (A1)

RATED THERMAL POWER (RTP)

1.1 RATED THERMAL POWER (RTP)

Rated power is a steady state reactor core output of 2568 MWt. (A1)

1.2 ~~REACTOR OPERATING CONDITIONS~~ (A1)

Table 1.1-1 MODE 5 & Note (b)

1.2.1 Cold Shutdown

<Apply Table 1.1-1; Note (b)> (A4)

The reactor is in the cold shutdown condition when it is subcritical by at least 1 percent Δk/k and $T_{avg}$ is no more than 200 F. ~~Pressure is defined by Specification 3.1.2.~~ (A5)

Table 1.1-1 MODE 4 & Note (b)

1.2.2 Hot Shutdown

<Apply Table 1.1-1; Note (b)> (A4)

The reactor is in the hot shutdown condition when it is subcritical by at least 1 percent Δk/k and $T_{avg}$ is ~~at or~~ less than 280°F and greater than ~~525 F~~ 200°F. (A6)

1.2.3 ~~Reactor Critical~~

~~The reactor is critical when the neutron chain reaction is self-sustaining and $K_{eff}$ = 1.0.~~ (A10)

Table 1.1-1 MODE 3

1.2.4 Hot Standby

The reactor is in the hot standby condition when all of the following conditions exist: (A4)

A. $T_{avg}$ is greater than ~~525 F~~ or equal to 280 °F. (A7)

B. The ~~reactor is critical~~ reactivity condition is <0.99.

~~C. Indicated neutron power on the power range channels is less than 2 percent of rated power.~~

Table 1.1-1 MODE 1 & Note (a)

1.2.5 Power Operation

<Apply Table 1.1-1; Note (a)> (A4)

The reactor is in a power operating condition when the indicated neutron power is above ~~2~~ 5 percent of rated power. ~~as indicated on the power range channels.~~ (A12) (A13)

Table 1.1-1 MODE 6 & Note (c)

1.2.6 Refueling ~~Shutdown~~

<Apply Table 1.1-1; Note (c)> (A4)

The reactor is in the refueling ~~shutdown~~ condition when, ~~even with all rods removed, the reactor would be subcritical by at least 1 percent Δk/k and the coolant temperature at the decay heat removal pump suction is at the~~ one or more reactor vessel head closure bolts is/are less than fully tensioned. (A14)

~~refueling temperature (normally 140F).~~ ~~Pressure is defined by Specification 3.1.2.~~ ~~A refueling shutdown refers to a shutdown to replace or rearrange all or a portion of the fuel assemblies and/or control rods.~~ A14 A5 A1

CORE ALTERATION

1.2.7 ~~Refueling Operation~~ <Add CORE ALTERATION> A1

~~An operation involving a change in core geometry by manipulation of fuel or control rods when the reactor vessel head is removed.~~ A1

Table 1.1-1 MODE 2 & Note (a)

1.2.8 Startup <Apply Table 1.1-1, Note (a)> A4

The reactor shall be considered in the startup mode when the ~~shutdown margin is reduced with the intent of going critical.~~ reactivity condition is ≥ 0.99 and the THERMAL POWER is ≤ 5 % RTP. A13

1.3 OPERABLE - OPERABILITY

OPERABLE - OPERABILITY

A system, subsystem, train, component or device shall be OPERABLE or have OPERABILITY when it is capable of performing its specified safety function(s)~~.~~, and when ~~Implicit in this definition shall be the assumption that~~ all necessary attendant instrumentation, controls, normal ~~and~~ or emergency electrical power ~~sources~~, cooling and ~~or~~ seal water, lubrication and ~~or~~ other auxiliary equipment that are required for the system, subsystem, train, component or device to perform its specified function~~s~~(s) ~~are also~~ capable of performing their related support function(s). A1 A8 A1

~~1.4 PROTECTION INSTRUMENTATION LOGIC~~ A1

~~1.4.1 Instrument Channel~~

~~An instrument channel is the combination of sensor, wires, amplifiers and output devices which are connected for the purpose of measuring the value of a process variable for the purpose of observation, control and/or protection. An instrument channel may be either analog or digital.~~ A10

~~1.4.2 Reactor Protection System~~

~~The reactor protection system is shown in Figures 7-1 and 7-9 of the FSAR. It is that combination of protective channels and associated circuitry which forms the automatic system that protects the reactor by control rod trip. It includes the four protection channels, their associated instrument channel inputs, manual trip switch, all rod drive control protective trip breakers and activating relays or coils.~~

~~A protection channel, as shown in Figure 7-1 of the FSAR (one of three or one of four independent channels, complete with sensors, sensor power supply~~ A10

<Add ACTIONS definition> A11

<Add MODE definition> A11

< Add LEAKAGE DEFINITION > A11

units, amplifiers and bistable modules provided for every reactor protection safety parameter), is a combination of instrument channels forming a single digital output to the protection system's coincidence logic. Each protection channel includes two key operated bypass switches, a protection channel bypass switch and a shutdown bypass switch. A10

1.4.4 Reactor Protection System Logic

This system utilizes reactor trip module relays (coils and contacts) in all four of the protection channels as shown in Figure 7-1 of the FSAR, to provide reactor trip signals for de-energizing the six control rod drive trip breakers. The control rod drive trip breakers are arranged to provide a one-out-of-two-times-two logic. Each element of the one-out-of-two-times-two logic is controlled by a separate set of two-out-of-four logic contacts from the four reactor protection channels. A10

1.4.5 Safety Features System

This system utilizes relay contact output from individual channels arranged in three analog sub-systems and two two-out-of-three logic sub-systems as shown in Figure 7-6 of the FSAR. The digital sub-system is wired to provide appropriate signals for the actuation of redundant safety features equipment on a two-of-three basis for any given parameter. A10

1.4.6 Degree of Redundancy

The difference between the number of operable channels and the number of channels which, when tripped, will cause an automatic system trip. A10

1.5 INSTRUMENTATION SURVEILLANCE A1

1.5.1 Trip Test

A trip test is a test of logic elements in a protection channel to verify their associated trip action. A9

1.5.2 Channel ~~Test~~ FUNCTIONAL < CHANNEL FUNCTIONAL TEST DEFINITION AS PRESENTED IN THE ITS. > A9

CHANNEL FUNCTIONAL TEST

A channel test is the injection of an internal or external test signal into the channel to verify its proper response, including alarm and/or trip initiating action, where applicable.

1.5.3 ~~Instrument~~ Channel Check <CHANNEL CHECK DEFINITION>

CHANNEL CHECK

An instrument channel check is a verification of acceptable instrument performance by observation of its behavior and/or state; this verification includes comparison of output and/or state of independent channels measuring the same variable. A1

< Add CONTROL RODS DEFINITION > A11

< Add AXIAL POWER SHAPING RODS DEFINITION > A11

<Add PHYSICS TESTS definition> A11

<Add CHANNEL CALIBRATION definition>

1.5.4 ~~Instrument~~ Channel Calibration

~~An instrument channel calibration is a test, and adjustment (if necessary), to establish that the channel output responds with acceptable range and accuracy to known values of the parameter which the channel measures or an accurate simulation of these values. Calibration shall encompass the entire channel, including equipment actuation, alarm or trip and shall be deemed to include the channel test.~~ A1

~~1.5.5 Heat Balance Check~~

~~A heat balance check is a comparison of the indicated neutron power and core thermal power.~~ A10

~~1.5.6 Heat Balance Calibration~~

~~An adjustment of the power range channel amplifiers output to agree with the core thermal power as determined by a weighted primary and secondary heat balance considering all heat losses.~~ A16

<LATER> (3.3A) Between 0 and 15% power, only the primary heat balance is considered. From 15 to 100% power the heat balance is weighted linearly with only the secondary heat balance being considered at 100% power. LATER

~~1.6 POWER DISTRIBUTION~~ A1

QUADRANT POWER TILT (QPT)

1.6.1 Quadrant Power Tilt CAPS (QPT)

~~Quadrant power tilt~~ (QPT) shall be defined by the following equation and is expressed as a percentage A1

$$QPT = 100\left(\frac{\text{Power in any core quadrant}}{\text{Average power of all quadrants}} - 1\right)$$

AXIAL POWER IMBALANCE

1.6.2 ~~Reactor~~ AXIAL Power Imbalance CAPS

~~Reactor~~ AXIAL power CAPS imbalance ~~is~~ shall be the power in the top half of the core minus the power in the bottom half of the core, expressed as a percentage of RATED THERMAL POWER (RTP) ~~rated power.~~ ~~Imbalance is monitored continuously by the RPS using input from the power range channels. Imbalance limits are defined in Specification 2.1 and imbalance setpoints are defined in Specification 2.3.~~ A1

<Add THERMAL POWER definition> A11

<Add ALLOWABLE THERMAL POWER definition>

~~1.7 REACTOR BUILDING~~

<LATER> (3.6)

~~Reactor building integrity exists when the following conditions are satisfied:~~

~~a. The equipment hatch is closed and sealed and both doors of the personnel lock and emergency lock are closed and sealed, or b. below.~~

~~b. At least one door on each of the personnel lock and emergency lock is closed and sealed during personnel access or repair.~~

~~c. All non-automatic reactor building isolation valves and blind flanges are closed as required.~~

~~d. All automatic reactor building isolation valves are operable or deactivated in the closed position.~~

~~e. The reactor building leakage determined at the last testing interval satisfies Specification 4.4.1.~~

LATER

~~1.8 FIRE SUPPRESSION WATER SYSTEM~~

~~The fire suppression water system consists of: water sources, pumps, and distribution piping with associated sectionalizing isolation valves. Such valves include the hose standpipe shutoff valves and the first valve ahead of the water flow alarm device on each sprinkler system.~~

A10

STAGGERED TEST BASIS

1.9 STAGGERED TEST BASIS

A staggered test basis shall consist ~~of:~~

~~a. A test schedule for n systems, subsystems, trains or designated components obtained by dividing the specified test interval into n equal subintervals.~~

~~b. The testing of one system, subsystem, train or designated component at the beginning of each subinterval.~~

A15

of the testing of one of the systems, subsystems, channels, or other designated components during the interval specified by the Surveillance Frequency, so that all systems, subsystems, channels, or other designated components are tested during n Surveillance Frequency intervals, where n is the total number of systems, subsystems, channels, or other designated components in the associated function.

Dose Equivalent I-131

1.10 **Dose Equivalent I-131**

The Dose Equivalent I-131 shall be the concentration of I-131 (microcurie/gram) which alone would produce the same thyroid dose as the quantity and isotopic mixture of I-131, I-132, I-133, I-134 and I-135 actually present. The thyroid dose conversion factors used for this calculation shall be those listed in Table III of TID-14844, "Calculation of Distance Factors for Power and Test Reactor Sites." A1

1.11 **Liquid Radwaste Treatment System**

A Liquid Radwaste Treatment System is a system designed and used for holdup, filtration, and/or demineralization of radioactive liquid effluents prior to their release to the evironment.

1.12 **Purge - Purging**

Purge or Purging is the controlled process of discharging air or gas from a confinement to reduce the airborne radioactivity concentration in such a manner that replacement air or gas is required to purify the confinement.

1.13 **Member(s) of the Public**

Member(s) of the Public shall include all persons who are not occupationally associated with the plant. This category does not include employees of the utility, its contractors or vendors. Also excluded from this category are persons who enter the site to service equipment or to make deliveries. This category does include persons who use portions of the site for recreational, occupational or other purposes not associated with the plant. A10

1.14 **Exclusion Area**

The exclusion area is that area surrounding ANO within a minimum radius of .65 miles of the reactor buildings and controlled to the extent necessary by the licensee for purposes of protection of individuals from exposure to radiation and radioactive materials.

1.15 **Unrestricted Area**

An unrestricted area shall be any area beyond the exclusion area boundary.

1.16 **Core Operating Limits Report**

COLR

The CORE OPERATING LIMITS REPORT is the ANO-1 specific document that provides core operating limits for the current operating reload cycle. These cycle-specific core operating limits shall be determined for each reload cycle in accordance with Technical Specification ~~6.12.3~~. Plant operation within these operating limits is addressed in individual specifications. A1

5.6.5

<Add SHUTDOWN MARGIN Definition> A11

(LATER) (3.4A)

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

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

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

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

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

3.1.1.6 Decay Heat Removal

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

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

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

3. Decay Heat Removal Loop (A)** A8

4. Decay Heat Removal Loop (B)**

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

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

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

**The normal or emergency power source may be inoperable when the reactor is in a cold shutdown condition. A8

<Insert CTS 23A>

3.1.4 Reactor Coolant System Activity

Specification

<LATER> (3.4B)

3.1.4.1 Whenever the reactor is operating under steady-state conditions, the following conditions shall be met. — LATER

a. The total specific activity of the primary coolant shall not exceed 72/$\bar{E}$ µCi/gm where $\bar{E}$ is the sum of the average beta energy and average gamma energy per disintegration in MEV/disintegration. — (A1)

$\bar{E}$ - AVERAGE DISINTEGRATION ENERGY

b. The I-131 dose equivalent of the radioiodine activity in the primary coolant shall not exceed 3.5 µCi/gm.

c. If the radioactivity in the primary coolant exceeds the limits given above, corrective action shall be taken immediately to return the coolant activity to within these specifications. If the specific activity limits given above cannot be achieved within 24 hours, the reactor shall be brought to a hot shutdown condition using normal operating procedures. If the coolant radioactivity is not reduced to acceptable limits within an additional 48 hours, the reactor shall be brought to a cold shutdown condition and the cause of the out-of-specification operation ascertained.

Bases

<LATER> (3.4B)

Rupture of a steam generator tube would allow primary coolant activity to enter the secondary coolant. The major portion of this activity is noble gases and would be released to the atmosphere from the condenser vacuum pump or a relief valve. Activity would continue to be released until the operator could reduce the primary system pressure below the setpoint of the secondary relief valves and could isolate the faulty steam generator. The worst credible set of circumstances is considered to be a double-ended break of a single steam generator tube, followed by isolation of the faulty steam generator within 34 minutes after the tube break. Assuming the full differential pressure across the steam generator, no more than one-quarter of the total primary coolant could be released to the secondary coolant in this period. The decay heat during this period of 1 hour for pressure reduction will generate steam in the secondary system representing less than 15 weight percent of the secondary system. — LATER

The parameters assumed in the dose analysis for the single steam generator tube failure included the following values:

1) total primary coolant volume (mass) = 5.2 x $10^5$ lbs.

2) total secondary coolant volume (mass) = 2 x $10^6$ lbs.

3) leakage rate from primary to secondary system = 1 gpm.

4) fission product decay heat energy for 1 hour = 1.56 x $10^8$ BTU.

1.2
1.3
1.4

**<INSERT CTS 23A>**

< Add Section 1.2, Logical Connectors > A18

< Add Section 1.3, Completion Times > A19

< Add Section 1.4, Frequency > A20

3.1.6 Leakage

Specification

3.1.6.1 If the total reactor coolant leakage rate exceeds 10 gpm, the reactor shall be shutdown within 24 hours of detection.

3.1.6.2 If unidentified reactor coolant leakage (exceeding normal evaporative losses) exceeds 1 gpm or if any reactor coolant leakage is evaluated as unsafe, the reactor shall be shutdown within 24 hours of detection.

<LATER> (3.4B) LATER

ANO-250

3.1.6.3.a If it is determined that any reactor coolant leakage exists through a non-isolable fault in a reactor coolant system strength boundary (such as the reactor vessel, piping, valve body, etc., except steam generator tubes), the reactor shall be shutdown and a cooldown to the cold shutdown condition shall be initiated within 24 hours of detection.

Pressure Boundary LEAKAGE

A1

3.1.6.3.b If the leakage through the tubes of any one steam generator equals or exceeds 150 gallons per day (0.104 gpm), a reactor shutdown shall be initiated within 4 hours and the reactor shall be in the cold shutdown condition within the next 30 hours.

3.1.6.4 Deleted

3.1.6.5 Action to evaluate the safety implication of reactor coolant leakage shall be initiated within 4 hours of detection. The nature, as well as the magnitude of the leak, shall be considered in this evaluation. The safety evaluation shall assure that the exposure of offsite personnel to radiation is within the guidelines of 10CFR20.

<LATER> (3.4B) LATER

3.1.6.6 If reactor shutdown is required per Specification 3.1.6.1, 3.1.6.2, or 3.1.6.3 the reactor shall not be restarted until the leak is repaired or until the problem is otherwise corrected.

3.1.6.7 When the reactor is at power operation, three reactor coolant leak detection systems of different operating principles shall be in operation. One of these systems is sensitive to radioactivity and consists of a radioactive gas detector and an air particulate activity detector. Both of these instruments may be out-of-service simultaneously for a period of no more than 72 hours provided two other means are available to detect leakage and reactor building air samples are taken and analyzed in the laboratory at least once per shift; otherwise, be in at least Hot Standby within the next 6 hours and in Cold Shutdown within the following 30 hours.

3.1.6.8 Loss of reactor coolant through reactor coolant pump seals and system valves to connecting systems which

1.1

3.8 FUEL LOADING AND REFUELING

Applicability

<LATER> (3.9) Applies to fuel loading and refueling operations. LATER

Objective

To assure that fuel loading, refueling and fuel handling operations are performed in a responsible manner.

Specification

3.8.1 Radiation levels in the reactor building refueling area shall be monitored by instrument RE-8017. Radiation levels in the spent fuel storage area shall be monitored by instrument RE-8009. If any of these instruments become inoperable, portable survey instrumentation, having the appropriate ranges and sensitivity to fully protect individuals involved in refueling operation, shall be used until the permanent instrumentation is returned to service. R TRM

<LATER> (3.9) 3.8.2 Core subcritical neutron flux shall be continuously monitored by at least two neutron flux monitors, each with continuous indication available, whenever core geometry is being changed. When core geometry is not being changed, at least one neutron flux monitor shall be in service. LATER

3.8.3.a. At least one decay heat removal loop shall be in operation.* Otherwise, suspend all operations involving an increase in the reactor decay heat load or a reduction in boron concentration of the reactor coolant system, and close all containment penetrations providing access from the containment atmosphere to the outside atmosphere within 4 hours.

b. When the water level above the top of the irradiated fuel assemblies seated within the reactor pressure vessel is less than 23 feet, two decay heat removal loops shall be operable** A8

Otherwise, immediately initiate corrective action to return the required loops to operable status as soon as possible.

3.8.4 During reactor vessel head removal and while loading and unloading fuel from the reactor, the boron concentration shall be maintained at not less than that required for refueling shutdown.

3.8.5 Direct communications between the control room and the refueling personnel in the reactor building shall exist whenever changes in core geometry are taking place. R TRM

<LATER> (3.9) *The decay heat removal loop may be removed from operation for up to 1 hour per 8 hour period during the performance of core alterations. LATER

**The normal or emergency power source may be inoperable for each shutdown cooling loop. A8

# NO SIGNIFICANT HAZARDS CONSIDERATIONS
# GENERIC EVALUATIONS

**"R" - Relocation of requirements:**

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

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

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

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

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

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

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

# 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.

## NO SIGNIFICANT HAZARDS CONSIDERATIONS
## GENERIC EVALUATIONS

**"M" - More restrictive changes to requirements:**

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

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

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

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

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

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

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

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

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

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

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

# NO SIGNIFICANT HAZARDS CONSIDERATIONS STATEMENTS

## ITS Section 1.0: Use and Application

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

**No unit specific "Less Restrictive" changes identified.**

# ITS DISCUSSION OF DIFFERENCES
## ITS Section 1.0: Use and Application

1 DE I-131 - The DOSE EQUIVALENT I-131 markup reflects that ANO Unit-1 CTS 1.10 presently specifies that the dose conversion factors specified in TID-14844 be used in the determination of DOSE EQUIVALENT I-131. Therefore, the second reference provided in the RSTS is shown as deleted, or more appropriately, as not having been adopted. This change is consistent with current license basis.

2 Not used.

3 Not used.

4 PTLR - The definition of PRESSURE AND TEMPERATURE LIMITS REPORT (PTLR) is not adopted. ANO-1 will maintain the RCS Pressure and Temperature Curves and Limits in the ITS and will not implement a PTLR at this time. Since a PTLR is not implemented, the definition serves no purpose and has been deleted. This change is consistent with current license basis.

5 PHYSICS TESTS - The specific chapter reference in part "a." of the PHYSICS TESTS definition was deleted and the plant specific usage of SAR versus FSAR was incorporated. This change was made due to the non-standard nature of the ANO-1 SAR. Removal of the reference to a specific chapter simply insured that all physics testing referenced in the SAR were encompassed by this definition. This change is consistent with current license basis.

6 The definitions of EMERGENCY FEEDWATER INITIATION AND CONTROL (EFIC) RESPONSE TIME, ENGINEERED SAFETY FEATURE (ESF) RESPONSE TIME, and REACTOR PROTECTION SYSTEM (RPS) RESPONSE TIME were not incorporated. These terms and the referenced testing were not incorporated into ITS because they were not consistent with CTS . Response time testing of these systems, as required by specifications in NUREG-1430, is not required by CTS. This change is consistent with current license basis.

7 EFPD - Incorporates TSTF-125, Rev. 1.

8 CHANNEL FUNCTIONAL TEST - Incorporates TSTF-124.
CHANNEL CALIBRATION - Incorporates TSTF-124.

9 NUCLEAR ENTHALPY RISE HOT CHANNEL FACTOR and NUCLEAR HEAT FLUX HOT CHANNEL FACTOR - will not be incorporated into the Definitions section of the ITS because these terms are not used in any specific ITS LCO. Consistent with current license basis and unit specific surveillance capability, ITS 3.2.5 will require that core linear heat rate (LHR) limits be maintained in accordance with the limits established in the COLR. This change is consistent with current license basis.

10 Not used.

# ITS DISCUSSION OF DIFFERENCES

1.0-03 11 APSRs - The definition of AXIAL POWER SHAPING RODS (APSRs) has been modified to specify that these are the control components with part length absorbers. This specifically excludes the full length control components (regulating rods) when they are being used to control the axial power distribution of the reactor. This change was approved by the NRC in for the Oconee Nuclear Station ITS conversion. A draft generic change, designated as ANO-1-062, has been submitted to the BWOG for processing.

12 CHANNEL CALIBRATION - Incorporates TSTF-019.

1.0-04 13 NOT USED.

1.0-04 14 LEAKAGE - Incorporates TSTF-040.

15 Not used.

16 Not used.

17 La - As a result of a meeting between the NEI Tech Spec Task Force and the NRC Tech Spec Branch and Containment System Branch on October 18, 1995 concerning 10 CFR 50, Appendix J, Option B implementation, a definition of La is not adopted in the ITS. "La" will be described in the program description for the Reactor Building (Containment) Leak Rate Testing Program. This is consistent with current license basis.

18 Not used.

1.0-05 19 Not used.

20 Incorporates TSTF-205, Rev 3.

21 The definition of RATED THERMAL POWER is revised to retain the CTS usage of "steady state." This clarifies the definition and is consistent with the ANO-1 CTS and with NRC enforcement guidance concerning rated thermal power level control. The definition of ALLOWABLE THERMAL POWER is also revised for consistency.

ANO-242 22 Incorporates TSTF-284, Rev 3.

## 1.0 USE AND APPLICATION

### 1.1 Definitions

CTS

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

The defined terms of this section appear in capitalized type and are applicable throughout these Technical Specifications and Bases.

---

| Term | Definition | |
|---|---|---|
| ACTIONS | ACTIONS shall be that part of a Specification that prescribes Required Actions to be taken under designated Conditions within specified Completion Times. | N/A |
| ALLOWABLE THERMAL POWER | ALLOWABLE THERMAL POWER shall be the maximum steady state reactor core heat transfer rate to the reactor coolant permitted by consideration of the number and configuration of reactor coolant pumps (RCPs) in operation. | 21 N/A |
| AXIAL POWER IMBALANCE | AXIAL POWER IMBALANCE shall be the power in the top half of the core, expressed as a percentage of RATED THERMAL POWER (RTP), minus the power in the bottom half of the core, expressed as a percentage of RTP. | 1.6.2 |
| AXIAL POWER SHAPING RODS (APSRs) | APSRs shall be the control components with part length absorbers used to control the axial power distribution of the reactor core. The APSRs are positioned manually by the operator and are not trippable. | 11 N/A |
| CHANNEL CALIBRATION | A CHANNEL CALIBRATION shall be the adjustment, as necessary, of the channel output such that it responds within the necessary range and accuracy to known values of the parameter that the channel monitors. The CHANNEL CALIBRATION shall encompass all devices in the channel required for channel OPERABILITY and ~~the entire channel, including the required sensor, alarm, display, and trip functions, and shall include~~ the CHANNEL FUNCTIONAL TEST. Calibration of instrument channels with resistance temperature detector (RTD) or thermocouple sensors may consist of an inplace qualitative assessment of sensor behavior and normal calibration of the remaining adjustable devices in the channel. ~~Whenever a~~ | 1.5.4 20 12 |

1.0-03

(continued)

CTS

## 1.1 Definitions

**CHANNEL CALIBRATION (continued)**

~~sensing element is replaced, the next required CHANNEL CALIBRATION shall include an inplace cross calibration that compares the other sensing elements with the recently installed sensing element.~~ The CHANNEL CALIBRATION may be performed by means of any series of sequential, overlapping, or total channel steps ~~so that the entire channel is calibrated.~~ 12 1.5.4 20

~~The CHANNEL CALIBRATION shall also include testing of safety related Reactor Protection System (RPS), Engineered Safety Feature Actuation System (ESFAS), and Emergency Feedwater Initiation and Control (EFIC) bypass functions for each channel affected by the bypass operation.~~ 8

**CHANNEL CHECK**

A CHANNEL CHECK shall be the qualitative assessment, by observation, of channel behavior during operation. This determination shall include, where possible, comparison of the channel indication and status to other indications or status derived from independent instrument channels measuring the same parameter. 1.5.3

**CHANNEL FUNCTIONAL TEST**

of all devices [illegible] channel required for channel OPERA[illegible]

A CHANNEL FUNCTIONAL TEST shall be the injection of a simulated or actual signal into the channel as close to the sensor as practicable to verify OPERABILITY ~~including required alarms, interlocks, display, and trip functions.~~ 20 1.5.2 ~~The ESFAS CHANNEL FUNCTIONAL TEST shall also include testing of ESFAS safety related bypass functions for each channel affected by bypass operation.~~ 8 <INSERT SENTENCE>

**CONTROL RODS**

CONTROL RODS shall be all full length safety and regulating rods that are used to shut down the reactor and control power level during maneuvering operations. N/A

**CORE ALTERATION**

CORE ALTERATION shall be the movement of any fuel, sources, or reactivity control components, within the reactor vessel with the vessel head removed and fuel in the vessel. Suspension of CORE 1.2.7

20

The CHANNEL FUNCTIONAL TEST may be performed by means of any series of sequential, overlapping, or total steps.

(continued)

1.1 Definitions

CTS

| | | |
|---|---|---|
| CORE ALTERATION (continued) | ALTERATIONS shall not preclude completion of movement of a component to a safe position. | |
| CORE OPERATING LIMITS REPORT (COLR) | The COLR is the ANO-1 ~~unit~~ specific document that provides cycle specific parameter limits for the current reload cycle. These cycle specific PARAMETER limits shall be determined for each reload cycle in accordance with Specification 5.6.5. Plant operation within these limits is addressed in individual Specifications. | 1.16 |
| DOSE EQUIVALENT I-131 | DOSE EQUIVALENT I-131 shall be that concentration of I-131 (microcuries/gram) that alone would produce the same thyroid dose as the quantity and isotopic mixture of I-131, I-132, I-133, I-134, and I-135 actually present. The thyroid dose conversion factors used for this calculation shall be those listed in ~~[~~Table III of TID-14844, AEC, 1962, "Calculation of Distance Factors for Power and Test Reactor Sites." ~~or those listed in Table E-7 of Regulatory Guide 1.109, Rev. 1, NRC, 1977, or ICRP 30, Supplement to Part 1, page 192-212, Table titled, "Committed Dose Equivalent in Target Organs or Tissues per Intake of Unit Activity"].~~ (1) | 1.10 |
| Ē—AVERAGE DISINTEGRATION ENERGY | Ē shall be the average (weighted in proportion to the concentration of each radionuclide in the reactor coolant at the time of sampling) of the sum of the average beta and gamma energies per disintegration (in MeV) for isotopes, other than iodines, with half lives > [15] minutes, making up at least 95% of the total noniodine activity in the coolant. | 3.1.4.1.a |
| ~~EFFECTIVE FULL POWER DAY (EFPD)~~ | ~~EFPD shall be the ratio of the number of hours of production of a given THERMAL POWER to 24 hours, multiplied by the ratio of the given THERMAL POWER to the RTP. One EFPD is equivalent to the thermal energy produced by operating the reactor core at RTP for one full day.~~ (7) | |
| ~~EMERGENCY FEEDWATER INITIATION AND CONTROL (EFIC) RESPONSE TIME~~ | ~~The EFIC RESPONSE TIME shall be that time interval from when the monitored parameter exceeds its EFIC actuation setpoint at the channel sensor until the emergency feedwater equipment is~~ (6) | N/A |

(continued)

CTS

## 1.1 Definitions

| | |
|---|---|
| ~~EMERGENCY FEEDWATER INITIATION AND CONTROL (EFIC) RESPONSE TIME (continued)~~ | ~~capable of performing its function (i.e., valves travel to their required positions, pumps discharge pressures reach their required values, etc.). Times shall include diesel generator starting and sequence loading delays, where applicable. The response time may be measured by means of any series of sequential, overlapping, or total steps so that the entire response time is measured.~~ (6) |
| ~~ENGINEERED SAFETY FEATURE (ESF) RESPONSE TIME~~ | ~~The ESF RESPONSE TIME shall be that time interval from when the monitored parameter exceeds its ESF actuation setpoint at the channel sensor until the ESF equipment is capable of performing its safety function (i.e., the valves travel to their required positions, pump discharge pressures reach their required values, etc.). Times shall include diesel generator starting and sequence loading delays, where applicable. The response time may be measured by means of any series of sequential, overlapping, or total steps so that the entire response time is measured.~~ (6) |
| ~~$L_a$~~ | ~~The maximum allowable containment leakage rate, $L_a$, shall be [0.25]% of containment air weight per day at the calculated peak containment pressure ($P_a$).~~ (17) |
| LEAKAGE | LEAKAGE shall be: |
| | a. <u>Identified LEAKAGE</u> |
| | 1. LEAKAGE, such as that from pump seals or valve packing (except RCP seal water injection or leakoff), that is captured and conducted to collection systems or a sump or collecting tank; N/A |
| | 2. LEAKAGE into the containment atmosphere from sources that are both specifically located and known either not to interfere with the operation of leakage detection systems or not to be pressure boundary LEAKAGE; or |

1.0-01

(continued)

CTS

## 1.1 Definitions

LEAKAGE (continued)

3. Reactor Coolant System (RCS) LEAKAGE through a steam generator (SG) to the Secondary System;

b. Unidentified LEAKAGE

All LEAKAGE (except RCP seal water injection and leakoff) that is not identified LEAKAGE ~~or controlled LEAKAGE;~~ (14)

c. Pressure Boundary LEAKAGE

LEAKAGE (except SG LEAKAGE) through a nonisolable fault in an RCS component body, pipe wall, or vessel wall. 3.1.6.3.a

MODE

A MODE shall correspond to any one inclusive combination of core reactivity condition, power level, average reactor coolant temperature, and reactor vessel head closure bolt tensioning specified in Table 1.1-1 with fuel in the reactor vessel. 1.2.1 1.2.2 1.2.4 1.2.5 1.2.6 1.2.8

~~NUCLEAR HEAT FLUX HOT CHANNEL FACTOR $F_Q(Z)$~~

~~$F_Q(Z)$ shall be the maximum local linear power density in the core divided by the core average fuel rod linear power density, assuming nominal fuel pellet and fuel rod dimensions.~~ (9)

~~NUCLEAR ENTHALPY RISE HOT CHANNEL FACTOR ($F^N_{\Delta H}$)~~

~~($F^N_{\Delta H}$) shall be the ratio of the integral of linear power along the fuel rod on which minimum departure from nucleate boiling ratio occurs, to the average fuel rod power.~~

OPERABLE—OPERABILITY

A system, subsystem, train, component, or device shall be OPERABLE or have OPERABILITY when it is capable of performing its specified safety function(s) and when all necessary attendant instrumentation, controls, normal or emergency electrical power, cooling and seal water, lubrication, and other auxiliary equipment that are required for the system, subsystem, train, component, or device to perform its specified safety function(s) are also capable of performing their related support function(s). 1.3

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(continued)

CTS

1.1 Definitions (continued)

| | | |
|---|---|---|
| PHYSICS TESTS | PHYSICS TESTS shall be those tests performed to measure the fundamental nuclear characteristics of the reactor core and related instrumentation.<br><br>These tests are:<br><br>a. Described in ~~Chapter [14, Initial Test Program] of the FSAR,~~ the SAR;<br><br>b. Authorized under the provisions of 10 CFR 50.59; or<br><br>c. Otherwise approved by the Nuclear Regulatory Commission. | N/A<br>5 |
| ~~PRESSURE AND TEMPERATURE LIMITS REPORT (PTLR)~~ | ~~The PTLR is the unit specific document that provides the reactor vessel pressure and temperature limits, including heatup and cooldown rates, for the current reactor vessel fluence period. These pressure and temperature limits shall be determined for each fluence period in accordance with Specification 5.6.6. Plant operation within these operating limits is addressed in LCO 3.4.3, "RCS Pressure and Temperature (P/T) Limits," and LCO 3.4.12, "Low Temperature Overpressure Protection (LTOP) System."~~ | 4 |
| QUADRANT POWER TILT (QPT) | QPT shall be defined by the following equation and is expressed as a percentage.<br><br>$QPT = 100 \left( \frac{\text{Power in any Core Quadrant}}{\text{Average Power of all Quadrants}} - 1 \right)$ | 1.6.1 |
| RATED THERMAL POWER (RTP) | RTP shall be a total steady state reactor core heat transfer rate to the reactor coolant of ~~[2544]~~ 2568 MWt. | 21<br>1.1 |
| ~~REACTOR PROTECTION SYSTEM (RPS) RESPONSE TIME~~ | ~~The RPS RESPONSE TIME shall be that time interval from when the monitored parameter exceeds its RPS trip setpoint at the channel sensor until electrical power is interrupted at the control rod drive trip breakers. The response time may be measured by means of any series of sequential, overlapping, or total steps so that the entire response time is measured.~~ | 6 |

(continued)

CTS

1.1 Definitions (continued)

SHUTDOWN MARGIN (SDM)

SDM shall be the instantaneous amount of reactivity by which the reactor is subcritical or would be subcritical from its present condition assuming: N/A

a. All full length CONTROL RODS (safety and regulating) are fully inserted except for the single CONTROL ROD of highest reactivity worth, which is assumed to be fully withdrawn. With any CONTROL ROD not capable of being fully inserted, the reactivity worth of these CONTROL RODS must be accounted for in the determination of SDM;

b. In MODES 1 and 2, the fuel and moderator temperatures are changed to the nominal zero power design level; and

c. There is no change in APSR position.

STAGGERED TEST BASIS

A STAGGERED TEST BASIS shall consist of the testing of one of the systems, subsystems, channels, or other designated components during the interval specified by the Surveillance Frequency, so that all systems, subsystems, channels, or other designated components are tested during *n* Surveillance Frequency intervals, where *n* is the total number of systems, subsystems, channels, or other designated components in the associated function. 1.9

THERMAL POWER

THERMAL POWER shall be the total reactor core heat transfer rate to the reactor coolant. N/A

Table 1.1-1 (page 1 of 1)
MODES

| MODE | TITLE | REACTIVITY CONDITION ($k_{eff}$) | % RATED THERMAL POWER[a] | AVERAGE REACTOR COOLANT TEMPERATURE (°F) | CTS |
|---|---|---|---|---|---|
| 1 | Power Operation | ≥ 0.99 | > 5 | NA | 1.2.5 |
| 2 | Startup | ≥ 0.99 | ≤ 5 | NA | 1.2.8 |
| 3 | Hot Standby | < 0.99 | NA | ≥ ~~[330]~~ 280 | 1.2.4 |
| 4 | Hot Shutdown[b] | < 0.99 | NA | ~~[330]~~ 280 > $T_{avg}$ > ~~[200]~~ 200 | 1.2.2 |
| 5 | Cold Shutdown[b] | < 0.99 | NA | ≤ ~~[200]~~ 200 | 1.2.1 |
| 6 | Refueling[c] | NA | NA | NA | 1.2.6 |

(a) Excluding decay heat. 1.2.5 & 1.2.8

(b) All reactor vessel head closure bolts fully tensioned. 1.2.1 & 1.2.2

(c) One or more reactor vessel head closure bolts less than fully tensioned. 1.2.6

CTS

N/A

## 1.0 USE AND APPLICATION

## 1.2 Logical Connectors

PURPOSE

The purpose of this section is to explain the meaning of logical connectors.

Logical connectors are used in Technical Specifications (TS) to discriminate between, and yet connect, discrete Conditions, Required Actions, Completion Times, Surveillances, and Frequencies. The only logical connectors that appear in TS are AND and OR. The physical arrangement of these connectors constitutes logical conventions with specific meanings.

BACKGROUND

Several levels of logic may be used to state Required Actions. These levels are identified by the placement (or nesting) of the logical connectors and by the number assigned to each Required Action. The first level of logic is identified by the first digit of the number assigned to a Required Action and the placement of the logical connector in the first level of nesting (i.e., left justified with the number of the Required Action). The successive levels of logic are identified by additional digits of the Required Action number and by successive indentations of the logical connectors.

When logical connectors are used to state a Condition, Completion Time, Surveillance, or Frequency, only the first level of logic is used, and the logical connector is left justified with the statement of the Condition, Completion Time, Surveillance, or Frequency.

EXAMPLES

The following examples illustrate the use of logical connectors.

(continued)

CTS

N/A

## 1.2 Logical Connectors

EXAMPLES (continued)

EXAMPLE 1.2-1

ACTIONS

| CONDITION | REQUIRED ACTION | COMPLETION TIME |
|---|---|---|
| A. LCO not met. | A.1 Verify . . .<br><br>AND<br><br>A.2 Restore . . . | |

In this example the logical connector AND is used to indicate that when in Condition A, both Required Actions A.1 and A.2 must be completed.

(continued)

CTS

N/A

## 1.2 Logical Connectors

EXAMPLES (continued)

EXAMPLE 1.2-2

ACTIONS

| CONDITION | REQUIRED ACTION | COMPLETION TIME |
|---|---|---|
| A. LCO not met. | A.1 Trip . . .<br>OR<br>A.2.1 Verify . . .<br>AND<br>A.2.2.1 Reduce . . .<br>OR<br>A.2.2.2 Perform . . .<br>OR<br>A.3 Align . . . | |

This example represents a more complicated use of logical connectors. Required Actions A.1, A.2, and A.3 are alternative choices, only one of which must be performed as indicated by the use of the logical connector OR and the left justified placement. Any one of these three Actions may be chosen. If A.2 is chosen, then both A.2.1 and A.2.2 must be performed as indicated by the logical connector AND. Required Action A.2.2 is met by performing A.2.2.1 or A.2.2.2. The indented position of the logical connector OR indicates that A.2.2.1 and A.2.2.2 are alternative choices, only one of which must be performed.

CTS
N/A

## 1.0 USE AND APPLICATION

### 1.3 Completion Times

PURPOSE

The purpose of this section is to establish the Completion Time convention and to provide guidance for its use.

BACKGROUND

Limiting Conditions for Operation (LCOs) specify minimum requirements for ensuring safe operation of the unit. The ACTIONS associated with an LCO state Conditions that typically describe the ways in which the requirements of the LCO can fail to be met. Specified with each stated Condition are Required Action(s) and Completion Time(s).

DESCRIPTION

The Completion Time is the amount of time allowed for completing a Required Action. It is referenced to the time of discovery of a situation (e.g., inoperable equipment or variable not within limits) that requires entering an ACTIONS Condition unless otherwise specified, providing the unit is in a MODE or specified condition stated in the Applicability of the LCO. Required Actions must be completed prior to the expiration of the specified Completion Time. An ACTIONS Condition remains in effect and the Required Actions apply until the Condition no longer exists or the unit is not within the LCO Applicability.

If situations are discovered that require entry into more than one Condition at a time within a single LCO (multiple Conditions), the Required Actions for each Condition must be performed within the associated Completion Time. When in multiple Conditions, separate Completion Times are tracked for each Condition starting from the time of discovery of the situation that required entry into the Condition.

Once a Condition has been entered, subsequent trains, subsystems, components, or variables expressed in the Condition, discovered to be inoperable or not within limits, will not result in separate entry into the Condition, unless specifically stated. The Required Actions of the Condition continue to apply to each additional failure, with Completion Times based on initial entry into the Condition.

(continued)

CTS
NA

## 1.3 Completion Times

DESCRIPTION (continued)

However, when a <u>subsequent</u> train, subsystem, component, or variable, expressed in the Condition, is discovered to be inoperable or not within limits, the Completion Time(s) may be extended. To apply this Completion Time extension, two criteria must first be met. The subsequent inoperability:

a. Must exist concurrent with the <u>first</u> inoperability; and

b. Must remain inoperable or not within limits after the first inoperability is resolved.

The total Completion Time allowed for completing a Required Action to address the subsequent inoperability shall be limited to the more restrictive of either:

a. The stated Completion Time, as measured from the initial entry into the Condition, plus an additional 24 hours; or

b. The stated Completion Time as measured from discovery of the subsequent inoperability.

The above Completion Time extensions do not apply to those Specifications that have exceptions that allow completely separate re-entry into the Condition (for each train, subsystem, component, or variable expressed in the Condition) and separate tracking of Completion Times based on this re-entry. These exceptions are stated in individual Specifications.

The above Completion Time extension does not apply to a Completion Time with a modified "time zero." This modified "time zero" may be expressed as a repetitive time (i.e., "once per 8 hours," where the Completion Time is referenced from a previous completion of the Required Action versus the time of Condition entry) or as a time modified by the phrase "from discovery . . ." Example 1.3-3 illustrates one use of this type of Completion Time. The 10 day Completion Time specified for Conditions A and B in Example 1.3-3 may not be extended.

(continued)

CTS
N/A

## 1.3 Completion Times (continued)

EXAMPLES

The following examples illustrate the use of Completion Times with different types of Conditions and changing Conditions.

EXAMPLE 1.3-1

ACTIONS

| CONDITION | REQUIRED ACTION | COMPLETION TIME |
|---|---|---|
| B. Required Action and associated Completion Time not met. | B.1 Be in MODE 3. | 6 hours |
| | AND | |
| | B.2 Be in MODE 5. | 36 hours |

Condition B has two Required Actions. Each Required Action has its own separate Completion Time. Each Completion Time is referenced to the time that Condition B is entered.

The Required Actions of Condition B are to be in MODE 3 within 6 hours AND in MODE 5 within 36 hours. A total of 6 hours is allowed for reaching MODE 3 and a total of 36 hours (not 42 hours) is allowed for reaching MODE 5 from the time that Condition B was entered. If MODE 3 is reached within 3 hours, the time allowed for reaching MODE 5 is the next 33 hours because the total time allowed for reaching MODE 5 is 36 hours.

If Condition B is entered while in MODE 3, the time allowed for reaching MODE 5 is the next 36 hours.

(continued)

CTS
N/A

## 1.3 Completion Times

EXAMPLES (continued)

EXAMPLE 1.3-2

ACTIONS

| CONDITION | REQUIRED ACTION | COMPLETION TIME |
|---|---|---|
| A. One pump inoperable. | A.1 Restore pump to OPERABLE status. | 7 days |
| B. Required Action and associated Completion Time not met. | B.1 Be in MODE 3.<br>AND<br>B.2 Be in MODE 5. | 6 hours<br><br>36 hours |

When a pump is declared inoperable, Condition A is entered. If the pump is not restored to OPERABLE status within 7 days, Condition B is also entered and the Completion Time clocks for Required Actions B.1 and B.2 start. If the inoperable pump is restored to OPERABLE status after Condition B is entered, Condition A and B are exited, and therefore, the Required Actions of Condition B may be terminated.

When a second pump is declared inoperable while the first pump is still inoperable, Condition A is not re-entered for the second pump. LCO 3.0.3 is entered, since the ACTIONS do not include a Condition for more than one inoperable pump. The Completion Time clock for Condition A does not stop after LCO 3.0.3 is entered, but continues to be tracked from the time Condition A was initially entered.

While in LCO 3.0.3, if one of the inoperable pumps is restored to OPERABLE status and the Completion Time for Condition A has not expired, LCO 3.0.3 may be exited and operation continued in accordance with Condition A.

(continued)

CTS
N/A

## 1.3 Completion Times

EXAMPLES

EXAMPLE 1.3-2 (continued)

While in LCO 3.0.3, if one of the inoperable pumps is restored to OPERABLE status and the Completion Time for Condition A has expired, LCO 3.0.3 may be exited and operation continued in accordance with Condition B. The Completion Time for Condition B is tracked from the time the Condition A Completion Time expired.

On restoring one of the pumps to OPERABLE status, the Condition A Completion Time is not reset, but continues from the time the first pump was declared inoperable. This Completion Time may be extended if the pump restored to OPERABLE status was the first inoperable pump. A 24 hour extension to the stated 7 days is allowed, provided this does not result in the second pump being inoperable for > 7 days.

(continued)

CTS
NA

## 1.3 Completion Times

EXAMPLES (continued)

EXAMPLE 1.3-3

ACTIONS

| CONDITION | REQUIRED ACTION | COMPLETION TIME |
|---|---|---|
| A. One Function X train inoperable. | A.1 Restore Function X train to OPERABLE status. | 7 days<br>AND<br>10 days from discovery of failure to meet the LCO |
| B. One Function Y train inoperable. | B.1 Restore Function Y train to OPERABLE status. | 72 hours<br>AND<br>10 days from discovery of failure to meet the LCO |
| C. One Function X train inoperable.<br>AND<br>One Function Y train inoperable. | C.1 Restore Function X train to OPERABLE status.<br>OR<br>C.2 Restore Function Y train to OPERABLE status. | 72 hours<br><br>72 hours |

(continued)

CTS
NA

## 1.3 Completion Times

EXAMPLES

EXAMPLE 1.3-3 (continued)

When one Function X train and one Function Y train are inoperable, Condition A and Condition B are concurrently applicable. The Completion Times for Condition A and Condition B are tracked separately for each train starting from the time each train was declared inoperable and the Condition was entered. A separate Completion Time is established for Condition C and tracked from the time the second train was declared inoperable (i.e., the time the situation described in Condition C was discovered).

If Required Action C.2 is completed within the specified Completion Time, Conditions B and C are exited. If the Completion Time for Required Action A.1 has not expired, operation may continue in accordance with Condition A. The remaining Completion Time in Condition A is measured from the time the affected train was declared inoperable (i.e., initial entry into Condition A).

The Completion Times of Conditions A and B are modified by a logical connector, with a separate 10 day Completion Time measured from the time it was discovered the LCO was not met. In this example, without the separate Completion Time, it would be possible to alternate between Conditions A, B, and C in such a manner that operation could continue indefinitely without ever restoring systems to meet the LCO. The separate Completion Time modified by the phrase "from discovery of failure to meet the LCO" is designed to prevent indefinite continued operation while not meeting the LCO. This Completion Time allows for an exception to the normal "time zero" for beginning the Completion Time "clock." In this instance, the Completion Time "time zero" is specified as commencing at the time the LCO was initially not met, instead of at the time the associated Condition was entered.

(continued)

CTS
N/A

1.3 Completion Times

EXAMPLES (continued)

EXAMPLE 1.3-4

ACTIONS

| CONDITION | REQUIRED ACTION | COMPLETION TIME |
|---|---|---|
| A. One or more valves inoperable. | A.1 Restore valve(s) to OPERABLE status. | 4 hours |
| B. Required Action and associated Completion Time not met. | B.1 Be in MODE 3.<br>AND<br>B.2 Be in MODE 4. | 6 hours<br><br>12 hours |

A single Completion Time is used for any number of valves inoperable at the same time. The Completion Time associated with Condition A is based on the initial entry into Condition A and is not tracked on a per valve basis. Declaring subsequent valves inoperable, while Condition A is still in effect, does not trigger the tracking of separate Completion Times.

Once one of the valves has been restored to OPERABLE status, the Condition A Completion Time is not reset, but continues from the time the first valve was declared inoperable. The Completion Time may be extended if the valve restored to OPERABLE status was the first inoperable valve. The Condition A Completion Time may be extended for up to 4 hours provided this does not result in any subsequent valve being inoperable for > 4 hours.

If the Completion Time of 4 hours (plus the extension) expires while one or more valves are still inoperable, Condition B is entered.

(continued)

CTS
N/A

## 1.3 Completion Times

EXAMPLES (continued)

EXAMPLE 1.3-5

ACTIONS

---------------------------NOTE---------------------------
Separate Condition entry is allowed for each inoperable valve.
----------------------------------------------------------

| CONDITION | REQUIRED ACTION | COMPLETION TIME |
|---|---|---|
| A. One or more valves inoperable. | A.1 Restore valve to OPERABLE status. | 4 hours |
| B. Required Action and associated Completion Time not met. | B.1 Be in MODE 3.<br>AND<br>B.2 Be in MODE 4. | 6 hours<br><br>12 hours |

The Note above the ACTIONS Table is a method of modifying how the Completion Time is tracked. If this method of modifying how the Completion Time is tracked was applicable only to a specific Condition, the Note would appear in that Condition rather than at the top of the ACTIONS Table.

The Note allows Condition A to be entered separately for each inoperable valve, and Completion Times tracked on a per valve basis. When a valve is declared inoperable, Condition A is entered and its Completion Time starts. If subsequent valves are declared inoperable, Condition A is entered for each valve and separate Completion Times start and are tracked for each valve.

(continued)

CTS
N/A

## 1.3 Completion Times

EXAMPLES

EXAMPLE 1.3-5 (continued)

If the Completion Time associated with a valve in Condition A expires, Condition B is entered for that valve. If the Completion Times associated with subsequent valves in Condition A expire, Condition B is entered separately for each valve and separate Completion Times start and are tracked for each valve. If a valve that caused entry into Condition B is restored to OPERABLE status, Condition B is exited for that valve.

Since the Note in this example allows multiple Condition entry and tracking of separate Completion Times, Completion Time extensions do not apply.

EXAMPLE 1.3-6

ACTIONS

| CONDITION | REQUIRED ACTION | COMPLETION TIME |
|---|---|---|
| A. One channel inoperable. | A.1 Perform SR 3.x.x.x.<br>OR<br>A.2 Reduce THERMAL POWER to ≤ 50% RTP. | Once per 8 hours<br><br>8 hours |
| B. Required Action and associated Completion Time not met. | B.1 Be in MODE 3. | 6 hours |

1.0-05

(continued)

CTS
NA

## 1.3 Completion Times

EXAMPLES

EXAMPLE 1.3-6 (continued)

Entry into Condition A offers a choice between Required Action A.1 or A.2. Required Action A.1 has a "once per" Completion Time, which qualifies for the 25% extension, per SR 3.0.2, to each performance after the initial performance. The initial 8 hour interval of Required Action A.1 begins when Condition A is entered and the initial performance of Required Action A.1 must be complete within the first 8 hour interval. If Required Action A.1 is followed and the Required Action is not met within the Completion Time (plus the extension allowed by SR 3.0.2), Condition B is entered. If Required Action A.2 is followed and the Completion Time of 8 hours is not met, Condition B is entered.

If after entry into Condition B, Required Action A.1 or A.2 is met, Condition B is exited and operation may then continue in Condition A.

(continued)

CTS
N/A

## 1.3 Completion Times

EXAMPLES (continued)

EXAMPLE 1.3-7

ACTIONS

| CONDITION | REQUIRED ACTION | COMPLETION TIME |
|---|---|---|
| A. One subsystem inoperable. | A.1 Verify affected subsystem isolated. | 1 hour<br><br>AND<br><br>Once per 8 hours thereafter |
| | AND<br><br>A.2 Restore subsystem to OPERABLE status. | 72 hours |
| B. Required Action and associated Completion Time not met. | B.1 Be in MODE 3.<br><br>AND<br><br>B.2 Be in MODE 5. | 6 hours<br><br><br><br>36 hours |

Required Action A.1 has two Completion Times. The 1 hour Completion Time begins at the time the Condition is entered and each "Once per 8 hours thereafter" interval begins upon performance of Required Action A.1.

If after Condition A is entered, Required Action A.1 is not met within either the initial 1 hour or any subsequent 8 hour interval from the previous performance (plus the extension allowed by SR 3.0.2), Condition B is entered. The Completion Time clock for Condition A does not stop

(continued)

CTS
N/A

1.3 Completion Times

| | |
|---|---|
| EXAMPLES | EXAMPLE 1.3-7 (continued)<br><br>after Condition B is entered, but continues from the time Condition A was initially entered. If Required Action A.1 is met after Condition B is entered, Condition B is exited and operation may continue in accordance with Condition A, provided the Completion Time for Required Action A.2 has not expired. |
| IMMEDIATE COMPLETION TIME | When "Immediately" is used as a Completion Time, the Required Action should be pursued without delay and in a controlled manner. |

CTS

N/A

1.0 USE AND APPLICATION

1.4 Frequency

| | |
|---|---|
| PURPOSE | The purpose of this section is to define the proper use and application of Frequency requirements. |
| DESCRIPTION | Each Surveillance Requirement (SR) has a specified Frequency in which the Surveillance must be met in order to meet the associated LCO. An understanding of the correct application of the specified Frequency is necessary for compliance with the SR.<br><br>The "specified Frequency" is referred to throughout this section and each of the Specifications of Section 3.0, Surveillance Requirement (SR) Applicability. The "specified Frequency" consists of the requirements of the Frequency column of each SR, as well as certain Notes in the Surveillance column that modify performance requirements.<br><br>~~Situations where a Surveillance could be required (i.e., its Frequency could expire), but where it is not possible or not desired that it be performed until sometime after the associated LCO is within its Applicability, represent potential SR 3.0.4 conflicts. To avoid these conflicts, the SR (i.e., the Surveillance or the Frequency) is stated such that it is only "required" when it can be and should be performed. With an SR satisfied, SR 3.0.4 imposes no restriction.~~ 22 |
| EXAMPLES | The following examples illustrate the various ways that Frequencies are specified. In these examples, the Applicability of the LCO (LCO not shown) is MODES 1, 2, and 3. |

AND-242

<INSERT 1.4-1>

(continued)

<INSERT 1.4-1>

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Sometimes special situations dictate when the requirements of a Surveillance are to be met. They are "otherwise stated" conditions allowed by SR 3.0.1. They may be stated as clarifying Notes in the Surveillance, as part of the Surveillances, or both.

Situations where a Surveillance could be required (i.e., its Frequency could expire), but where it is not possible or not desired that it be preformed until sometime after the associated LCO is within its Applicability, represent potential SR 3.0.4 conflicts. To avoid these conflicts, the SR (i.e., the Surveillance or the Frequency) is stated such that it is only "required" when it can be and should be performed. With an SR satisfied, SR3.0.4 imposes no restriction.

The use of "met" or "performed" in these instances conveys specific meanings. A Surveillance is "met" only when the acceptance criteria are satisfied. Known failure of the requirements of a Surveillance, even without a Surveillance specifically being "performed," constitutes a Surveillance not "met." "Performance" refers only to the requirement to specifically determine the ability to meet the acceptance criteria.

Some Surveillances contain notes that modify the Frequency of performance or the conditions during which the acceptance criteria must be satisfied. For these Surveillances, the MODE-entry restrictions of SR 3.0.4 may not apply. Such a Surveillance is not required to be performed prior to entering a MODE or other specified condition in the Applicability of the associated LCO if any of the following three conditions are satisfied:

a. The Surveillance is not required to be met in the MODE or other specified condition to be entered: or

b. The Surveillance is required to be met in the MODE or other specified condition to be entered, but has been performed within the specified Frequency (i.e., it is current) and is known not to be failed; or

c. The Surveillance is required to be met, but not performed, in the MODE or other specified condition to be entered, and is known no to be failed.

Examples 1.4-3, 1.4-4, 1.4-5, and 1.4-6 discuss these special situations.

CTS
N/A

## 1.4 Frequency

EXAMPLES (continued)

EXAMPLE 1.4-1

SURVEILLANCE REQUIREMENTS

| SURVEILLANCE | FREQUENCY |
|---|---|
| Perform CHANNEL CHECK. | 12 hours |

Example 1.4-1 contains the type of SR most often encountered in the Technical Specifications (TS). The Frequency specifies an interval (12 hours) during which the associated Surveillance must be performed at least one time. Performance of the Surveillance initiates the subsequent interval. Although the Frequency is stated as 12 hours, an extension of the time interval to 1.25 times the stated Frequency is allowed by SR 3.0.2 for operational flexibility. The measurement of this interval continues at all times, even when the SR is not required to be met per SR 3.0.1 (such as when the equipment is inoperable, a variable is outside specified limits, or the unit is outside the Applicability of the LCO). If the interval specified by SR 3.0.2 is exceeded while the unit is in a MODE or other specified condition in the Applicability of the LCO, and the performance of the Surveillance is not otherwise modified (refer to Example 1.4-3), then SR 3.0.3 becomes applicable.

If the interval as specified by SR 3.0.2 is exceeded while the unit is not in a MODE or other specified condition in the Applicability of the LCO for which performance of the SR is required, the Surveillance must be performed within the Frequency requirements of SR 3.0.2 prior to entry into the MODE or other specified condition. Failure to do so would result in a violation of SR 3.0.4.

(continued)

CTS
N/A

## 1.4 Frequency

EXAMPLES (continued)

EXAMPLE 1.4-2

SURVEILLANCE REQUIREMENTS

| SURVEILLANCE | FREQUENCY |
|---|---|
| Verify flow is within limits. | Once within 12 hours after ≥ 25% RTP<br><br>AND<br><br>24 hours thereafter |

Example 1.4-2 has two Frequencies. The first is a one time performance Frequency, and the second is of the type shown in Example 1.4-1. The logical connector "AND" indicates that both Frequency requirements must be met. Each time reactor power is increased from a power level < 25% RTP to ≥ 25% RTP, the Surveillance must be performed within 12 hours.

The use of "once" indicates a single performance will satisfy the specified Frequency (assuming no other Frequencies are connected by "AND"). This type of Frequency does not qualify for the extension allowed by SR 3.0.2. "Thereafter" indicates future performances must be established per SR 3.0.2, but only after a specified condition is first met (i.e., the "once" performance in this example). If reactor power decreases to < 25% RTP, the measurement of both intervals stops. New intervals start upon reactor power reaching 25% RTP.

(continued)

CTS
N/A

## 1.4 Frequency

EXAMPLES (continued)

EXAMPLE 1.4-3

SURVEILLANCE REQUIREMENTS

| SURVEILLANCE | FREQUENCY |
|---|---|
| ------------------NOTE------------------<br>Not required to be performed until 12 hours after ≥ 25% RTP.<br>----------------------------------------<br><br>Perform channel adjustment. | 7 days |

The interval continues whether or not the unit operation is < 25% RTP between performances.

As the Note modifies the required performance of the Surveillance, it is construed to be part of the "specified Frequency." Should the 7 day interval be exceeded while operation is < 25% RTP, this Note allows 12 hours after power reaches ≥ 25% RTP to perform the Surveillance. The Surveillance is still considered to be performed within the "specified Frequency." Therefore, if the Surveillance were not performed within the 7 day (plus the extension allowed by SR 3.0.2) interval, but operation was < 25% RTP, it would not constitute a failure of the SR or failure to meet the LCO. Also, no violation of SR 3.0.4 occurs when changing MODES, even with the 7 day Frequency not met, provided operation does not exceed 12 hours with power ≥ 25% RTP.

Once the unit reaches 25% RTP, 12 hours would be allowed for completing the Surveillance. If the Surveillance were not performed within this 12 hour interval, there would then be a failure to perform a Surveillance within the specified Frequency, and the provisions of SR 3.0.3 would apply. 22

<INSERT 1.4-4A>
<INSERT 1.4-4B>
<INSERT 1.4-4C>

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<INSERT 1.4-4A>

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EXAMPLE 1.4-4

SURVEILLANCE REQUIREMENTS

| SURVEILLANCE | FREQUENCY |
|---|---|
| -----------------------NOTE-----------------------<br>Only required to be met in MODE 1<br>--------------------------------------------------<br><br>Verify leakage rates are within limits. | 24 hours |

Example 1.4-4 specifies that the requirements of this Surveillance do not have to be met until the unit is in MODE 1. The interval measurement for the Frequency of this Surveillance continues at all times, as described in Example 1.4-1. However, the Note constitutes an "otherwise stated" exception to the Applicability of this Surveillance. Therefore, if the Surveillance were not performed within the 24 hour interval (plus the extension allowed by SR 3.0.2), but the unit was not in MODE 1, there would be no failure of the SR nor failure to meet the LCO. Therefore, no violation of SR 3.0.4 occurs when changing MODES, even with the 24 hour Frequency exceeded, provided the MODE change was not made into MODE 1. Prior to entering MODE 1 (assuming again that the 24 hour Frequency were not met), SR 3.0.4 would require satisfying the SR.

<INSERT 1.4-4B>

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EXAMPLE 1.4-5

SURVEILLANCE REQUIREMENTS

| SURVEILLANCE | FREQUENCY |
|---|---|
| ---------- NOTE ---------- Only required to be performed in MODE 1 | |
| Perform complete cycle of the valve. | 7 days |

The interval continues, whether or not the unit operation is in MODE 1,2 or 3 (the assumed Applicability of the associated LCO) between performances.

As the Note modifies the required performance of the Surveillance, the Note is construed to be part of the "specified Frequency." Should the 7 day interval be exceeded while operation is not in MODE 1, this Note allows entry into and operation in MODES 2 and 3 to perform the Surveillance. The Surveillance is still considered to be performed within the "specified Frequency" if completed prior to entering MODE 1. Therefore, if the Surveillance were not performed within the 7 day (plus the extension allowed by SR 3.0.2) interval, but operation was not in MODE 1, it would not constitute a failure of the SR or failure to meet the LCO. Also, no violation of SR 3.0.4 occurs when changing MODES, even with the 7 day Frequency not met, provided operation does not result in entry into MODE 1.

Once the unit reaches MODE 1, the requirement for the Surveillance to be performed within its specified Frequency applies and would require that the Surveillance had been performed. If the Surveillance were not performed prior to entering MODE 1, there would then be a failure to perform a Surveillance within the specified Frequency, and the provisions of SR 3.0.3 would apply.

<INSERT 1.4-4C>

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EXAMPLE 1.4-6

SURVEILLANCE REQUIREMENTS

| SURVEILLANCE | FREQUENCY |
|---|---|
| -----NOTE----- Not required to be met in MODE 3 ----- | |
| Verify parameter is within limits. | 24 hours |

Example 1.4-6 specifies that the requirements of this Surveillance do not have to be met while the unit is in MODE 3 (the assumed Applicability of the associated LCO is MODES 1,2, and 3). The interval measurement for the Frequency of this Surveillance continues at all times, as described in Example 1.4-1. However, the Note constitutes an "otherwise stated" exception to the Applicability of this Surveillance. Therefore, if the Surveillance were not performed within the 24 hour interval (plus the extension allowed by SR 3.0.2), and the unit was in MODE 3, there would be no failure of the SR nor failure to meet the LCO. Therefore, no violation of SR 3.0.4 occurs when changing MODES to enter MODE 3, even with the 24 hour Frequency exceeded, provided the MODE change does not result in entry into MODE 2. Prior to entering MODE 2 (assuming again that the 24 hour Frequency were not met), SR 3.0.4 would require satisfying the SR.

## 2.0 SAFETY LIMITS (SLs)

### 2.1 SLs

#### 2.1.1 Reactor Core SLs

2.1.1.1 In MODES 1 and 2, the maximum local fuel pin centerline temperature shall be ≤ 5080 - (6.5 x $10^{-3}$ x (Burnup, MWD/MTU)°F) for TACO2 applications and ≤ 4642 - (5.8 x $10^{-3}$ x (Burnup, MWD/MTU)°F) for TACO 3 applications.

2.1.1.2 In MODES 1 and 2, the departure from nucleate boiling ratio shall be maintained greater than the limits of 1.3 for the BAW-2 correlation and 1.18 for the BWC correlation.

2.1.1.3 In MODES 1 and 2, Reactor Coolant System (RCS) core outlet temperature and pressure shall be maintained above and to the left of the Variable Low RCS Pressure-Temperature Protective Limits as specified in the Core Operating Limits Report, so that the safety limits are met.

#### 2.1.2 RCS Pressure SL

In MODES 1, 2, 3, 4, and 5, the RCS pressure shall be maintained ≤ 2750 psig.

### 2.2 SL Violations

With any SL violation, the following actions shall be completed:

2.2.1 In MODE 1 or 2, if SL 2.1.1.1 or SL 2.1.1.2 is violated, be in MODE 3 within 1 hour.

2.2.2 In MODE 1 or 2, if SL 2.1.1.3 is violated, restore RCS pressure and temperature within limits AND be in MODE 3 within 1 hour.

2.2.3 In MODE 1 or 2, if SL 2.1.2 is violated, restore compliance within limits AND be in MODE 3 within 1 hour.

2.2.4 In MODES 3, 4, and 5, if SL 2.1.2 is violated, restore RCS pressure to ≤ 2750 psig within 5 minutes.

2.2.5 Within 1 hour, notify the NRC Operations Center, in accordance with 10 CFR 50.72.

# B 2.0 SAFETY LIMITS (SLS)

## B 2.1.1 Reactor Core SLs

BASES

### BACKGROUND

GDC 10 (Ref. 1) requires that reactor core SLs ensure specified acceptable fuel design limits are not exceeded during steady state operation, normal operational transients, and abnormalities. This is accomplished by having a departure from nucleate boiling (DNB) design basis, which corresponds to a 95% probability at a 95% confidence level (95/95 DNB criterion) that DNB will not occur and by requiring that the fuel centerline temperature stays below the melting temperature.

Although DNB is not an observable parameter during reactor operation, the observable parameters of neutron power, reactor coolant flow, temperature and pressure can be related to DNB through the use of a critical heat flux (CHF) correlation. The BAW-2 (Ref. 2) and BWC (Ref. 3) correlations have been developed to predict DNB and the location of DNB for axially uniform and non-uniform heat flux distributions. The BAW-2 correlation applies to Mark-B fuel and the BWC correlation applies to Mark-BZ fuel. The local DNB ratio (DNBR), defined as the ratio of the heat flux that would cause DNB at a particular core location to the actual heat flux, is indicative of the margin to DNB. The minimum value of the DNBR, during steady state operation, normal operational transients and anticipated transients is limited to 1.30 (BAW-2) and 1.18 (BWC).

The 95 percent confidence level that DNB will not occur is preserved by ensuring that the DNBR remains greater than the DNBR design limit based on the applicable CHF correlation for the core design. In the development of the applicable DNBR design limit, uncertainties in the core state variables, power peaking factors, manufacturing-related parameters, and the CHF correlation may be statistically combined to determine a statistical DNBR design limit. This statistical design limit protects the respective CHF design limit. Additional retained thermal margin may also be applied to the statistical DNBR design limit to yield a higher thermal design limit for use in establishing DNB-based core safety and operating limits. In all cases, application of statistical DNB design methods preserves a 95 percent probability at a 95 percent confidence level that DNB will not occur (Ref. 4).

The restrictions of this SL prevent overheating of the fuel and cladding and possible cladding perforation that would result in the release of fission products to the reactor coolant. Overheating of the fuel is prevented by maintaining the steady state peak linear heat rate (LHR) below the level at which fuel centerline melting occurs. The maximum fuel centerline temperatures are given by the relationships defined in SL 2.1.1.1 for the respective fuel designs and are dependent on whether the TACO2 (Ref. 5) or TACO3 (Ref. 6) analysis was utilized. Overheating of the fuel cladding is prevented by restricting fuel operation to within the nucleate boiling regime, where the heat transfer coefficient is large and the cladding surface temperature is slightly above the coolant saturation temperature.

Fuel centerline melting occurs when the local LHR, or power peaking, in a region of the fuel is high enough to cause the fuel centerline temperature to reach the melting point of the fuel. Expansion of the pellet upon centerline melting may cause the pellet to stress the cladding to the point of failure, allowing an uncontrolled release of activity to the reactor coolant.

Operation above the boundary of the nucleate boiling regime could result in excessive cladding temperature because of the onset of DNB and the resultant sharp reduction in heat transfer coefficient. Inside the steam film, high cladding temperatures are reached, and a cladding-water (zirconium-water) reaction may take place. This chemical reaction results in oxidation of the fuel cladding. The oxidized cladding then exists in a structurally weaker form. This weaker form may lose its integrity, resulting in an uncontrolled release of activity to the reactor coolant.

The proper functioning of the Reactor Protection System (RPS) prevents violation of the reactor core SLs.

---

APPLICABLE SAFETY ANALYSES

The fuel cladding must not sustain damage as a result of normal operation and abnormalities. The reactor core SLs are established to preclude violation of the following fuel design criteria:

a. There must be at least 95% probability at a 95% confidence level (95/95 DNB criterion) that the hot fuel rod in the core does not experience DNB; and

b. The hot fuel pellet in the core must not experience fuel centerline melting.

The RPS setpoints, in combination with all the LCOs, are designed to prevent any analyzed combination of transient conditions for Reactor Coolant System (RCS) temperature, pressure, and THERMAL POWER level that would result in a DNBR of less than the DNBR limit and preclude the existence of flow instabilities (Ref. 7).

Automatic enforcement of these reactor core SLs is provided by the following:

a. RCS High Pressure trip;

b. RCS Low Pressure trip;

c. Nuclear Overpower trip;

d. RCS Variable Low Pressure trip (also known as Pressure Temperature Trip);

e. Reactor Coolant Pump to Power trip;

f. Nuclear Overpower RCS Flow and AXIAL POWER IMBALANCE trip; and

g. RCS High Temperature trip.

The SL represents a design requirement for establishing the RPS trip setpoints identified previously.

---

SAFETY LIMITS

SL 2.1.1.1, SL 2.1.1.2, and SL 2.1.1.3 ensure that the minimum DNBR is not less than the safety analyses limit and that fuel centerline temperature stays below the melting point, or the average enthalpy in the hot leg is less than or equal to the enthalpy of saturated liquid, or the exit quality is within the limits defined by the DNBR correlation. In addition, the COLR identifies the pressure/temperature operating region that keeps the reactor from reaching an SL when operating up to design power.

The COLR presents the most limiting condition of pressure/temperature combinations for all possible reactor coolant pump maximum THERMAL POWER combinations. Analyses have been performed which bound the three pump and two pump (one pump in each loop) allowed operating conditions based on the expected minimum flow rates and maximum ALLOWABLE THERMAL POWER for these operating conditions.

The SLs are preserved by monitoring the process variable AXIAL POWER IMBALANCE to ensure that the core operates within the fuel design criteria. AXIAL POWER IMBALANCE protective limits are preserved by their corresponding RPS setpoints in LCO 3.3.1, "Reactor Protection System (RPS) Instrumentation," and are provided in the COLR. The trip setpoints are derived by adjusting the measurement system independent AXIAL POWER IMBALANCE protective limits given in the COLR to allow for measurement system observability and instrumentation errors.

The AXIAL POWER IMBALANCE protective limits are separate and distinct from the AXIAL POWER IMBALANCE operating limits defined by LCO 3.2.3, "AXIAL POWER IMBALANCE Operating Limits." The AXIAL POWER IMBALANCE operating limits in LCO 3.2.3, also specified in the COLR, preserve initial conditions of the safety analyses but are not reactor core SLs.

---

APPLICABILITY

SL 2.1.1.1, SL 2.1.1.2, and SL 2.1.1.3 only apply in MODES 1 and 2 because these are the only MODES in which the reactor is critical. Automatic protection functions are required to be OPERABLE during MODES 1 and 2 to ensure operation within the reactor core SLs. Automatic protection actions serve to prevent RCS heatup to reactor core SL conditions or to initiate a reactor trip function, which forces the unit into MODE 3. Setpoints for the reactor trip functions are specified in LCO 3.3.1.

In MODES 3, 4, 5, and 6, Applicability is not required, since the reactor is not generating significant THERMAL POWER

## SAFETY LIMIT VIOLATIONS

The following SL violation responses are applicable to the reactor core SLs.

2.2.1 AND 2.2.2

If SL 2.1.1.1, SL 2.1.1.2, or SL 2.1.1.3 is violated, the requirement to go to MODE 3 places the plant in a MODE in which these SLs are not applicable.

The allowed Completion Time of 1 hour recognizes the importance of bringing the plant to a MODE of operation where these SLs are not applicable and reduces the probability of fuel damage.

2.2.5

If SL 2.1.1.1, SL 2.1.1.2, or SL 2.1.1.3 is violated, the NRC Operations Center must be notified within 1 hour, in accordance with 10 CFR 50.72 (Ref. 8).

## REFERENCES

1. SAR, Section 1.4, GDC 10.

2. BAW-10000A, "Correlation of Critical Heat Flux in a Bundle Cooled by Pressurized Water," Babcock & Wilcox, Lynchburg, VA, May 1976 .

3. BAW-10143P-A, "BWC Correlation of Critical Heat Flux," Babcock & Wilcox, Lynchburg, VA, April 1985.

4. BAW-10179P-A, "Safety Criteria and Methodology for Acceptable Cycle Reload Analyses," Rev. 2, Babcock & Wilcox, Lynchburg, VA, October 1997.

5. BAW-10141P-A, Rev. 1, "TACO2 Fuel Pin Performance Analysis," Babcock & Wilcox, Lynchburg, VA, June 1983.

5. BAW-10162P-A, "TACO3 Fuel Pin Thermal Analysis Code," Babcock & Wilcox, Lynchburg, VA, October 1989.

7. SAR, Chapters 3 & 14.

8. 10 CFR 50.72.

# B 2.0 SAFETY LIMITS (SLS)

## B 2.1.2 Reactor Coolant System (RCS) Pressure SL

BASES

---

### BACKGROUND

In SAR, Section 1.4 (Ref. 1), GDC 14, "Reactor Coolant Pressure Boundary (RCPB)," and GDC 15, "Reactor Coolant System Design", address RCPB design and protection, respectively. The ANO-1 discussion regarding how GDC 15 is accomplished states that analysis and evaluation of all normal and abnormal operating conditions and transients are integrally related to all RCS and associated systems design. SAR Chapter 14 (Ref. 2) lists these abnormal operating conditions and transients and terms them "abnormalities". In addition, GDC 28, "Reactivity Limits" (Ref. 1), specifies that reactivity accidents including rod ejection do not result in damage to the RCPB greater than limited local yielding.

The design pressure of the RCS is 2500 psig. During normal operation and abnormalities, the RCS pressure is kept from exceeding the design pressure by more than 10% in order to remain in accordance with the design codes (Ref. 3 and 4). Hence, the safety limit is 2750 psig. To ensure system integrity, all RCS components were hydrostatically tested at 125% of design pressure prior to initial operation, according to the design code requirements. Inservice leak testing at not less than 2155 psig is also required, prior to MODE 2, following any opening of the reactor coolant system in accordance with ASME code, Section XI; IWA-5000. When performed at the end of refueling outages, this leak test also satisfies the requirements of IWB-2500, Table IWB-2500-1; Category B-P items B15.10, B15.20, B15.30, B15.40, B15.50, B15.60, and B15.70 for all Class I pressure retaining components (Ref. 5).

---

### APPLICABLE SAFETY ANALYSES

The RCS pressurizer safety valves, operating in conjunction with the Reactor Protection System trip settings, ensure that the RCS pressure SL will not be exceeded.

The RCS pressurizer safety valves are sized to prevent system pressure from exceeding the design pressure by more than 10%, in accordance with Section III of the ASME code for Nuclear Power Plant Components (Ref. 3). The design basis transient that is most influential for establishing the required relief capacity, and hence the valve size requirements and lift settings, is a rod withdrawal event from low power.

The startup event analysis (rod withdrawal at low power) (Ref. 2) is performed using conservative assumptions relative to pressure control devices.

More specifically, no credit is taken for operation of the following:

a. Electromatic relief valve (ERV);

b. Steam line turbine bypass valves;

c. Control system runback of reactor and turbine power; and

d. Pressurizer spray valve.

---

## SAFETY LIMITS

The maximum transient pressure allowed in the RCS pressure vessel under the ASME code, Section III, is 110% of design pressure. The maximum transient pressure allowed in the RCS piping, valves, and fittings under USAS B31.7 (Ref. 4), is 110% of design pressure. Therefore, the SL on maximum allowable RCS pressure is 2750 psig.

Overpressurization of the RCS can result in a breach of the RCPB. If such a breach occurs in conjunction with a fuel cladding failure, fission products could enter the containment atmosphere, raising concerns relative to limits on radioactive releases specified in 10 CFR 100, "Reactor Site Criteria" (Ref. 6).

---

## APPLICABILITY

SL 2.1.2 applies in MODES 1, 2, 3, 4, and 5 because this SL could be approached or exceeded in these MODES during overpressurization events. The SL is not applicable in MODE 6 because the reactor vessel head closure bolts are not fully tightened, making it unlikely that the RCS can be pressurized significantly.

---

## SAFETY LIMIT VIOLATIONS

The following SL violation responses are applicable to the RCS pressure SL.

<u>2.2.3</u>

If the RCS pressure SL is violated when the reactor is in MODE 1 or 2, the requirement is to restore compliance and be in MODE 3 within 1 hour.

Exceeding the RCS pressure SL may cause immediate RCS failure and create a potential for radioactive releases in excess of 10 CFR 100, "Reactor Site Criteria," limits (Ref. 6).

The allowed Completion Time of 1 hour is based on the importance of reducing power level to a MODE where the potential for challenges to safety systems is minimized.

2.2.4

If the RCS pressure SL is exceeded in MODE 3, 4, or 5, RCS pressure must be restored to within the SL value within 5 minutes.

Exceeding the RCS pressure SL in MODE 3, 4, or 5 is potentially more severe than exceeding this SL in MODE 1 or 2, since the reactor vessel temperature may be lower and the vessel material, consequently, less ductile. As such, pressure must be reduced to less than the SL within 5 minutes. This action does not require reducing MODES, since this would require reducing temperature, which would compound the problem by adding thermal gradient stresses to the existing pressure stress.

2.2.5

If the RCS pressure SL is violated, the NRC Operations Center must be notified within 1 hour, in accordance with 10 CFR 50.72 (Ref. 7).

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REFERENCES

1. SAR, Section 1.4, GDC 14, GDC 15, and GDC 28, 1988.
2. SAR, Chapter 14.
3. ASME Boiler and Pressure Vessel Code, Section III, 1965-S67, Article NB-7000.
4. USAS B31.7, Nuclear Power Piping, 1969.
5. ASME Boiler and Pressure Vessel Code, Section XI, Article IW-5000.
6. 10 CFR 100.
7. 10 CFR 50.72.

# CTS DISCUSSION OF CHANGES

## ITS Section 2.0: Safety Limits

**ADMINISTRATIVE**

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

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

A3 The requirement of CTS 6.7.1.b. to submit a report to the NRC "pursuant to the requirements of 10 CFR 50.36" was removed. This requirement is a duplication of the requirement found in 10 CFR 50.36 "Technical Specifications" paragraph (c)(1) and as such was redundant. The removal of this requirement from the CTS was administrative in nature because this requirement was contained elsewhere, namely 10 CFR 50.36.

2.0-01

A4 Not used.

**TECHNICAL CHANGE -- MORE RESTRICTIVE**

M1 CTS 2.1.1, 2.1.2, & 2.1.3 establish the APPLICABILITY for the Reactor Core Safety Limits as "when the reactor is critical." ITS 2.1.1 will establish APPLICABILITY as MODES 1 and 2 which include Keff greater than or equal to 0.99. Thus, MODE 2 is more restrictive than CTS since it does not become applicable until Keff = 1.0. The additional Applicability is included because limiting accidents and transients are postulated which begin in this MODE. This requirement is consistent with NUREG-1430.

M2 CTS 2.2 does not establish required actions should the RCS Pressure Safety Limit be violated in MODES 3, 4, and 5. Therefore, the required actions of RSTS 2.2.4 are adopted in the ITS. The information shown as inserted on the CTS mark-up as ITS 2.2.4 represents more restrictive requirements than those presently imposed.

M3 CTS 6.7.1.a required that the Unit be placed in hot shutdown within one hour following the violation of a CTS defined Safety Limit. ITS 2.2.1, 2.2.2, and 2.2.3 will require that the Unit be placed in MODE 3. The ITS requirement is more restrictive in that it will require that the Unit have a Keff value of less than 0.99. The CTS requires that the Unit be taken 1% $\Delta k/k$ subcritical. The Keff requirement is 0.01% $\Delta k/k$ more restrictive.

# CTS DISCUSSION OF CHANGES

## TECHNICAL CHANGE -- LESS RESTRICTIVE

L1 CTS 2.2.1 establishes APPLICABILITY for the RCS Pressure Safety Limit as being "when there are fuel assemblies in the reactor vessel." ITS 2.1.2 will establish APPLICABILITY as MODES 1, 2, 3, 4 & 5. In essence, the ITS would be marginally less restrictive as it would not apply during MODE 6 while the CTS would apply after the first assembly was placed in the vessel. Although a short time period may exist between MODE 5 and reactor vessel head removal in MODE 6, during which the Safety Limit will no longer apply, the consequences of a postulated overpressure event are mitigated by the implementation of low temperature overpressurization protection requirements and administrative controls.

## LESS RESTRICTIVE -- ADMINISTRATIVE DELETION OF REQUIREMENTS

LA1 This information has been moved to the Bases. This information provides details of design or process which are not directly pertinent to the actual requirement, i.e., Safety Limit, but rather describe an acceptable method of compliance. Since these details are not necessary to adequately describe the actual regulatory requirement, they can be moved to a licensee controlled document without a significant impact on safety. Placing these details in controlled documents provides adequate assurance that they will be maintained. The Bases will be controlled by the Bases Control Process in Chapter 5 of the proposed Technical Specifications. The specific relocations are:

| CTS Location | New Location |
|---|---|
| 2nd sentence of SL 2.1.1 | B 2.1.1 Applicable Safety Analyses |
| 2nd sentence of SL 2.1.2 | B 2.1.1 Applicable Safety Analyses |
| 4.3.2 | B 2.1.2 Background |

2. SAFETY LIMITS AND LIMITING SAFETY SYSTEM SETTINGS A1

2.1 SAFETY LIMITS, REACTOR CORE

Applicability

2.1 APPL Applies to reactor thermal power, reactor power imbalance, reactor coolant system pressure, coolant temperature, and coolant flow when the reactor is critical. M1

MODES 1 and 2

Objective

To maintain the integrity of the fuel cladding. A1

Specification

2.1.1.1 2.1.1 The maximum local fuel pin centerline temperature shall be $\leq 5080 - (6.5 \times 10^{-3} \times$ (Burnup, MWD/MTU) °F) for TACO2 applications and $\leq 4642 - (5.8 \times 10^{-3} \times$ (Burnup, MWD/MTU) °F) for TACO3 applications. edit Operation within this limit is ensured by compliance with the Axial Power Imbalance protective limits preserved by Table 2.3-1 "Reactor Protection System Trip Setting Limits," as specified in the COLR. LA1 Bases

2.1.1.2 2.1.2 The departure from nucleate boiling ratio shall be maintained greater than the limits of 1.3 for the BAW-2 correlation and 1.18 for the BWC correlation. Operation within this limit is ensured by compliance with Specification 2.1.3 and with the Axial Power Imbalance protective limits preserved by Table 2.3-1 "Reactor Protection System Trip Setting Limits," as specified in the COLR. LA1 Bases

2.1.1.3 2.1.3 Reactor Coolant System (RCS) core outlet temperature and pressure shall be maintained above and to the left of the Variable Low RCS Pressure-Temperature Protective Limits as specified in the COLR.

Bases

To maintain the integrity of the fuel cladding and to prevent fission product release, it is necessary to prevent overheating of the cladding under normal operating conditions. This is accomplished by operating within the nucleate boiling regime of heat transfer, wherein the heat transfer coefficient is large enough so that the clad surface temperature is only slightly greater than the coolant temperature. The upper boundary of the nucleate boiling regime is termed departure from nucleate boiling (DNB). At this point there is a sharp reduction of the heat transfer coefficient, which could result in high cladding temperatures and the possibility of cladding failure. Although DNB is not an observable parameter during reactor operation, the observable parameters of neutron power, reactor coolant flow, temperature, and pressure can be related to DNB through the use of a critical heat flux (CHF) correlation. The BAW-2(1) and BWC(2) correlations have been developed to predict DNB and the location of DNB for axially uniform and non-uniform heat flux distributions. The BAW-2 correlation applies to Mark-B fuel and the BWC correlation applies to Mark-BZ fuel. The local DNB ratio (DNBR), defined as the ratio of the heat flux that would cause DNB at a particular core location to the actual heat flux, is indicative of the margin to DNB. The minimum value of the DNBR, during steady-state operation, normal operational transients, and anticipated transients is limited to 1.30 (BAW-2) and 1.18 (BWC). A2

2.0

A2

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

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

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

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

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

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

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

2.0

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

AZ

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

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

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

REFERENCES


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

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

(3) FSAR, Section 3.2.3.1.1.c.

2.0

(SLs)

## 2.2 SAFETY LIMITS - REACTOR SYSTEM PRESSURE A1

~~Applicability~~

~~Applies to the limit on reactor coolant system pressure.~~

~~Objective~~

~~To maintain the integrity of the reactor coolant system and to prevent the release of significant amounts of fission product activity.~~ A1

Specification

2.1.2 2.2.1 The reactor coolant system pressure shall not exceed 2750 psig ~~when there are fuel assemblies in the reactor vessel.~~ L1

2.1.2 Appl MODES 1, 2, 3, 4, and 5

<LATER> (3.4B) ~~2.2.2 The setpoint of the pressurizer code safety valves shall be in accordance with ASME, Boiler and Pressurizer Vessel Code, Section III, Article 9, Summer 1968.~~ LATER

~~Bases~~

~~The reactor coolant system (1) serves as a barrier to prevent radionuclides in the reactor coolant from reaching the atmosphere. In the event of a fuel cladding failure, the reactor coolant system is a barrier against the release of fission products. Establishing a system pressure limit helps to assure the integrity of the reactor coolant system. The maximum transient pressure allowable in the reactor coolant system pressure vessel under the ASME code, Section III, is 110 percent of design pressure.(2) The maximum transient pressure allowable in the reactor coolant system piping, valves, and fittings under ANSI Section B31.7 is 110 percent of design pressure. Thus, the safety limit of 2750 psig (110 percent of the 2500 psig design pressure) has been established.(2) The settings for the reactor high pressure trip (2355 psig) and the pressurizer code safety valves (2500 psig ±1%)(3) have been established to assure that the reactor coolant system pressure safety limit is not exceeded. When testing the pressurizer code safety valves, the "as found" lift setpoint may be 2500 psig +1, -5%. However, if found outside of a ±1% tolerance band, they shall be reset to 2500 psig ±1%. The initial hydrostatic test is conducted at 3125 psig (125 percent of design pressure) to verify the integrity of the reactor coolant system. Additional assurance that the reactor coolant system pressure does not exceed the safety limit is provided by setting the pressurizer electromatic relief valve at 2450 psig.(4)~~ A2

~~REFERENCES~~

~~(1) FSAR, Section 4~~

~~(2) FSAR, Section 4.3.11.1~~

~~(3) FSAR, Section 4.2.4~~

~~(4) FSAR, Table 4-1~~

4.3 TESTING FOLLOWING OPENING OF SYSTEM

<u>Applicability</u>

Applies to test requirements for Reactor Coolant System integrity.

<u>Objective</u>

To assure Reactor Coolant System integrity prior to return to criticality following normal opening, modification, or repair.

<u>Specification</u>

4.3.1 When Reactor Coolant System repairs or modifications have been made, these repairs or modifications shall be inspected and tested to meet all applicable code requirements prior to the reactor being made critical.

4.3.2 Following any opening of the Reactor Coolant System, it shall be leak tested at not less than 2155 psig, prior to the reactor being made critical, in accordance with the ASME Boiler and Pressure Vessel Codes, Section XI; IWA-5000.

4.3.3 The limitations of Specification 3.1.2 shall apply.

<LATER> (5.0)

LATER

LA1 Bases

<LATER> (5.0)

LATER

<u>Bases</u>

Repairs or modifications made to the Reactor Coolant System are inspectable and testable under applicable codes, such as B 31.7, and ASME Boiler and Pressure Vessel Code, Section XI.

For normal opening, the integrity of the Reactor Coolant System in terms of strength, is unchanged. The ASME Boiler and Pressure Vessel Code, Section XI; IWA-5000 requires a system leak test at nominal operating pressure (2155 psig) following system opening. At the end of refueling outages, this test also satisfies the requirements of IWB-2500, Table IWB-2500-1; Category B-P items B15.10, B15.20, B15.30, B15.40, B15.50, B15.60, and B15.70 for all Class I pressure retaining components.

A2

<u>REFERENCES</u>

(1) FSAR, Section 4
(2) ASME Boiler and Pressure Vessel Code, Section XI

2.0

6.6 DELETED

SL Violations — A1

2.2 6.7 ~~SAFETY LIMIT VIOLATION~~

6.7.1 The following actions shall be taken in the event a Safety Limit is violated:

2.2.1, 2.2.2, 2.2.3 a. The facility shall be placed in at least MODE 3 ~~hot shutdown~~ within ~~one hour.~~ Within 1 hour — M3 — A1

2.2.5 b. The Nuclear Regulatory Commission shall be notified pursuant to 10 CFR 50.72 ~~and a report submitted pursuant to the requirements of 10 CFR 50.36 and Specification 6.6.~~ — A3

~~6.8 PROCEDURES AND PROGRAMS~~

<LATER> (5.0)

~~6.8.1 Written procedures shall be established, implemented and maintained covering the activities referenced below:~~

~~a. The applicable procedures recommended in Appendix "A" of Regulatory Guide 1.33, November, 1972.~~

~~b. Refueling operations.~~

~~c. Surveillance and test activities of safety related equipment.~~

~~d. (Deleted)~~

~~e. (Deleted)~~

~~f. Fire Protection Program Implementation.~~

~~g. New and spent fuel storage.~~

~~h. Offsite Dose Calculation Manual and Process Control Program implementation at the site.~~

LATER

2.2.4 In MODES 3, 4, and 5, if SL 2.1.2 is violated, restore RCS pressure to ≤ 2750 psig within 5 minutes.

2.0-01

# NO SIGNIFICANT HAZARDS CONSIDERATIONS
# GENERIC EVALUATIONS

**"R" - Relocation of requirements:**

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

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

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

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

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

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

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

# 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.

# NO SIGNIFICANT HAZARDS CONSIDERATIONS
# GENERIC EVALUATIONS

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

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

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

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

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

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

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

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

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

# NO SIGNIFICANT HAZARDS CONSIDERATIONS
# GENERIC EVALUATIONS

**"M" - More restrictive changes to requirements:**

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

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

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

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

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

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

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

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

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

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

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

# NO SIGNIFICANT HAZARDS CONSIDERATIONS STATEMENTS

## ITS Section 2.0: Safety Limits

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

### 2.0 L1

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

The change results in a modification of the Applicability of the Safety Limits. The Safety Limits are not accident initiators. Therefore, the probability of any previously evaluated accident is not significantly increased. The accident mitigation features of the plant are not affected by this change. Following implementation of this change, the reactor coolant system (RCS) Safety Limit must be met in MODES 1, 2, 3, 4, and 5. The current Applicability is stated as "when there are fuel assemblies in the vessel." This change results in a relaxation of the Applicability in that during MODE 6 the Safety Limit will no longer apply. Although a short time period may exist between entry into MODE 6 (when the first reactor vessel head bolt is detensioned), and actual reactor vessel head removal (following which overpressurization is not possible), the consequences of an overpressure event are mitigated by the implementation of low temperature overpressurization protection requirements and administrative controls. Therefore, the consequences of any previously evaluated accident are not significantly increased.

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

The Safety Limits are not accident initiators. Therefore, the scope of the change does not establish a potential new accident precursor.

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

This change does involve an incremental reduction in the margin of safety since the RCS pressure Safety Limit will no longer be applicable when fuel is in the reactor vessel and the unit is in MODE 6. However, this reduction is not considered significant in that sufficient controls exist to prevent the occurrence of and mitigate the effects of postulated low temperature overpressure events.

# ITS DISCUSSION OF DIFFERENCES
## ITS Section 2.0: Safety Limits

1 NUREG 2.1.1.1 - The plant specific information from CTS 2.1 for maximum local fuel pin centerline temperature was inserted in ITS 2.1.1.1. Two separate temperatures were inserted to account for the two analyzed fuel assembly types used at ANO-1. This information is consistent with the current licensing basis.

2 NUREG 2.1.1- Incorporates TSTF-126.

3 NUREG 2.2- Incorporates TSTF-005, Rev 1 with the exception that NUREG 2.2.5 is retained as a unit specific preference. This requirement is consistent with the current licensing basis.

4 NUREG 2.2 - The wording in ITS 2.2.3 and 2.2.4 was modified to be consistent with the wording used in ITS 2.2.1 and 2.2.2. The words "not met" were replaced with the word "violated." This change precludes the potential misinterpretation of an unintended distinction, is administrative in nature and has been made for consistency with similar ITS.

5 Bases - Reference to the Main Steam Safety Valves (MSSVs) as contributors in preventing the violation of Reactor Core Safety Limits was deleted at each occurrence. Chapter 3 and 14 of the Unit 1 SAR do not explicitly credit the MSSVs as functioning to prevent exceeding Reactor Core Safety Limits, since the startup evaluation does not model the secondary side.

Reference to the RCS High Temperature trip as a contributor in preventing the violation of Reactor Core Safety Limits was added. Although Chapter 3 and 14 of the Unit 1 SAR do not explicitly credit this trip function, it is relied upon to set boundaries for the analyses.

6 Bases - ANO-1 uses the terms "RCS Variable Low Pressure trip" and "Pressure Temperature trip" interchangeably. Therefore, both terms are presented in the Bases.

7 Bases - Specific detail relating to the two critical heat flux correlations at ANO-1 has been included in the ITS B 2.1.1 Background information. This information is consistent with the ANO-1 current licensing basis. References 2 and 3 have been added to reference the respective topical reports associated with the heat flux correlations.

8 Bases - Specific reference to the ASME code was deleted in favor of reference to "design codes" which more accurately reflects the number of codes to which the plant was designed and built.

9 Bases - The word "event" was added in paragraph two (2) of the APPLICABLE SAFETY ANALYSES section to more clearly define the basis for the relief valve capacity. This wording is consistent with the wording in the SAR which provides evaluation of individual rod, multiple rod and rod bank events. The last sentence on

page B2.0-6 of the BWOG STS was deleted as it does not accurately establish the plant conditions established in the ANO-1 SAR Safety Analyses supporting the determination of required relief valve capacity. These plant conditions are established in the ANO-1 SAR.

10 Bases - The ANO-1 Design Code for piping, valves and fittings was USAS B31.7 which provides for a maximum transient pressure of 110% of design pressure. Because this is the same allowance as stated under the ASME Code, Section III, the sentence starting with "The most limiting of these..." is unnecessary as both are equally limiting. In addition, the text cites Reference 6 which was also modified to accurately reflect the correct design code.

11 Bases - Power operated relief valve (PORV) has been replaced by the ANO-1 specific designation "electromatic relief valve (ERV). This change was made for consistency with ANO-1 documentation.

12 Bases - The background discussion for LCO 2.1.2 has been revised to incorporate the ANO-1 current licensing basis with respect to reactor coolant system (RCS) leak testing. Specifically, the NUREG-1430 description of the RCS inservice operational hydrotest at 100% design pressure has been replaced with a description of the CTS 3.3.2 RCS leak test performed at not less than 2155 psig. Information from the CTS 4.3.2 Bases describing other requirements satisfied by the performance of this leak test has also been included.

13 Bases - Specific detail was added to item 2 of the RCS Pressure Safety Limit REFERENCES specifying that the 1965, Summer '67 Addenda was the reference ASME Boiler and Pressure Vessel Code , Section III, used for determining the design requirements for the RCS pressurizer safety valves for ANO-1.

14 Bases - The Insert adds specific reference to the analysis code (TACO2 or TACO3) used in the fuel design analysis for determining the maximum fuel centerline temperature. This analysis is performed in accordance with the calculational methods described in BAW-10141 or BAW 10162 which were cited as references in section B 2.1.1.

15 Bases - The term AOO is used in the GDCs, but the ANO-1 licensing basis is contingent upon discussion of "abnormalities" as defined and listed in SAR, Section 14.1. The ANO-1 SAR was written partially based on the guidance given in a "Guide to the Organization and Contents of Safety Analysis Reports" issued by the Atomic Energy Commission on June 30, 1966. This document discusses what transients or "abnormalities" should be considered for Core and Coolant Boundary Protection Analysis. Statements concerning the GDC criteria are modified in the ITS to reference the current licensing basis description in the Unit 1 SAR.

# ITS DISCUSSION OF DIFFERENCES

16 Bases - The word "significantly" is added to the last sentence of the Applicability discussion for 2.1.2. This is added to clarify that some pressurization due to the formation steam can be expected if the head is in place and not fully detensioned and removed. However, in agreement with the RSTS bases, the amount of pressurization is not expected to be significant and thus the Specification should not be applicable in MODE 6.

17 NUREG 2.2 - ITS 2.2.2 and 2.2.3 were editorially changed to reflect a Logical Connector structure consistent with the requirements of Section 1.2 of the ITS.

18 Bases - For ANO-1, the startup event (rod withdrawal from low power) is the limiting event for Pressurizer Safety Valve design; and thus, the Bases were modified to identify that this was the limiting event. The cited overpressure protection analyses were not the bases used and reference to them was deleted.

19 NUREG 2.1.1.3 is revised to retain the reference to the "Variable Low RCS Pressure/Temperature Protection Limits as specified in the COLR." This limit is maintained in the COLR (as recently approved in Amendment No. 186) since it is a cycle specific parameter. Use of this reference to the COLR also eliminates the need for NUREG Figure 2.1.1-1. An editorial change has been made, adding the phrase ", so that the safety limits are met" to the end of ITS 2.1.1.3 to ensure that there is no confusion over the limits that are located in the Core Operating Limits Report.

2.0-02

The Bases are also revised to reference the COLR rather than the safety limit of ITS 2.1.1.3 since the COLR actually provides the pressure/temperature relationship. This modification improves clarity by providing a direct reference to the location of the limits. Additionally, a Bases paragraph is incorporated to establish that the COLR represents the most limiting condition of pressure/temperature combinations for reactor coolant pump maximum THERMAL POWER combinations. Analyses have been performed for three pump operations and one pump per loop operations which demonstrate the four pump curve is bounding. Incorporation of this statement clarifies the acceptability of operation with less than four RCPs.

20 Bases – Information related to Statistical Core Design (SCD) methodology has been added to maintain consistency with other Unit 1 LBDs. SCD was first integrated into the reload process for protection from DNB for the first time in Cycle 15. This method is described fully in topical report BAW-10187P-A and referenced in the reload methodology topical BAW-10179P-A. A reference to BAW-10179P-A has also been incorporated into the References Section.

CTS

## 2.0 SAFETY LIMITS (SLs)

5080 – (6.5 x $10^{-3}$ x (BURNUP, MWD/MTU) °F) for TACO2 applications AND ≤ 4642 – (5.8 x $10^{-3}$ x (BURNUP MWD/MTU) °F) for TACO3 applications. (1)

### 2.1 SLs

#### 2.1.1 Reactor Core SLs

2.1.1.1 In MODES 1 and 2, the maximum local fuel pin centerline temperature shall be ≤ ~~[5080 – (6.5 x $10^{-3}$ MWD/MTU) °F].~~ ~~Operation within this limit is ensured by compliance with the AXIAL POWER IMBALANCE protective limits preserved by the Reactor Protection System setpoints in LCO 3.3.1, "Reactor Protection System (RPS) Instrumentation," as specified in the COLR.~~ (2) 2.1.1

2.1.1.2 In MODES 1 and 2, the departure from nucleate boiling ratio shall be maintained greater than the limits of ~~[1.3~~ 1.3 for the BAW-2 correlation and 1.18 for the BWC correlation~~]~~. ~~Operation within this limit is ensured by compliance with SL 2.1.1.3 and with the AXIAL POWER IMBALANCE protective limits preserved by the RPS setpoints in LCO 3.3.1, as specified in the COLR.~~ (2) 2.1.2

2.1.1.3 In MODES 1 and 2, Reactor Coolant System (RCS) core outlet temperature and pressure shall be maintained above and to the left of the ~~SL shown in Figure 2.1.1-1.~~ Variable Low RCS Pressure-Temperature Protective Limits as specified in the COLR, so that the safety limits are met. (19) 2.1.3

#### 2.1.2 RCS Pressure SL

In MODES 1, 2, 3, 4, and 5, the RCS pressure shall be maintained ≤ ~~[2750]~~ 2750 psig. 2.2.1

### 2.2 SL Violations

With any SL violation, the following actions shall be completed:

2.2.1 In MODE 1 or 2, if SL 2.1.1.1 or SL 2.1.1.2 is violated, be in MODE 3 within 1 hour. 6.7.1.a

2.2.2 In MODE 1 or 2, if SL 2.1.1.3 is violated, restore RCS pressure and temperature within limits ~~and~~ AND be in MODE 3 within 1 hour. (17) 6.7.1.a

(continued)

2.0-02

CTS

## 2.0 SLs

2.2 SL Violations (continued) (4)

2.2.3 In MODE 1 or 2, if SL 2.1.2 is ~~not met~~ violated, restore compliance within limits ~~and~~ AND be in MODE 3 within 1 hour. (17) (4) 6.7.1.a

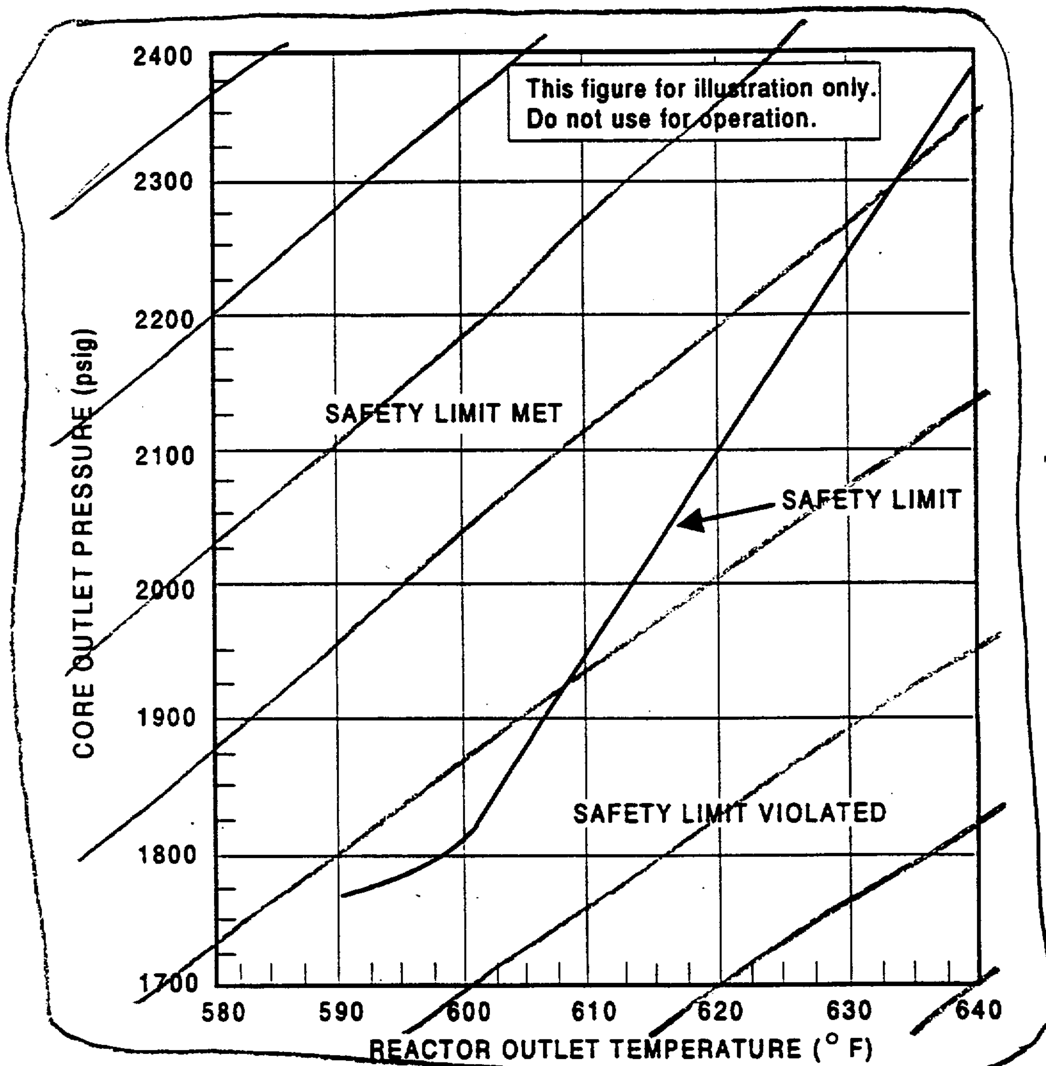
2.2.4 In MODES 3, 4, and 5, if SL 2.1.2 is ~~not met~~ violated, restore RCS pressure to ≤ ~~[2750]~~ 2750 psig within 5 minutes. N/A

2.2.5 Within 1 hour, notify the NRC Operations Center, in accordance with 10 CFR 50.72. 6.7.1.b

~~2.2.6 Within 24 hours, notify the [Vice President—Nuclear Operations].~~

~~2.2.7 Within 30 days, a Licensee Event Report (LER) shall be prepared pursuant to 10 CFR 50.73. The LER shall be submitted to the NRC and the [Plant Superintendent, and Vice President—Nuclear Operations].~~

~~2.2.8 Operation of the plant shall not be resumed until authorized by the NRC.~~ (3)




19

~~Figure 2.1.1-1 (page 1 of 1)~~
~~Reactor Coolant System Departure from Nucleate Boiling Safety Limits~~

# B 2.0 SAFETY LIMITS (SLs)

## B 2.1.1 Reactor Core SLs

BASES

BACKGROUND

GDC 10 (Ref. 1) requires that reactor core SLs ensure specified acceptable fuel design limits are not exceeded during steady state operation, normal operational transients, and ~~anticipated operational occurrences (AOOs)~~. abnormalities 15 This is accomplished by having a departure from nucleate boiling (DNB) design basis, which corresponds to a 95% probability at a 95% confidence level (95/95 DNB criterion) that DNB will not occur and by requiring that the fuel centerline temperature stays below the melting temperature. INSERT B2.0-1A 7

INSERT B2.0-1B 20

The restrictions of this SL prevent overheating of the fuel and cladding and possible cladding perforation that would result in the release of fission products to the reactor coolant. Overheating of the fuel is prevented by maintaining the steady state peak linear heat rate (LHR) below the level at which fuel centerline melting occurs. INSERT B2.0-1C 14 Overheating of the fuel cladding is prevented by restricting fuel operation to within the nucleate boiling regime, where the heat transfer coefficient is large and the cladding surface temperature is slightly above the coolant saturation temperature.

Fuel centerline melting occurs when the local LHR, or power peaking, in a region of the fuel is high enough to cause the fuel centerline temperature to reach the melting point of the fuel. Expansion of the pellet upon centerline melting may cause the pellet to stress the cladding to the point of failure, allowing an uncontrolled release of activity to the reactor coolant.

Operation above the boundary of the nucleate boiling regime could result in excessive cladding temperature because of the onset of DNB and the resultant sharp reduction in heat transfer coefficient. Inside the steam film, high cladding temperatures are reached, and a cladding-water (zirconium-water) reaction may take place. edit This chemical reaction results in oxidation of the fuel cladding ~~to~~. The oxidized cladding then exists in a structurally weaker form. edit This weaker form may lose its integrity, resulting in an uncontrolled release of activity to the reactor coolant.

(continued)

**<INSERT B2.0-1A>**

Although DNB is not an observable parameter during reactor operation, the observable parameters of neutron power, reactor coolant flow, temperature and pressure can be related to DNB through the use of a critical heat flux (CHF) correlation. The BAW-2 (Ref. 2) and BWC (Ref. 3) correlations have been developed to predict DNB and the location of DNB for axially uniform and non-uniform heat flux distributions. The BAW-2 correlation applies to Mark-B fuel and the BWC correlation applies to Mark-BZ fuel. The local DNB ratio (DNBR), defined as the ratio of the heat flux that would cause DNB at a particular core location to the actual heat flux, is indicative of the margin to DNB. The minimum value of the DNBR, during steady state operation, normal operational transients and anticipated transients is limited to 1.30 (BAW-2) and 1.18 (BWC).

**<INSERT B2.0-1B>**

The 95 percent confidence level that DNB will not occur is preserved by ensuring that the DNBR remains greater than the DNBR design limit based on the applicable CHF correlation for the core design. In the development of the applicable DNBR design limit (Ref. 4), uncertainties in the core state variables, power peaking factors, manufacturing-related parameters, and the CHF correlation may be statistically combined to determine a statistical DNBR design limit. This statistical design limit protects the respective CHF design limit. Additional retained thermal margin may also be applied to the statistical DNBR design limit to yield a higher thermal design limit for use in establishing DNB-based core safety and operating limits. In all cases, application of statistical DNB design methods preserves a 95 percent probability at a 95 percent confidence level that DNB will not occur.

**<INSERT B2.0-1C>**

The maximum fuel centerline temperatures are given by the relationships defined in SL 2.1.1.1 for the respective fuel designs and are dependent on whether the TACO2 (Ref. 5) or TACO3 (Ref. 6) analysis was utilized.

BASES

**BACKGROUND (continued)**

The proper functioning of the Reactor Protection System (RPS) ~~and main steam safety valves (MSSVs)~~ prevents violation of the reactor core SLs. 5

**APPLICABLE SAFETY ANALYSES**

The fuel cladding must not sustain damage as a result of normal operation and ~~AOOs~~ abnormalities. The reactor core SLs are established to preclude violation of the following fuel design criteria: 15

a. There must be at least 95% probability at a 95% confidence level (95/95 DNB criterion) that the hot fuel rod in the core does not experience DNB; and

b. The hot fuel pellet in the core must not experience fuel centerline melting.

The RPS setpoints (Ref. ~~2~~ 1), in combination with all the LCOs, ~~is~~ are designed to prevent any ~~anticipated~~ analyzed combination of transient conditions for Reactor Coolant System (RCS) temperature, pressure, and THERMAL POWER level that would result in a ~~departure from nucleate boiling ratio~~ ~~(~~DNBR~~)~~ of less than the DNBR limit and preclude the existence of flow instabilities. edit edit edit

Automatic enforcement of these reactor core SLs is provided by the following:

a. RCS High Pressure trip;

b. RCS Low Pressure trip;

c. Nuclear Overpower trip;

d. RCS Variable Low Pressure trip~~;~~ (also known as Pressure Temperature trip); 6

e. Reactor Coolant Pump to Power trip;

f. Nuclear Overpower RCS Flow and Axial Power Imbalance (caps) trip; and edit

g. ~~MSSVs.~~ RCS High Temperature trip 5

The SL represents a design requirement for establishing the RPS trip setpoints identified previously.

(continued)

BASES (continued)

SAFETY LIMITS

SL 2.1.1.1, SL 2.1.1.2, and SL 2.1.1.3 ensure that the minimum DNBR is not less than the safety analyses limit and that fuel centerline temperature stays below the melting point, or the average enthalpy in the hot leg is less than or equal to the enthalpy of saturated liquid, or the exit quality is within the limits defined by the DNBR correlation. In addition, ~~SL 2.1.1.a~~ ~~shows~~ the COLR identifies the pressure/temperature operating region that keeps the reactor from reaching an SL when operating up to design power~~, and it defines the safe operating region from brittle fracture concerns~~. 19 edit

INSERT B2.0-3A 19

The SLs are preserved by monitoring the process variable AXIAL POWER IMBALANCE to ensure that the core operates within the fuel design criteria. AXIAL POWER IMBALANCE protective limits are ~~provided in the COLR~~. The trip setpoints are derived by adjusting the measurement system independent AXIAL POWER IMBALANCE protective limits given in the COLR to allow for measurement system observability and instrumentation errors.

~~Operation within these limits is ensured by compliance with the AXIAL POWER IMBALANCE protective limits~~ preserved by their corresponding RPS setpoints in LCO 3.3.1, "Reactor Protection System (RPS) Instrumentation," ~~as specified~~ and are provided in the COLR. The AXIAL POWER IMBALANCE protective limits are separate and distinct from the AXIAL POWER IMBALANCE operating limits defined by LCO 3.2.3, "AXIAL POWER IMBALANCE Operating Limits." The AXIAL POWER IMBALANCE operating limits in LCO 3.2.3, also specified in the COLR, preserve initial conditions of the safety analyses but are not reactor core SLs. edit

APPLICABILITY

SL 2.1.1.1, SL 2.1.1.2, and SL 2.1.1.3 only apply in MODES 1 and 2 because these are the only MODES in which the reactor is critical. Automatic protection functions are required to be OPERABLE during MODES 1 and 2 to ensure operation within the reactor core SLs. ~~The MSSVs, or~~ automatic protection actions~~,~~ serve to prevent RCS heatup to reactor core SL conditions or to initiate a reactor trip function, which forces the unit into MODE 3. Setpoints for the reactor trip functions are specified in LCO 3.3.1. 5

(continued)

<INSERT B2.0-3A>

The COLR presents the most limiting condition of pressure/temperature combinations for all possible reactor coolant pump maximum THERMAL POWER combinations. Analyses have been performed which bound the three pump and two pump (one pump in each loop) allowed operating conditions based on the expected minimum flow rates and maximum ALLOWABLE THERMAL POWER for these operating conditions.

BASES

**APPLICABILITY (continued)**

In MODES 3, 4, 5, and 6, Applicability is not required, since the reactor is not generating significant THERMAL POWER.

**SAFETY LIMIT VIOLATIONS**

The following SL violation responses are applicable to the reactor core SLs.

2.2.1 and 2.2.2

If SL 2.1.1.1, SL 2.1.1.2, or SL 2.1.1.3 is violated, the requirement to go to MODE 3 places the plant in a MODE in which these SLs are not applicable.

The allowed Completion Time of 1 hour recognizes the importance of bringing the plant to a MODE of operation where these SLs are not applicable and reduces the probability of fuel damage.

2.2.5

If SL 2.1.1.1, SL 2.1.1.2, or SL 2.1.1.3 is violated, the NRC Operations Center must be notified within 1 hour, in accordance with 10 CFR 50.72 (Ref. ~~2~~ 8).

edit

~~2.2.6~~

~~If SL 2.1.1.1, SL 2.1.1.2, or SL 2.1.1.3 is violated, the appropriate senior management of the nuclear plant and the utility shall be notified within 24 hours. This 24 hour period provides time for the plant operators and staff to take the appropriate immediate action and assess the condition of the unit before reporting to senior management.~~

3

~~2.2.7~~

~~If SL 2.1.1.1, SL 2.1.1.2, or SL 2.1.1.3 is violated, a Licensee Event Report shall be prepared and submitted within 30 days to the NRC in accordance with 10 CFR 50.73 (Ref. 4). A copy of the report shall also be submitted to the senior~~

(continued)

BASES

~~SAFETY LIMIT VIOLATIONS~~

~~2.2.7 (continued)~~

~~management of the nuclear plant, and the utility Vice President—Nuclear Operations.~~

~~2.2.8~~

~~If SL 2.1.1.1, SL 2.1.1.2, or SL 2.1.1.3 is violated, restart of the unit shall not commence until authorized by the NRC. This requirement ensures the NRC that all necessary reviews, analyses, and actions are completed before the unit begins its restart to normal operation.~~ 3

REFERENCES

SAR, Section 1.4,

1. ~~10 CFR 50, Appendix A,~~ GDC 10. edit 7 14 20

INSERT B2.0-5A

7. ~~2.~~ ~~F~~SAR, ~~Section~~ Chapters ~~[ ]~~ 3 & 14. edit

8. ~~3.~~ 10 CFR 50.72.

~~4. 10 CFR 50.73.~~ 3

<INSERT B2.0-5A>

2. BAW-10000A, "Correlation of Critical Heat Flux in a Bundle Cooled by Pressurized Water," Babcock & Wilcox, Lynchburg, VA, May 1976 . 7

3. BAW-10143P-A, "BWC Correlation of Critical Heat Flux," Babcock & Wilcox, Lynchburg, VA, April 1985.

4. BAW-10187P-A, "Statistical Core Design for B&W Designed 177 FA Plants," Babcock & Wilcox, Lynchburg, VA, March 1994 20

5. BAW-10141P-A, Rev. 1, "TACO2 Fuel Pin Performance Analysis," Babcock & Wilcox, Lynchburg, VA, June 1983.

6. BAW-10162P-A, "TACO3 Fuel Pin Thermal Analysis Code," Babcock & Wilcox, Lynchburg, VA, October 1989. 14

## B 2.0 SAFETY LIMITS (SLs)

### B 2.1.2 Reactor Coolant System (RCS) Pressure SL

BASES

BACKGROUND

<INSERT B2.0-6A>

~~According to 10 CFR 50, Appendix A, GDC 14, "Reactor Coolant Pressure Boundary," and GDC 15, "Reactor Coolant System Design" (Ref. 1), the reactor coolant pressure boundary (RCPB) design conditions are not to be exceeded during normal operation nor during anticipated operational occurrences (AOOs).~~ (15) GDC 28, "Reactivity Limits" (Ref. 1), specifies that reactivity accidents including rod ejection do not result in damage to the RCPB greater than limited local yielding.

The design pressure of the RCS is 2500 psig. During normal operation and ~~AOOs~~ abnormalities, the RCS pressure is kept from exceeding the design pressure by more than 10% in order to remain in accordance with ~~Section III of the ASME Code (Ref. 2)~~ the design codes (Ref. 3 and 4). Hence, the safety limit is 2750 psig. To ensure system integrity, all RCS components ~~are~~ were hydrostatically tested at 125% of design pressure prior to initial operation, according to the ~~ASME Code~~ design code requirements. (8) ~~Inservice operational hydrotesting at 100% of design pressure is also required whenever the reactor vessel head has been removed or if other pressure boundary joint alterations have occurred. Following inception of unit operation, RCS components shall be pressure tested, in accordance with the requirements of ASME Code, Section XI (Ref. 3)~~ (12)

<INSERT B2.0-6B>

APPLICABLE SAFETY ANALYSES

The RCS pressurizer safety valves, operating in conjunction with the Reactor Protection System trip settings, ensure that the RCS pressure SL will not be exceeded.

The RCS pressurizer safety valves are sized to prevent system pressure from exceeding the design pressure by more than 10%, in accordance with Section III of the ASME Code for Nuclear Power Plant Components (Ref. ~~2~~ 3). The design basis transient that is most influential for establishing the required relief capacity, and hence the valve size requirements and lift settings, is a rod withdrawal event from low power. (9) ~~During the transient, no control actions are assumed except that the safety valves on the secondary plant are assumed to open~~

(continued)

<INSERT B2.0-6A>

In SAR, Section 1.4 (Ref. 1), GDC 14, "Reactor Coolant Pressure Boundary (RCPB)," and GDC 15, "Reactor Coolant System Design", address RCPB design and protection, respectively. The ANO-1 discussion regarding how GDC 15 is accomplished states that analysis and evaluation of all normal and abnormal operating conditions and transients are integrally related to all RCS and associated systems design. SAR Chapter 14 (Ref. 2) lists these abnormal operating conditions and transients and terms them "abnormalities". In addition,

<INSERT B2.0-6B

Inservice leak testing at not less than 2155 psig is also required, prior to MODE 2, following any opening of the reactor coolant system in accordance with ASME code, Section XI; IWA-5000. When performed at the end of refueling outages, this leak test also satisfies the requirements of IWB-2500, Table IWB-2500-1; Category B-P items B15.10, B15.20, B15.30, B15.40, B15.50, B15.60, and B15.70 for all Class I pressure retaining components (Ref. 5).

BASES

APPLICABLE SAFETY ANALYSES (continued)

~~when the steam pressure reaches the secondary plant safety valve settings, and nominal feedwater supply is maintained.~~ 9

The startup event analysis (rod withdrawal at low power event) (Ref 2) is ~~The overpressure protection analyses (Ref. 4) and the safety analyses (Ref. 5) are~~ performed using conservative assumptions relative to pressure control devices. 18

2.0-03

More specifically, no credit is taken for operation of the following:

a. Electromatic Relief Valve (ERV); ~~Pressurizer power operated relief valves (PORVs);~~ 11

b. Steam line turbine bypass valves;

c. Control system runback of reactor and turbine power; and

d. Pressurizer spray valve.

SAFETY LIMITS

The maximum transient pressure allowed in the RCS pressure vessel under the ASME Code, Section III, is 110% of design pressure. The maximum transient pressure allowed in the RCS piping, valves, and fittings under USAS B31.7 ~~USAS, Section B31.1~~ (Ref. 4 ~~6~~), is 110% ~~120%~~ of design pressure. ~~The most limiting of these two allowances is the 110% of design pressure;~~ therefore, the SL on maximum allowable RCS pressure is 2750 psig. 10

Overpressurization of the RCS can result in a breach of the RCPB. If such a breach occurs in conjunction with a fuel cladding failure, fission products could enter the containment atmosphere, raising concerns relative to limits on radioactive releases specified in 10 CFR 100, "Reactor Site Criteria" (Ref. 6 ~~7~~).

APPLICABILITY

SL 2.1.2 applies in MODES 1, 2, 3, 4, and 5 because this SL could be approached or exceeded in these MODES during overpressurization events. The SL is not applicable in MODE 6 because the reactor vessel head closure bolts are not fully tightened, making it unlikely that the RCS can be pressurized significantly. 16

(continued)

BASES (continued)

**SAFETY LIMIT VIOLATIONS**

The following SL violation responses are applicable to the RCS pressure SL.

2.2.3

If the RCS pressure SL is violated when the reactor is in MODE 1 or 2, the requirement is to restore compliance and be in MODE 3 within 1 hour.

Exceeding the RCS pressure SL may cause immediate RCS failure and create a potential for radioactive releases in excess of 10 CFR 100, "Reactor Site Criteria," limits (Ref. 6).

The allowed Completion Time of 1 hour is based on the importance of reducing power level to a MODE ~~of operation~~ where the potential for challenges to safety systems is minimized. Edit

2.2.4

If the RCS pressure SL is exceeded in MODE 3, 4, or 5, RCS pressure must be restored to within the SL value within 5 minutes.

Exceeding the RCS pressure SL in MODE 3, 4, or 5 is potentially more severe than exceeding this SL in MODE 1 or 2, since the reactor vessel temperature may be lower and the vessel material, consequently, less ductile. As such, pressure must be reduced to less than the SL within 5 minutes. This action does not require reducing MODES, since this would require reducing temperature, which would compound the problem by adding thermal gradient stresses to the existing pressure stress.

2.2.5

If the RCS pressure SL is violated, the NRC Operations Center must be notified within 1 hour, in accordance with 10 CFR 50.72 (Ref. 7). EDIT

(continued)

BASES

~~SAFETY LIMIT VIOLATIONS (continued)~~

~~2.2.6~~

~~If the RCS pressure SL is violated, the appropriate senior management of the nuclear plant and the utility shall be notified within 24 hours. This 24 hour period provides time for the plant operators and staff to take the appropriate immediate action and assess the condition of the unit before reporting to senior management.~~

~~2.2.7~~

~~If the RCS pressure SL is violated, a Licensee Event Report shall be prepared and submitted within 30 days to the NRC, in accordance with 10 CFR 50.73 (Ref. 9). A copy of the report shall also be provided to the senior management of the nuclear plant, and the utility Vice President—Nuclear Operations and the [offsite reviewers specified in Specification 5.2.2] ["Offsite Review and Audit"].~~

~~2.2.8~~

~~If the RCS pressure SL is violated, restart of the unit shall not commence until authorized by the NRC. This requirement ensures the NRC that all necessary reviews, analyses, and actions are completed before the unit begins its restart to normal operation.~~ ③

REFERENCES

1. SAR, Section 1.4, ~~10 CFR 50, Appendix A,~~ GDC 14, GDC 15, and GDC 28~~, 1988~~. EDIT.

3 ~~2~~. ASME Boiler and Pressure Vessel Code, Section III, 1965 — S67, Article NB-7000. ⑬

5 ~~3~~. ASME Boiler and Pressure Vessel Code, Section XI, Article IW-5000.

~~4. BAW-10043, May 1972.~~ Chapter ⑱

2 ~~5~~. ~~F~~SAR, ~~Section~~ 14~~.1~~.

4 ~~6~~. ~~ASME~~ USAS B31.7 ~~B31.1~~, Nuclear Power ~~Standard Code for Pressure~~ Piping, 1969. ~~1967.~~ ⑩

(continued)

**BASES**

REFERENCES (continued)

6 ~~7~~. 10 CFR 100.

7 ~~8~~. 10 CFR 50.72.

~~9. 10 CFR 50.73.~~ 3

## 3.1 REACTIVITY CONTROL SYSTEMS

### 3.1.1 SHUTDOWN MARGIN (SDM)

LCO 3.1.1 The SDM shall be within the limit specified in the COLR.

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

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

APPLICABILITY: MODES 1 and 2.

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. | 7 days |
| | AND | |
| | A.2 Establish appropriate operating restrictions and SRs. | 7 days |
| B. Required Action and associated Completion Time not met. | B.1 Be in MODE 3. | 6 hours |

SURVEILLANCE REQUIREMENTS

| SURVEILLANCE | | FREQUENCY |
|---|---|---|
| SR 3.1.2.1 | -------------------------NOTES-------------------------<br>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.<br><br>2. This Surveillance is not required to be performed prior to entry into MODE 2.<br>---------------------------------------------------------------<br><br>Verify measured core reactivity balance is within ± 1% $\Delta k/k$ of predicted values. | Once prior to entering MODE 1 after each fuel loading<br><br><u>AND</u><br><br>--------NOTE--------<br>Only required after 60 EFPD<br>------------------------<br><br>31 EFPD thereafter |

## 3.1 REACTIVITY CONTROL SYSTEMS

### 3.1.3 Moderator Temperature Coefficient (MTC)

LCO 3.1.3 The MTC shall be non-positive whenever THERMAL POWER is $\geq$ 95% RTP and shall be less positive than $0.9 \times 10^{-4}$ $\Delta k/k/°F$ whenever THERMAL POWER is < 95% RTP.

APPLICABILITY: MODES 1 and 2.

ACTIONS

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

SURVEILLANCE REQUIREMENTS

| SURVEILLANCE | FREQUENCY |
|---|---|
| SR 3.1.3.1 Verify MTC is within the limits. | Once prior to entering MODE 1 after each fuel loading |

## 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% of its group average height.

APPLICABILITY: MODES 1 and 2.

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<br><br>AND<br><br>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 $\leq$ 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 | |

| CONDITION | REQUIRED ACTION | | COMPLETION TIME |
|---|---|---|---|
| A. (continued) | A.2.2.3 | --------------NOTE--------------- Only required when THERMAL POWER is > 20% RTP. ----------------------------------- | |
| | | Perform SR 3.2.5.1. | 72 hours |
| B. Required Action and associated Completion Time for Condition A not met. | B.1 | Be in MODE 3. | 6 hours |
| C. More than one CONTROL ROD inoperable, or not aligned within 6.5% of its group average height, or both. | C.1.1 | Verify SDM to be within the limit provided in the COLR. | 1 hour |
| | OR | | |
| | C.1.2 | Initiate boration to restore SDM to within limit. | 1 hour |
| | AND | | |
| | C.2 | Be in MODE 3. | 6 hours |

SURVEILLANCE REQUIREMENTS

| SURVEILLANCE | | FREQUENCY |
|---|---|---|
| SR 3.1.4.1 | Verify individual CONTROL ROD positions are within 6.5% of their group average height. | 12 hours |
| SR 3.1.4.2 | Verify CONTROL ROD freedom of movement for each individual CONTROL ROD that is not fully inserted. | 92 days |

| SURVEILLANCE | | FREQUENCY |
|---|---|---|
| SR 3.1.4.3 | ------------------------------NOTE------------------------------<br>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.<br>------------------------------------------------------------------------<br>Verify the rod drop time for each CONTROL ROD, from the fully withdrawn position, is $\leq$ 1.66 seconds from power interruption at the CONTROL ROD drive breakers to ¾ insertion (25% withdrawn position) with Tavg $\geq$ 525°F. | Once prior to reactor criticality after each removal of the reactor vessel head |

## 3.1 REACTIVITY CONTROL SYSTEMS

### 3.1.5 Safety Rod Insertion Limits

LCO 3.1.5 Each safety rod shall be fully withdrawn.

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

Not required for any safety rod inserted to perform SR 3.1.4.2.

------------------------------------------------------------------------------------------

APPLICABILITY: MODES 1 and 2.

ACTIONS

| CONDITION | REQUIRED ACTION | | COMPLETION TIME |
|---|---|---|---|
| A. One safety rod not fully withdrawn. | A.1.1 | Verify SDM to be within the limit provided in the COLR. | 1 hour |
| | <u>OR</u> | | |
| | A.1.2 | Initiate boration to restore SDM to within limit. | 1 hour |
| | <u>AND</u> | | |
| | A.2 | Declare the rod inoperable. | 1 hour |
| B. More than one safety rod not fully withdrawn. | B.1.1 | Verify SDM to be within the limit provided in the COLR. | 1 hour |
| | <u>OR</u> | | |
| | B.1.2 | Initiate boration to restore SDM to within limit. | 1 hour |
| | <u>AND</u> | | |
| | B.2 | Be in MODE 3. | 6 hours |

SURVEILLANCE REQUIREMENTS

| SURVEILLANCE | | FREQUENCY |
|---|---|---|
| SR 3.1.5.1 | Verify each safety rod is fully withdrawn. | 12 hours |

## 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 to within 6.5% of its group average height.

APPLICABILITY: MODES 1 and 2.

ACTIONS

| CONDITION | REQUIRED ACTION | COMPLETION TIME |
|---|---|---|
| A. One APSR inoperable, or not aligned to within 6.5% of its group average height, or both. | A.1 Perform SR 3.2.5.1. | 2 hours<br><br><u>AND</u><br><br>2 hours after each APSR movement |
| B. Required Action and associated Completion Time not met. | B.1 Be in MODE 3 | 6 hours |

SURVEILLANCE REQUIREMENTS

| SURVEILLANCE | | FREQUENCY |
|---|---|---|
| SR 3.1.6.1 | Verify position of each APSR is within 6.5% of the group average height. | 12 hours |

## 3.1 REACTIVITY CONTROL SYSTEMS

### 3.1.7 Position Indicator Channels

LCO 3.1.7 One position indicator channel for each CONTROL ROD and APSR shall be OPERABLE.

APPLICABILITY: MODES 1 and 2.

ACTIONS

-----NOTES-----
Separate Condition entry is allowed for each CONTROL ROD and APSR.

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| CONDITION | REQUIRED ACTION | COMPLETION TIME |
|---|---|---|
| A. The required position indicator channel inoperable for one or more rods. | A.1 Declare the rod(s) inoperable. | Immediately |

SURVEILLANCE REQUIREMENTS

| SURVEILLANCE | | FREQUENCY |
|---|---|---|
| SR 3.1.7.1 | Perform CHANNEL CHECK of required position indicator channel. | 12 hours |
| SR 3.1.7.2 | Perform CHANNEL CALIBRATION of required position indicator channel. | 18 months |

## 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

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"; and
LCO 3.2.4, "QUADRANT POWER TILT (QPT)"

may be suspended, provided:

a. THERMAL POWER is maintained ≤ 85% RTP;

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;

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

Linear Heat Rate (LHR) is maintained within the limits specified in the COLR; and

d. SDM is within the limits provided in the COLR.

APPLICABILITY: MODE 1 during PHYSICS TESTS.

ACTIONS

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

| CONDITION | REQUIRED ACTION | COMPLETION TIME |
|---|---|---|
| B. THERMAL POWER > 85% RTP.<br><br>OR<br><br>Nuclear overpower trip setpoint > 10% higher than PHYSICS TESTS power level.<br><br>OR<br><br>Nuclear overpower trip setpoint > 90% RTP.<br><br>OR<br><br>------------NOTE--------------<br>Only required when THERMAL POWER is > 20% RTP.<br>------------------------------------<br><br>LHR not within limits. | B.1 Suspend PHYSICS TESTS exceptions. | 1 hour |

SURVEILLANCE REQUIREMENTS

| SURVEILLANCE | | FREQUENCY |
|---|---|---|
| SR 3.1.8.1 | Verify THERMAL POWER is ≤ 85% RTP. | 1 hour |
| SR 3.1.8.2 | ----------------------------NOTE-----------------------------<br>Only required when THERMAL POWER is > 20% RTP.<br>-----------------------------------------------------------------<br><br>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 test plateau |

| SURVEILLANCE | | FREQUENCY |
|---|---|---|
| SR 3.1.8.4 | Verify SDM to be within the limits provided in the COLR. | 24 hours |

## 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)";
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";
LCO 3.2.2, "AXIAL POWER SHAPING ROD (APSR) Insertion Limits"; and
LCO 3.4.2, "RCS Minimum Temperature for Criticality"

may be suspended, provided:

a. THERMAL POWER is $\leq$ 5% RTP;

b. Nuclear overpower trip setpoint is set to $\leq$ 5% RTP;

c. Nuclear instrumentation high startup rate CONTROL ROD withdrawal inhibit is OPERABLE; and

d. SDM is within the limits provided in the COLR.

APPLICABILITY: During PHYSICS TESTS initiated in MODE 2.

ACTIONS

| CONDITION | REQUIRED ACTION | COMPLETION TIME |
|---|---|---|
| A. THERMAL POWER not within limit. | A.1 Open control rod drive trip breakers. | Immediately |
| B. SDM not within limit. | B.1 Initiate boration to restore SDM to within limit. | 15 minutes |
| | <u>AND</u> | |
| | B.2 Suspend PHYSICS TESTS exceptions. | 1 hour |

| CONDITION | REQUIRED ACTION | COMPLETION TIME |
|---|---|---|
| C. Nuclear overpower trip setpoint is not within limit.<br><br><u>OR</u><br><br>Nuclear instrumentation high startup rate CONTROL ROD withdrawal inhibit inoperable. | C.1 Suspend PHYSICS TESTS exceptions. | 1 hour |

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 ≤ 5% RTP. | Within 8 hours prior to performance of PHYSICS TESTS |
| SR 3.1.9.3 | Verify SDM to be within the limit provided in the COLR. | 24 hours |

# 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). In MODES 3, 4, and 5, SDM requirements provide sufficient reactivity margin to 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.

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 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).

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).

During 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 MODES 3, 4, and 5, the SDM requirements are met by means of adjustments to the RCS boron concentration. 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.

---

APPLICABLE SAFETY ANALYSES

For analyzed events in MODES 1 and 2 while critical, 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 abnormalities, with assumption of the highest worth rod stuck out following a reactor trip.

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:

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 abnormalities, and ≤ 280 cal/gm energy deposition for the rod ejection accident).

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

In MODES 1 and 2 while critical, SDM satisfies Criterion 2 of 10 CFR 50.36 (Ref. 3). In MODE 2 while subcritical and in MODES 3, 4, and 5, SDM satisfies Criterion 4 of 10 CFR 50.36.

---

LCO

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 (regulating and safety groups) and through the soluble boron concentration.

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.

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APPLICABILITY

In MODES 3, 4, and 5, the SDM requirements are applicable to provide sufficient negative reactivity to ensure that the reactor remains subcritical.

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

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 addition tank (BAAT) or the borated water storage tank (BWST). The operator should borate with the best source available for the unit conditions.

---

SURVEILLANCE REQUIREMENTS

SR 3.1.1.1

The SDM is verified by performing a reactivity balance calculation. 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 reactivity effects:

a. RCS boron concentration;

b. CONTROL ROD position;

c. RCS average temperature;

d. Fuel burnup based on gross thermal energy generation;

e. Xenon concentration;

f. Samarium concentration;

g. Isothermal temperature coefficient (ITC);

h. Moderator temperature coefficient (MTC); and

i. Doppler defect.

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

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 may include performing a boron concentration analysis, and complete the calculation.

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REFERENCES

1. SAR, Section 1.4, GDC 26.

2. SAR, Chapter 3.

3. 10 CFR 50.36.

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B 3.1 REACTIVITY CONTROL SYSTEMS

B 3.1.2 Reactivity Balance

BASES

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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 abnormalities. Therefore, the reactivity balance is used as a measure of the agreement between the predicted core reactivity and the actual core reactivity 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 core reactivity with the actual core reactivity validates the nuclear methods used in the safety analysis and supports the SDM demonstrations 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 is critical in MODE 1 or 2, a reactivity balance exists where the net reactivity is zero (referred to as the actual core reactivity state). A comparison of predicted core reactivity and the actual core reactivity is convenient under such a balance, since parameters are being maintained relatively stable under steady state power conditions and the net reactivity is known to be zero. 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.

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, the excess positive reactivity of the fuel is compensated by burnable absorbers, CONTROL RODS, APSRs, thermal feedback from the fuel and moderator, fission product poisons (mainly xenon and samarium), epithermal energy neutron absorbers, neutron leakage and the reactor coolant system (RCS) boron concentration. During cycle operation, 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.

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APPLICABLE SAFETY ANALYSES

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

Accurate prediction of core reactivity is either an explicit or implicit assumption in the accident analysis evaluations. Every accident evaluation 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 unit data, and analytical benchmarks. Monitoring the core reactivity balance ensures that the nuclear methods provide an accurate representation of the core reactivity.

Design calculations and safety analyses are performed for each fuel cycle for the purpose of predetermining reactivity behavior and the requirements for reactivity control during the operating cycle.

The comparison between the actual reactivity condition of the critical reactor and the predicted initial core reactivity provides an opportunity for the normalization of the calculational models used to predict core reactivity. If the predicted core reactivity and the actual core reactivity at reference 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 reactivity requirements may not be accurate. If reasonable agreement between the actual and predicted core reactivity exists at BOC, then the prediction may be normalized to the measured boron concentration. Thereafter, any significant deviations in the predicted reactivity condition from the actual reactivity condition during the operating cycle may be an indication that the calculational model is not adequate for the operating cycle or that an unexpected change in core conditions has occurred.

The normalization of the predicted reactivity parameters to the actual reactivity value is typically performed after reaching RTP following startup from a refueling outage, with the RCS temperature, CONTROL RODS, and APSRs in their reference positions and fission product poisons at their expected equilibrium concentrations. 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 balance satisfies Criterion 2 of 10 CFR 50.36 (Ref. 3).

---

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 accident 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 the predicted reactivity from the actual reactivity condition of the reactor is larger than expected for normal operation and should therefore be evaluated.

When the predicted core reactivity is within 1% Δk/k of the actual reactivity value at steady state thermal conditions, the core is considered to be operating within acceptable design limits.

---

APPLICABILITY

In MODES 1 and 2, the limits on the core reactivity balance must be maintained to ensure an acceptable SDM and continued adherence to the assumptions used in the accident analyses. 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 the net reactivity condition of the reactor can not be easily determined and changes to core reactivity due to fuel depletion cannot occur.

In MODE 6, boron concentration requirements (LCO 3.9.1, "Refueling Boron Concentration") ensure that fuel movements are performed within acceptable bounds.

---

ACTIONS

A.1 and A.2

Should an anomaly develop between the actual core reactivity and the predicted core reactivity, an evaluation of the core design and safety analysis must be performed. Core conditions are evaluated to determine their consistency with the input assumptions used in the core design 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 7 days is based on the low probability of an abnormality or accident occurring during this period, and allows sufficient time to assess the physical condition of the core and complete the evaluation of the core design and safety analysis.

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 the reactivity balance, then a recalculation of the reactivity balance 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 appropriate reactivity parameter may be renormalized, and operation in MODE 1 may continue. If operational restrictions or additional surveillance requirements are necessary to ensure the reactor core is acceptable for continued operation, then they must be defined.

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

B.1

If the core reactivity balance cannot be restored to within the $\pm$ 1% $\Delta k/k$ limit, the unit must be brought to a MODE in which the LCO does not apply. 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 Completion Time of 6 hours is reasonable, based on operating experience to reach the required unit conditions from RTP in an orderly manner and without challenging unit systems.

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SURVEILLANCE REQUIREMENTS

SR 3.1.2.1

Core reactivity is verified by a periodic reactivity balance calculation that compares the predicted core reactivity to the actual core reactivity condition (net reactivity of zero condition). The comparison is made considering that core conditions are fixed or stable, including CONTROL ROD and APSR positions, moderator temperature, fuel temperature, fuel depletion, xenon concentration, and samarium concentration. The Surveillance is performed once prior to entering MODE 1 after each fuel loading as an initial check on core reactivity 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 may take place within the first 60 effective full power days (EFPD) after each fuel loading. The required 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. 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. Another Note is included in the SRs to indicate that the performance of the Surveillance is not required for entry into MODE 2.

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REFERENCES

1. SAR, Section 1.4, GDC 26, GDC 28, and GDC 29.

2. SAR, Chapter 3A and 14.

3. 10 CFR 50.36

B 3.1 REACTIVITY CONTROL SYSTEMS

B 3.1.3 Moderator Temperature Coefficient (MTC)

BASES

BACKGROUND

According to GDC 11 (Ref. 1), the reactor core and associated Reactor Coolant System (RCS) shall be designed so that in the power operating range the net affect of the prompt inherent nuclear feedback characteristic tends to compensate for a rapid increase in reactivity.

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). Therefore, with a negative MTC a coolant temperature increase will cause a reactivity decrease. Reactivity increases that cause a coolant temperature increase will thus be self limiting, and stable power operation will result.

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 are evaluated to ensure the MTC does not become more negative than the value assumed in the safety analyses.

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.

The acceptance criteria for the specified MTC are:

a. The MTC values must remain within the bounds of those used in the accident analysis; and

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 the startup accident.

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 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.

MTC values are bounded in reload safety evaluations, assuming steady state conditions at BOC and EOC.

In MODES 1 and 2 while critical, MTC satisfies Criterion 2 of 10 CFR 50.36 (Ref. 3). In MODE 2 while subcritical, MTC satisfies Criterion 4 of 10 CFR 50.36.

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## LCO

LCO 3.1.3 requires the MTC to be within specified limits 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 $\Delta k/k/°F$ (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 non-positive MTC, when THERMAL POWER is ≥ 95% RTP, ensures that core operation will be stable.

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

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## APPLICABILITY

In MODE 1, the limits on MTC must be maintained to ensure that any accident initiated from 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 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 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 measurement is used for normalization.

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## ACTIONS

A.1

MTC is a 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 assumptions. 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 RTP in an orderly manner and without challenging unit systems.

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## SURVEILLANCE REQUIREMENTS

SR 3.1.3.1

The SR for measurement of the MTC at the beginning of each fuel cycle provides 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.

The requirement for measurement, prior to initial operation in MODE 1, satisfies the confirmatory check on the most positive (least negative) MTC value. MTC values are extrapolated and compensated to permit direct comparison to the specified MTC limits.

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## REFERENCES

1. SAR, Section 1.4, GDC 11.

2. SAR, Chapter 3A and 14.

3. 10 CFR 50.36.

B 3.1 REACTIVITY CONTROL SYSTEMS

B 3.1.4 CONTROL ROD Group Alignment Limits

BASES

BACKGROUND

The OPERABILITY of the CONTROL 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 SDM.

The applicable criteria for these design requirements are 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).

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 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.

Limits on CONTROL ROD alignment and OPERABILITY have been established, and all 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.

CONTROL RODS are moved by their control rod drive mechanisms (CRDMs). Each CRDM moves its rod 3/4 inch for one revolution of the leadscrew, but at varying rates depending on the signal output from the Control Rod Drive Control System (CRDCS).

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 CONTROL RODS provide required negative reactivity worth for immediate reactor shutdown upon a reactor trip. The regulating rods provide reactivity control during normal operation and transients, and their movement is normally controlled in automatic by a rod control system.

The axial position of the CONTROL RODS is indicated by three independent systems, which are the relative position indicators, the absolute position indicators, and the zone reference indicators (see LCO 3.1.7, "Position Indicator Channels").

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. 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. 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.

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 the signals from a series of reed switches spaced along a tube.

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.

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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 integrity; and

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

Two 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 when one CONTROL 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). 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).

Continued operation of the reactor with a misaligned or dropped CONTROL ROD is allowed if the local core LHRs 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 local core LHRs must be verified directly by incore mapping. Bases Section 3.2, "Power Distribution Limits," contains a more complete discussion of the relation of LHR to the operating limits.

In MODES 1 and 2 while critical, the CONTROL ROD group alignment limits satisfy Criterion 2 of 10 CFR 50.36 (Ref. 4). In MODE 2 while subcritical, the CONTROL ROD group alignment limits satisfy Criterion 4 of 10 CFR 50.36.

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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 6.5% (approximately 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 average position calculator, and asymmetric alarm or fault detector outputs. Therefore, no additional uncertainties are required to be incorporated in the implementing procedures.

For the purpose of complying with this LCO, the position of a misaligned rod is not included in the calculation of the rod group average position. 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."

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

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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 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 reactor is shut down and resultant local power peaking would not exceed fuel design limits. In MODES 3, 4, 5, and 6, the OPERABILITY of the 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 LCO 3.9.1, "Boron Concentration," for boron concentration requirements during MODE 6.

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ACTIONS

A.1.1

Compliance with Required Actions of Condition A allows for continued power operation with one CONTROL ROD inoperable, or misaligned from its group average position, or both. 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.

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.

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 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. 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, the rod shall be considered inoperable.

A.2.2.1

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.2.2.2

The existing CONTROL ROD configuration must not cause an ejected rod to exceed the limit of 0.65% $\Delta k/k$ at RTP or 1.00% $\Delta k/k$ at zero power (Ref. 3). 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 duration of time that operation is expected to continue with a misaligned rod. 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.

A.2.2.3

Performance of SR 3.2.5.1 provides a determination of the local core LHRs using the Incore Detector System. Verification of the local core LHRs 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.

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."

B.1

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

C.1.1

More than one CONTROL ROD becoming inoperable or 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 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 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

If more than one 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. The allowed Completion Time of 6 hours is reasonable, based on operating experience, for reaching MODE 3 from RTP in an orderly manner and without challenging unit systems.

SURVEILLANCE REQUIREMENTS

SR 3.1.4.1

Verification that individual CONTROL 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. The specified Frequency takes into account other CONTROL ROD position information that is continuously available to the operator in the control room, so that during actual CONTROL ROD motion, deviations can immediately be detected.

SR 3.1.4.2

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 approximately 1.5% (approximately 2 inches) will not cause radial or axial power tilts, or oscillations, to occur. No additional allowances for instrument uncertainty are required to be incorporated in the implementing procedures for this parameter. 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 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 capable of being fully inserted, the CONTROL ROD(S) may continue to be considered OPERABLE unless inoperable for some other reason. At any time, if a CONTROL ROD(S) is immovable, a determination of the capability to fully insert (OPERABILITY) the CONTROL ROD(S) must be made, and appropriate action taken.

SR 3.1.4.3

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.

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

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 drop times are determined with less than four reactor coolant pumps operating, a Note allows operation to continue, provided operation is restricted to the pump combination utilized during the CONTROL ROD drop time determination or pump combinations providing less total reactor coolant flow.

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REFERENCES

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

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B 3.1 REACTIVITY CONTROL SYSTEMS

B 3.1.5 Safety Rod Insertion Limit

BASES

BACKGROUND

The insertion limits of the CONTROL RODS are initial condition assumptions in all safety analyses that assume CONTROL 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.

The applicable criteria for the reactivity and power distribution design requirements are 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).

Limits on safety rod insertion have been established, and all CONTROL ROD positions are monitored and controlled during operation in MODES 1 and 2 to ensure that the reactivity limits, ejected rod worth, and SDM limits are preserved.

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). In MODES 1 and 2, the regulating groups must be maintained above designated 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.

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. Prior to entry into MODE 2 from MODE 3, the safety groups must be fully withdrawn. The safety groups must be completely withdrawn from the core prior to withdrawing any regulating groups 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

On a reactor trip, all CONTROL RODS, 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 (see LCO 3.1.1, "SHUTDOWN MARGIN (SDM)") following a reactor trip from RTP. 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 the required SDM at rated no load temperature (Ref. 3).

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. 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).

In MODES 1 and 2 while critical, the safety rod insertion limits satisfy Criteria 2 and 3 of 10 CFR 50.36 (Ref. 5). In MODE 2 while subcritical, the safety rod insertion limits satisfy Criterion 4 of 10 CFR 50.36.

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LCO

The safety groups must be fully withdrawn any time the reactor is 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 achieve the required SDM following a reactor trip.

This LCO has been modified by a Note indicating the LCO requirement is suspended 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.

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APPLICABILITY

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 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.

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ACTIONS

A.1.1, A.1.2, and A.2

The safety rod must be declared inoperable within 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 safety rod may remain misaligned and not be providing its normal negative reactivity on tripping. 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.

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

B.1.1 and B.1.2

When more than one safety rod is not fully withdrawn, 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 as well as the CONTROL ROD of maximum worth.

B.2

If more than one safety rod is not fully withdrawn, 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 in an orderly manner and without challenging unit systems.

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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.

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.

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REFERENCES

1. SAR, Section 1.4, GDC 10, GDC 26, and GDC 28.

2. 10 CFR 50.46.

3. SAR, Chapters 3 and 4.

4. BAW-10179P-A, "Safety Criteria and Methodology for Acceptable Cycle Reload Analyses," Rev. 2.

5. 10 CFR 50.36.

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 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 GDC 10, "Reactor Design," and 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).

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 positions are monitored and controlled during power operation to ensure that the power distribution limits defined by the design peaking limits are preserved.

APSRs are moved by their control rod drive mechanisms (CRDMs). Each CRDM moves its rod 3/4 inch for one revolution of the leadscrew, but at varying rates depending on the signal output from the Control Rod Drive Control System (CRDCS).

The APSRs are arranged into 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.

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.

## APPLICABLE SAFETY ANALYSES

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.

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. Continued operation of the reactor with a misaligned APSR is allowed if Section 3.2, "Power Distribution Limits," are preserved.

Because ANO-1 uses gray APSRs, the APSR alignment limits satisfy Criterion 4 of 10 CFR 50.36 (Ref. 3).

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LCO

The limits on CONTROL ROD group alignment, safety rod 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% (approximately 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 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 APSR is not included in the calculation of the APSR group's average position.

Failure to meet the requirements of this LCO may produce unacceptable LHRs, which may constitute initial conditions inconsistent with the safety analysis.

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APPLICABILITY

The requirements on APSR 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 and alignment of APSRs have the potential to affect the safety of the unit. In MODES 3, 4, 5, and 6, the alignment limits do not apply because the reactor is shut down, and excessive local LHRs cannot occur from APSR misalignment.

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ACTIONS

A.1

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

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. 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, APSR group movement 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 the limits on power peaking are surveilled within 2 hours to determine if 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.

B.1

The 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 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 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.

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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. In addition, APSR position is continuously available to the operator in the control room so that during actual APSR motion, deviations can immediately be detected.

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REFERENCES

1. SAR, Section 1.4, GDC 10 and GDC 28.

2. 10 CFR 50.46.

3. 10 CFR 50.36.

B 3.1 REACTIVITY CONTROL SYSTEMS

B 3.1.7 Position Indicator Channels

BASES

BACKGROUND

According to the SAR discussion of GDC 13 (Ref. 1), adequate instrumentation and controls are provided to maintain operating variables within prescribed ranges for normal operation and monitor accident conditions as appropriate to assure adequate safety. 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.

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

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 local linear heat rates (LHRs), due to the asymmetric reactivity distribution, and a reduction in the total available CONTROL ROD worth for reactor shutdown. Therefore, CONTROL ROD and APSR alignment are related to core operation within design LHR limits and the core design requirement of a minimum SDM. CONTROL ROD and APSR position indication is needed to assess OPERABILITY and alignment.

Limits on CONTROL ROD and APSR alignment, and CONTROL ROD and APSR group position have been established, and all CONTROL ROD and APSR positions are monitored and controlled during operation to ensure that the power distribution and reactivity limits defined by the design LHR and SDM limits are preserved.

Three methods of CONTROL ROD and APSR position indication are provided in the Control Rod Drive Control System. The three means are by absolute position indicator, relative position indicator transducers, and zone reference indicators. 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. Switch contacts close when a permanent magnet mounted on the upper end of the CONTROL ROD or APSR assembly leadscrew extension comes near. As the leadscrew and CONTROL ROD or APSR move, the switches operate sequentially, producing an analog voltage proportional to position. 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. 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 and APSR position indicating readout devices located in the control room consist of single rod position meters on a position indication panel and group average position meters. A selector switch permits either relative or absolute position indication to be displayed on all of the individual position indication meters. Indicator lights are provided on the individual position indication panel to indicate when each CONTROL ROD or APSR is fully withdrawn, fully inserted, enabled, or transferred, and whether a rod position asymmetry alarm condition is present. Additional indicators show full insertion, full withdrawal, and enabled for motion for each CONTROL ROD and 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 and APSR position. Therefore, the potential for operation in violation of design LHR or SDM limits is increased.

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## APPLICABLE SAFETY ANALYSES

CONTROL ROD and APSR position accuracy is essential during power operation. LHR, 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. CONTROL ROD and APSR positions must be known in order to verify the core is operating within the group sequence, overlap, design LHRs, 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 CONTROL ROD and APSR 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 unit is operating within the bounds of the accident analysis assumptions.

In MODES 1 and 2 while critical, the CONTROL ROD and APSR position indicator channels satisfy Criterion 2 of 10 CFR 50.36 (Ref. 3). In MODE 2 while subcritical, the CONTROL ROD and APSR position indicator channels satisfy Criterion 4 of 10 CFR 50.36.

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## LCO

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

This requirement ensures that CONTROL ROD and APSR position indication during MODES 1 and 2 and PHYSICS TESTS is accurate, and that design assumptions are not challenged. OPERABILITY of the position indicator channel ensures that inoperable, misaligned, or mispositioned CONTROL RODS or APSRs can be detected. Therefore, LHR and SDM can be controlled within acceptable limits.

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APPLICABILITY

In MODES 1 and 2, OPERABILITY of the position indicator channel 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 significant THERMAL POWER.

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ACTIONS

A.1

If the required position indicator channel is inoperable for one or more rods, the position of the CONTROL ROD or APSR is not known with certainty. Therefore, each affected CONTROL ROD or APSR 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.

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SURVEILLANCE REQUIREMENTS

SR 3.1.7.1

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 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.

The required Frequency of 12 hours is adequate for verifying that no degradation in system OPERABILITY has occurred.

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.

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REFERENCES

1. SAR, Section 1.4, GDC 13.

2. SAR, Chapter 14.

3. 10 CFR 50.36.

B 3.1 REACTIVITY CONTROL SYSTEMS

B 3.1.8 PHYSICS TESTS Exceptions Systems - MODE 1

BASES

BACKGROUND

The purpose of this 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. All functions necessary to ensure that specified design conditions are not violated during normal operation and 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).

The key objectives of a test program are to:

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. 3).

The inclusion of this PHYSICS TESTS Exception LCO is acceptable based on the use of approved written procedures, administrative controls, the requirements of 10 CFR 50.59, and the LCO 3.1.8 provisions in effect during the conduct of PHYSICS TESTS. PHYSICS TESTS procedures are written and approved in accordance with established guidelines. The procedures include all information necessary to permit a detailed 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.

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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), ejected rod worth, and shutdown capability are maintained during the PHYSICS TESTS.

Reference 4 describes the initial testing of the facility, including PHYSICS TESTS. Table 13-2 (Ref. 5) summarizes the post-criticality tests. Requirements for reload fuel cycle PHYSICS TESTS are given in SAR Section 3A.9 (Ref. 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.

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";
LCO 3.2.2, "AXIAL POWER SHAPING ROD (APSR) Insertion Limits," for the restricted operation region only;
LCO 3.2.3, "AXIAL POWER IMBALANCE Operating Limits"; and
LCO 3.2.4, "QUADRANT POWER TILT (QPT)"

are suspended for PHYSICS TESTS, the fuel design criteria are preserved by maintaining the LHR (in MODE 1 PHYSICS TESTS) within 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 within the limit provided in the COLR. Therefore, surveillance of the LHR and SDM is required to verify that their limits are not exceeded. The limits for the LHR are specified in the COLR. Refer to the Bases for LCO 3.2.5 for a complete discussion of LHR. During PHYSICS TESTS, one or more of the LCOs that normally preserve the LHR limits may be suspended. However, the results of the safety analysis are not adversely impacted if verification that core LHRs are within their limits is obtained, while one or more of the LCOs is suspended. Therefore, SRs are placed on LHR during MODE 1 PHYSICS TESTS when THERMAL POWER exceeds 20% RTP to verify that the core LHRs remain within their limits. Periodic verification of these factors allows PHYSICS TESTS to be conducted while continuing to maintain the design criteria.

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. The limits for these variables are specified for each fuel cycle in the COLR.

As described in LCO 3.0.7, compliance with Test Exception LCOs is optional, and therefore no criteria of 10 CFR 50.36 (Ref. 6) apply. Test Exception LCOs provide flexibility to perform certain operations by appropriately modifying requirements of other LCOs. A discussion for the other LCOs is provided in their respective Bases.

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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, 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.

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:

a. THERMAL POWER is maintained ≤ 85% RTP;

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;

c. LHR is maintained within limits specified in the COLR while operating at greater than 20% RTP; and

d. SDM is verified to be within the limit provided in the COLR.

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 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.

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."

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 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.

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 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.

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.

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 LHR has exceeded its limit, then PHYSICS TESTS are suspended. This action is required because of direct indication that the core LHR, which is a fundamental initial condition for the safety analysis, is excessive. Suspension of PHYSICS TESTS exceptions requires restoration of each of the applicable LCOs to within specification.

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."

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

SR 3.1.8.2

Verification that core LHRs are within their limits ensures that core 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 LHR verification, 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 LHR limits.

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."

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 during the PHYSICS TESTS. Performing the verification once within 8 hours prior to the performance of PHYSICS TESTS at each testing plateau allows the operator adequate time for verifying the established trip setpoint before initiating PHYSICS TESTS.

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. Doppler defect;

d. Fuel burnup based on gross thermal energy generation;

e. Samarium concentration;

f. Xenon concentration; and

g. Moderator defect.

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. SAR, Section 3A.9.

4. SAR, Section 13.3, 13.4 and 13.6.

5. SAR, Section 13.4, Table 13-2.

6. 10 CFR 50.36.

B 3.1 REACTIVITY CONTROL SYSTEMS

B 3.1.9 PHYSICS TESTS Exceptions - MODE 2

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. All functions necessary to ensure that specified design conditions are not violated during normal operation and 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).

The key objectives of a test program are to:

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. 3).

The inclusion of this PHYSICS TESTS Exception LCO is acceptable based on the use of approved written procedures, administrative controls, the requirements of 10 CFR 50.59, and the LCO 3.1.9 provisions in effect during the conduct of PHYSICS TESTS. PHYSICS TESTS procedures are written and approved in accordance with established guidelines. The procedures include all information necessary to permit a detailed 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.

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APPLICABLE SAFETY ANALYSES

Reference 4 describes the initial testing of the facility, including PHYSICS TESTS. Table 13-2 (Ref. 5) summarizes the post-criticality tests. Requirements for reload fuel cycle PHYSICS TESTS are given in SAR Section 3A.9 (Ref. 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.

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";
LCO 3.2.2, "AXIAL POWER SHAPING ROD (APSR) Insertion Limits; and
LCO 3.4.2, "RCS Minimum Temperature for Criticality."

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 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 unit control. The narrow range temperature instrumentation goes on scale at 520°F. Therefore, it is considered safe to allow the minimum RCS 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.

As described in LCO 3.0.7, compliance with Test Exception LCOs is optional, and therefore no criteria of 10 CFR 50.36 (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.

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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.

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:

a. THERMAL POWER is ≤ 5% RTP;

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

c. Nuclear instrumentation high startup rate CONTROL ROD withdrawal inhibit is OPERABLE; and

d. SDM is within the limit provided in the COLR.

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.

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APPLICABILITY

This LCO is applicable when the reactor is either subcritical or critical with THERMAL POWER ≤ 5% RTP. 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. This LCO is applicable for initial criticality or low power testing, as described in SAR Section 3A.9 (Ref. 3). In MODE 1, 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 is not required because the excepted LCOs do not apply in these MODES.

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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).

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 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.

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.

C.1

If the nuclear overpower trip setpoint is > 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 Time is consistent with, or more conservative than, those specified for the individual LCOs addressed by PHYSICS TESTS exceptions.

If the nuclear instrumentation high startup rate CONTROL ROD withdrawal inhibit function is 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.

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.

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SURVEILLANCE REQUIREMENTS

SR 3.1.9.1

Verification that THERMAL POWER is ≤ 5% RTP ensures that 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.

SR 3.1.9.2

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 during PHYSICS TESTS. Performing the verification once within 8 hours prior to the performance of PHYSICS TESTS allows the operator adequate time for verifying the established trip setpoint before initiating PHYSICS TESTS.

SR 3.1.9.3

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;

f. Xenon concentration;

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 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.

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.

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REFERENCES

1. 10 CFR 50, Appendix B, Section XI.
2. 10 CFR 50.59.
3. SAR, Section 3A.9.
4. SAR, Section 13.3, 13.4 and 13.6.
5. SAR, Section 13.4, Table 13-2.
6. 10 CFR 50.36.

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# CTS DISCUSSION OF CHANGES

## ITS Section 3.1: Reactivity Control Systems

### ADMINISTRATIVE

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

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

A3 The CTS 4.7.1.2 defined rod misalignment as being a deviation from the group average position of more than nine (9) inches. For consistency with the plant instrumentation and NUREG-1430, 6.5% will be used to establish CONTROL ROD and APSR misalignment in the ITS. ITS Bases B 3.1.4 includes reference to the fact that 9 inches and 6.5% are considered equivalent. This is consistent with NUREG-1430.

A4 Not used.

A5 The second statement of CTS 3.5.2.5.1 provides an exception to the requirement that all safety rods be fully withdrawn as stated in CTS 3.1.3.5. This allowance relaxes the requirement to shutdown, per CTS 3.1.3.7, when a safety rod is not fully withdrawn, provided the rod is inoperable per CTS 3.5.2.2. Through the adoption of ITS 3.1.5 and its associated ACTIONS, this allowance for continued operation of the unit with an inoperable and not fully withdrawn safety rod will be maintained. Although it is represented in a significantly different format, the requirements of CTS 3.5.2.5.1 are maintained by the requirements of the ITS. Due to the continuation of essentially equivalent requirements, this change is administrative in nature. This change is consistent with NUREG-1430.

A6 The requirement that a CONTROL ROD which cannot be exercised be declared inoperable, which is presented in the first statement in CTS 4.7.1.3, is maintained in the ITS through the requirements of ITS SR 3.1.4.2, CONTROL ROD freedom of movement verification, and the application of ITS SR 3.0.1. Although no specific ITS item is cross-referenced to this CTS item, the requirement is embodied in the structure and requirements of ITS Specifications 3.1.4 and 3.1.5, and the application of SR 3.0.1. The lack of a direct cross-reference represents no actual change in requirements and is administrative in nature.

A7 CTS 3.1.3.1 establishes the minimum temperature for criticality of 525°F except during low power physics testing when the requirements of CTS 3.1.8.3 shall apply. CTS 3.1.3.2 and CTS 3.1.8.3 establish a minimum temperature for criticality in accordance with the criticality curves provided on CTS Figure 3.1.2-2. CTS 3.1.3.2 and CTS 3.1.8.3 implicitly duplicate the requirements of CTS 3.1.2, "Pressurization, Heatup and Cooldown Limitations," which has an implied Applicability of "at all times." Because of the duplicative nature of CTS 3.1.3.2 and CTS 3.1.8.3, they have been administratively deleted. This is acceptable because these minimum temperature requirements will exist in ITS LCO 3.4.3, "RCS Pressure and Temperature (P/T) Limits." ITS 3.4.3 will have Applicability "at all times" and is not excepted by the Physics Testing exceptions provided by LCO 3.1.8, "PHYSICS TEST Exceptions - MODE 1." and LCO 3.1.9, "PHYSICS TEST Exceptions - MODE 2." Therefore, this minimum temperature for criticality requirement will continue to exist in the ITS.

A8 The intent of CTS 3.1.8.1.A and 3.1.8.1.B is to ensure that, during Low Power Physics Testing, all Reactor Protection System (RPS) Setpoints are maintained per the requirements of the RPS setpoints section of CTS (Table 2.3-1) with the exception of the nuclear overpower trip setpoint which shall be less than 5 percent. The distinction of specifying the requirements separately below 1720 psig and above 1800 psig is made to ensure that the requirements are clearly applicable whether RPS is in Shutdown Bypass (<1720 psig), or out of Shutdown Bypass (>1800 psig). The requirement to maintain the nuclear overpower trip setpoint at less than 5 percent is specified only when above 1800 psig because the Shutdown Bypass nuclear overpower trip setpoint specified in CTS Table 2.3-1 is also 5%. The adoption of ITS 3.1.9 and its Applicability will maintain requirements consistent with those found in CTS 3.1.8.1.A and 3.1.8.1.B. Since ITS 3.1.9 does not suspend the requirements of ITS 3.3.1, "Reactor Protection System (RPS) Instrumentation," it is clear that all applicable RPS setpoint requirements of ITS Table 3.3.1-1 apply even during MODE 2 PHYSICS TESTING. Additionally, ITS 3.1.9 provides the requirement that the "Reactor trip setpoints on the OPERABLE nuclear overpower channels are set to ≤ 5% RTP." This maintains a reactor trip setpoint requirement consistent with CTS 3.1.8.1.B. Finally, by allowing RPS overpower trip setpoints no higher than 5% RTP, CTS requirements ensured that this testing was performed at less than 5% RTP. The specified setpoints maintain requirements consistent with ITS 3.1.9.a.

Because the adoption of ITS 3.1.9, in lieu of CTS 3.1.8.1.A and 3.1.8.1.B, though significantly different in format, maintains requirements consistent with CTS 3.1.8.1.A and 3.1.8.1.B, this change is administrative in nature. This change does not result in any new requirements nor does it result in the removal of any current requirements.

A9 CTS 3.5.2.3 established a requirement that "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." CTS 3.1.3.5 established requirements for safety rod withdrawal and limitations on regulating rod group insertion as established by Specification 3.5.2.5. The CTS did not explicitly establish a required action to verify that the potential ejected rod worth of a misaligned rod is within the assumptions used in the rod ejection analyses. However, it is an implicit requirement that CTS 3.5.2.3 would apply to misaligned CONTROL RODS. Therefore, CTS 3.5.2.3 is considered to embody the requirements of NUREG-1430 Required Action A.2.4 (ITS Required Action A.2.2.2).

# CTS DISCUSSION OF CHANGES

A10 CTS 3.1.3.5 requires that the safety rod groups be fully withdrawn prior to any other reduction in shutdown margin by deboration or regulating rod withdrawal during the approach to criticality. NUREG-1430 and ITS LCO 3.1.5 require that each safety rod be fully withdrawn during MODES 1 and 2. The NUREG and ITS are predicated on an "individual" rod basis and not a group position basis. Although this translates into an identical requirement to have all safety rods fully withdrawn in MODES 1 and 2, there will be no safety rod group position requirements or actions in the ITS, only individual safety rod requirements and actions. This change in presentation of requirements is considered administrative in nature and does not change the actual requirement that all safety rods be fully withdrawn during MODES 1 and 2. This change is consistent with NUREG-1430.

The Applicability for CTS 3.1.3.5 is "prior to any other reduction in shutdown margin by deboration or regulating rod withdrawal during the approach to criticality." This statement, as applied at ANO-1, requires compliance with regulating rod insertion limits while in Hot Standby and Startup reactor operating conditions (equivalent to ITS MODE 2). Although not explicitly applied to Power Operations (MODE 1), this Specification must be applied during these conditions to preserve the SHUTDOWN MARGIN requirements. Because the Applicability of ITS 3.1.5 maintains requirements consistent with the Applicability of CTS 3.1.3.5, as applied at ANO-1, this change is administrative in nature and neither adds any additional requirements nor removes any existing requirements.

A11 CTS 4.7.1.2 requires that if a CONTROL ROD is misaligned from its group average position by more than 9 inches (6.5%), it shall be declared inoperable and the limits of CTS 3.5.2.2 shall apply. CTS 3.5.2.2 includes some actions which are applicable to all inoperable CONTROL RODS and some actions which are specifically applicable only to CONTROL RODS which are inoperable due to misalignment. Although ITS 3.1.4 and 3.1.6 differentiate between inoperable and misaligned rods, these Specifications are written in such a way as to provide appropriate actions to compensate for either case. (The specific discussion of the differences between the actions of CTS 3.5.2.2 and ITS 3.1.4 and 3.1.6 are contained in separate DOCs.) Through the adoption of ITS 3.1.4 and 3.1.6, the intent of CTS 4.7.1.2 which is to ensure that the appropriate actions are taken in the event that a CONTROL ROD or APSR becomes misaligned from its group average position is maintained. No new requirements are added by this change and the only requirement removed is the requirement to declare the misaligned rod inoperable based only on misalignment. This difference is a result of the difference in philosophy of implementation between the CTS and ITS. Therefore, this change is considered administrative and represents no significant change to the requirements for operating with a misaligned rod.

A12 CTS markup was annotated to show adoption of ITS 3.1.7 Actions Note. This change is administrative in that the Note is required by the format and usage associated with the structure and presentation of the Actions in NUREG-1430.

3.1-07 A13 The CTS 4.7.1.2 provision that allowed the CONTROL ROD with the greatest deviation from the group average position to be evaluated first for the purpose of determining compliance with CTS requirements has been shown as administratively deleted. This allowance is not contained within nor does it support the requirements of NUREG-1430 or the ITS. CTS 3.5.2.2.1 does not allow operation with more than one control rod misaligned by more than nine inches from the group average. The deletion of this CTS allowance is acceptable because of the conservative nature of the ITS in addressing multiple CONTROL ROD deviations from their group average position. This change is consistent with NUREG-1430.

## TECHNICAL CHANGE -- MORE RESTRICTIVE

M1 CTS Specification 4.9 currently provides for the evaluation of reactivity anomalies during operation of the unit. The CTS requires that the reactivity anomaly be evaluated "periodically" by comparison of the actual boron concentration to the predicted boron concentration. Additional discussion of the process of anomaly determination is provided in the Bases of CTS Specification 4.9. This periodic evaluation is presently administratively controlled with a frequency of approximately once per month. Adoption of the NUREG-1430 Specification 3.1.2 will require that the Frequency be performed in accordance with a more restrictive schedule than that presently identified in the CTS. Specifically, ITS SR 3.1.2.1 will have a Frequency of "prior to entering MODE 1 after each fuel loading" and "31 EFPD thereafter" following 60 EFPD of cycle operation as established in the Note. These SR Frequencies are acceptable because they explicitly establish the time frame for the performance of the SR and are in accordance with current administrative practices. This change is consistent with NUREG-1430.

M2 CTS 4.9 provides for the evaluation of reactivity anomalies during operation of the unit. The CTS action requires that the reactivity anomaly be evaluated to determine the cause. No other specific power reduction or operating restriction is applied. ANO will adopt the NUREG-1430 LCO 3.1.2 ACTIONS with a specified Completion Time of 7 days for Condition A. This Required Action is more restrictive than the requirements established within the CTS. This change is appropriate because the Required Actions preserve the assumptions used in the accident analyses through the implementation of appropriate operating restrictions. This change is consistent with NUREG-1430.

M3 Not used.

M4 CTS 3.1.7.1 establishes the limits on Moderator Temperature Coefficient (MTC). The CTS states that the limits are applicable when "the reactor is not shutdown." The interpretation of this statement represents a condition where the reactor would be made 1% $\Delta K/K$ subcritical which represents a condition consistent with the CTS definition for Hot Shutdown. The slightly more restrictive Applicability of MODES 1 and 2 in ITS LCO 3.1.3 will provide requirements on MTC that are consistent with other reactivity control parameters in the ITS. This change is classified as slightly more restrictive due to the slight calculational difference that exists between a reactor shutdown by 1% $\Delta K/K$ and a reactor that has $K_{eff}$ of less than or equal to 0.99. This change is consistent with NUREG-1430.

M5 CTS 3.5.2, "Control Rod Group and Power Distribution Limits," has a defined Applicability of "during power operation." However, these CONTROL ROD OPERABILITY requirements are in practice applied during both CTS Power Operation and Hot Standby operating conditions. The CONTROL ROD OPERABILITY criteria defined by CTS 3.5.2 will correlate with requirements in ITS 3.1.4, 3.1.5, 3.1.6, 3.2.1 and 3.2.2. All of these ITS Specifications have an Applicability of MODES 1 and 2. By specifying Applicability in MODE 2, in addition to MODE 1, requirements will exist in the ITS where none were previously specified in the CTS. This Applicability represents more restrictive operating requirements than those specified in the CTS. This change is necessary to ensure that CONTROL ROD OPERABILITY exists in MODES that are consistent with the ITS SHUTDOWN MARGIN requirements preserved by the CONTROL ROD alignment and positioning. This change is consistent with NUREG-1430.

M6 The requirements of NUREG-1430 LCO 3.1.1, "SHUTDOWN MARGIN (SDM)," will be adopted as presented in ITS. No explicit requirements for SDM, as defined in ITS Section 1.1, at other than power operation conditions, exist in the CTS. When the RCS temperature was below the minimum temperature for criticality given in CTS 3.1.3.1, CTS 3.1.3.3 required a degree of subcriticality, based on the reactivity effect of depressurization, be maintained. In addition, there are subcriticality requirements contained in the CTS Section 1.0 definitions of Hot Shutdown, Cold Shutdown, and Refueling Shutdown. Adoption of ITS 3.1.1 is more restrictive in that specific LCO requirements, Required Actions, and Surveillance Requirements are established which were not previously, explicitly required in the CTS. This change is necessary to ensure that controls and compensatory measures are in place during MODES 3, 4, and 5 that ensure the subcriticality of the unit is maintained. This change is consistent with NUREG-1430.

M7 CTS 3.5.2.2.1 states "Operation with more than one inoperable rod ... shall not be permitted." The lack of a specified action time implies that CTS 3.0.3 applies. CTS 3.0.3 requires the unit to be in Hot Shutdown (ITS MODE 3) in 13 hours. The equivalent action established in NUREG-1430, LCO 3.1.4 Required Action C.2 and LCO 3.1.5 Required Action B.2, requires the unit to be in MODE 3 within 6 hours. ANO-1 will adopt these more restrictive requirements in order to provide explicit Completion Times where none are currently expressed. This change is consistent with NUREG-1430.

M8 The CTS requirement for performance of CONTROL ROD drop time testing is, per CTS 4.7.1.1, "following each refueling outage prior to return to power" and in Table 4.1-2 Item 1, "Each Refueling Shutdown." The NUREG-1430 SR 3.1.4.3 Frequency of "Prior to reactor criticality after each removal of the reactor vessel head" will be adopted to provide a test Frequency consistent with activities that have the potential of affecting the rod drop time. This change in Frequency imposes the additional requirement of performing CONTROL ROD drop time testing following any removal of the reactor vessel head not just following a refueling shutdown or outage. It additionally requires completion of this testing prior to criticality rather than "prior to return to power." Adoption of the ITS SR 3.1.4.3 Frequency is appropriate because it correlates the SR Frequency to the activity that has the greatest probability of affecting the CONTROL ROD capability and characteristics. This change is consistent with NUREG-1430.

M9 CTS 3.5.2.2.5 correlates to ITS 3.1.4 Required Action A.2.2.1. CTS 3.5.2.2.5 requires a reduction in power while operating with a misaligned CONTROL ROD; however, there is no specified Completion Time. ITS 3.1.4 Required Action A.2.2.1 similarly requires a reduction in THERMAL POWER, while operating with a misaligned CONTROL ROD, and includes the added restriction of a 2 hour Completion Time. The adoption of the Completion Time ensures conservative actions are expeditiously initiated to minimize the potential effects of power redistribution and subsequent power peaking. This change is consistent with NUREG-1430.

M10 The first two sentences of CTS 3.5.2.2.2 and the first sentence of CTS 3.5.2.2.3 correlate to ITS 3.1.4 Required Actions A.1.1, A.1.2, C.1.1, and C.1.2 with the exception of the second specified Completion Time for Required Action A.1.1. Therefore, the second Completion Time for ITS 3.1.4 Required Action A.1.1 is shown as being adopted. This addition will impose more stringent requirements on unit operation by specifying that SDM be verified on a 12 hour Frequency after the initial verification. While this is not a departure from current operating practices, it is an additional requirement not given in the CTS. This periodic verification of SDM is appropriate because of the potential effects associated with power level changes, power redistribution, and transient fission product poisons. This change is consistent with NUREG-1430.

M11 ITS SR 3.1.4.1, SR 3.1.5.1 and SR 3.1.6.1 requirements to verify that CONTROL RODS and APSRs are within 6.5% of their group average and that safety rods are fully withdrawn, on a 12 hour Frequency, has been adopted. No specific requirement for this verification is expressed in CTS. Current operating practice is to perform these verifications in conjunction with and on the same frequency as the check of the Absolute and Relative Position Indication instrumentation. This change is consistent with NUREG-1430.

M12 CTS 3.1.7.3 currently requires the unit to be placed "in at least HOT STANDBY" (reactor critical below 2% power) if the Moderator Temperature Coefficient (MTC) is outside its limits. The adoption of ITS 3.1.3 ACTION A will require the unit to be placed in MODE 3 if MTC is outside its limits. This conservative action is consistent with other ITS reactivity control Specifications and removes the unit from the Applicability established for ITS 3.1.3. This change is consistent with NUREG-1430.

M13 ITS 3.1.7 Applicability has been adopted. No explicit Applicability exists for the equivalent requirements found in CTS 4.7.1.3. The addition of the ITS 3.1.7 MODE 1 and 2 Applicability has been made to provide requirements for verification of CONTROL ROD and APSR position indication that are consistent with ITS LCO 3.1.4, 3.1.5, 3.1.6, 3.2.1 and 3.2.2 requirements governing CONTROL ROD positioning. This change is consistent with NUREG-1430.

M14 The CTS markup reflects the adoption of NUREG-1430 LCO 3.1.8 PHYSICS TESTS Exceptions - MODE 1 as it is presented in the ITS. The CTS excepted certain individual specifications with a statement such as "except for physics testing." [This is one frequent usage of the exception and is not intended to represent every usage of the exception in the CTS.] No differentiation was made in the CTS of the applicability of these exceptions with respect to the unit's THERMAL POWER level. Further, only a minimal number of specific requirements were presented in the CTS during the conduct of PHYSICS TESTS and no required actions were presented. ITS 3.1.8 LCO, ACTIONS and SRs have been shown as adopted to provide this power level (or MODE) dependency. Although the PHYSICS TEST exceptions existed in the CTS, the power level dependency did not exist. Thus, the ITS will result in more restrictive requirements. This change is consistent with NUREG-1430.

Additionally, the ACTIONS and SRs of ITS 3.1.9 PHYSICS TEST Exceptions-MODE 2 have been adopted. These items function to verify that the LCO requirements are satisfied and provide necessary remedial actions should the requirements not be satisfied. Because the CTS did not impose specific restrictions, required actions or additional surveillance requirements comparable to those established in the ITS, this change is more restrictive. The adoption of the additional requirements, Required Actions and SRs is appropriate due to the nature of PHYSICS TESTS. This change is consistent with NUREG-1430.

M15 ITS 3.1.2 Required Action A.2 and Required Action B.1 will be adopted. The Frequency of ITS SR 3.1.2.1 and the Notes modifying this SR are also adopted. The adoption of these requirements, where none existed previously, represents more restrictive requirements on the unit. These Required Actions provide appropriate guidance for continued unit operation with a reactivity anomaly that exceeds its limit and conservative action to place the unit in MODE 2 should the Required Actions and associated Completion Times of Condition A not be met. The SR Notes are necessary to provide guidance for completion of the SR. The SR Frequency adopted is appropriate to determine the presence of a reactivity anomaly shortly after unit startup but prior to significant unit operation with the anomalous condition. The adoption of the SR Frequency is specifically more restrictive because it specifies the performance of the SR "once prior to entering MODE 1 after each fuel loading." This change is consistent with NUREG-1430.

M16 The 72 hour Completion Time for ITS 3.1.4 Required Action A.2.2.2 (NUREG-1430 3.1.4 Required Action A.2.4) is shown on the CTS markup as being adopted in the ITS. This is more restrictive because no Completion Time was explicitly established in the CTS for the completion of ejected rod worth verification as required by CTS 3.5.2.3. The adoption of the Completion Time is appropriate to ensure that the verification is promptly initiated; thus, allowing implementation of compensatory measures, if appropriate. This change is consistent with NUREG-1430.

M17 The "no flow" rod drop time testing acceptance criteria is shown as being administratively deleted in the CTS 4.7.1.1 markup. This acceptance criteria and the conditions of the testing have not been demonstrated as being acceptable for satisfying the rod drop time surveillances that preserve the accident analysis assumptions. This allowance and its test criteria are not currently utilized by ANO-1. In fact SAR Section 3.A, does not allow completion of startup testing and entrance into MODE 1 without performing the full flow test. The deletion of this allowance from the CTS results in the ITS possessing more restrictive requirements than those established by the CTS. NUREG-1430 does not establish a similar "no flow" testing methodology or acceptance criteria, thus, this deletion of material is consistent with NUREG-1430.

3.1-07 M18 Not used.

M19 The CTS was annotated to show the adoption of ITS 3.1.4 Required Action A.2.2.3 with its Note (NUREG-1430 3.1.4 Required Action A.2.5) which will require verification of acceptable core linear heat rates (LHRs) during operation at less than or equal to 60% of the ALLOWABLE THERMAL POWER with a misaligned CONTROL ROD. This Required Action has a 72 hour Completion Time which is acceptable because core LHRs are limited by the THERMAL POWER reduction (ITS 3.1.4 Required Action A.2.2.1). The Required Action is preceded by a Note that specifies the Required Action is only required to be performed when THERMAL POWER is greater than 20% RTP. This establishes a requirement for verification of core power distribution during unit operation consistent with the OPERABILITY of the incore detector system. This change is consistent with NUREG-1430.

# CTS DISCUSSION OF CHANGES

M20 The methodology specified in CTS 3.5.2.2.2 for restoring SDM, if it is determined to be less than adequate, allows boration to be secured once the worth of the inoperable rod has been met or once the limits of CTS 3.5.2.5.3 are met (i.e., the regulating rod groups are withdrawn above the SDM insertion limit curve given in the COLR). The ITS requirement will be that SDM be calculated and verified to be within the limit specified in the COLR taking into consideration the reactivity worth of the inoperable CONTROL ROD. Therefore, when addressing a single inoperable CONTROL ROD, the ITS will not allow boration to be secured once the regulating groups have been positioned above the SDM limits established by the regulating rod insertion curves given in the COLR. [Note: this discussion does not impact other CTS and ITS Specifications that would require continued boration should the regulating groups be inserted beyond their SDM insertion limits.] Thus, the ITS will be more restrictive because it will exclude an option for compliance that is present in the CTS. The ITS method of SDM verification is consistent with current operating practices, though not specified by CTS. The adoption of the ITS requirements is appropriate because the regulating rod group insertion limits curve given in the COLR was not derived such that SDM was preserved with an additional inoperable rod, nor is it intended to address this condition. This change is consistent with NUREG-1430.

M21 CTS 3.5.2.2.3 requires the unit to be placed in Hot Standby (reactor critical and <2% power) if the required SHUTDOWN MARGIN (SDM) can not be verified or obtained within 1 hour. The CTS does not establish a specific completion time for this required action. The adoption of ITS 3.1.4 ACTION B will require the unit be placed in MODE 3 (i.e., $K_{eff} < 0.99$) within 6 hours if adequate SDM is not verified within one hour or if boration is not initiated to obtain SDM within one hour. Thus, ITS 3.1.4 ACTION B is more restrictive than the corresponding CTS requirement in that it requires the unit be taken to a lower MODE as a result of failure to satisfy SDM requirements. These additional requirements are necessary to remove the unit from an operating condition when boration has been inadequate to restore the necessary SDM. This change is consistent with NUREG-1430.

M22 CTS 3.5.2.2.3 correlates to ITS 3.1.4 Required Action B.1. CTS 3.5.2.2.3 requires that the unit be placed in Hot Standby if the preceding CTS actions have been unsuccessful in restoring the required SDM. The CTS does not specify a Completion Time. ITS 3.1.4 Required Action B.1 similarly addresses the Required Actions should the preceding ITS actions not be successfully implemented, and includes the added restriction of a 6 hour Completion Time. The adoption of the Completion Time ensures conservative actions are initiated to remove the unit from the LCO Applicability. This change is consistent with NUREG-1430.

M23 CTS 3.5.2.2.6 correlates to ITS 3.1.4 Required Action A.2.1. These Specifications allow the unit to continue to operate at unrestricted power levels above 60% ATP provided the inoperable regulating rod can be positioned such that it is contained within the allowable group alignment limits and the associated group positioned within the allowed group insertion limits. The CTS does not specify a Completion Time for this action. However, ITS 3.1.4 Required Action A.2.1 includes the added restriction of a 2 hour Completion Time. The adoption of the Completion Time ensures conservative actions are initiated to minimize the potential affects of power redistribution and subsequent power peaking. This change is consistent with NUREG-1430.

M24 CTS 3.5.2.2.6 correlates to ITS 3.1.6 Required Action A. 1. These Specifications allow the unit to continue to operate at unrestricted power levels above 60% ATP provided the inoperable APSR can be positioned such that it is contained within the allowable group alignment limits. The CTS does not specify a Completion Time for this action. However, ITS 3.1.6 Required Action A.1 includes the added restriction of a 2 hour Completion Time. The adoption of the Completion Time ensures conservative actions are initiated to minimize the potential affects of power redistribution and subsequent power peaking. This change is consistent with NUREG-1430.

M25 CTS 3.5.2.2.6 specifies that operation above 60% of ALLOWABLE THERMAL POWER (ATP) may continue with an APSR inoperable due to misalignment (as established by CTS 4.7.1.2) if the group is positioned such that the rod is no longer misaligned. This action restores compliance with the LCO; thus, no further action is required and power operation is unrestricted. The CTS establishes no required action if the unit is below 60% ATP. Further, the CTS does not specifically state the required action should an APSR not be capable of being aligned within its group alignment limits. The ITS will require THERMAL POWER to be reduced to ≤ 60% of the ALLOWABLE THERMAL POWER with a Completion Time of 2 hours. This change will incorporate an action that is implied by the current license basis.

3.1-10 M26 The required actions of CTS 3.5.2.2.6 do not specify a time limit for the completion of the required actions in the event of an inoperable or misaligned APSR, as discussed in DOC-M25 and DOC-M26. The Required Actions of ITS 3.1.6 Condition B provide guidance to ensure the unit is placed in a safe condition in the event the Required Actions and associated Completion Times of ITS 3.1.6 Condition A are not met. This change is consistent with NUREG-1430.

3.1-10 M27 CTS 3.5.2.2.6 states that with a rod in the axial power shaping group declared inoperable, 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 the allowable group average position limits. No time limit is provided for the implied reduction of power to less than 60 percent of the allowable power level. ITS 3.1.6 Required Action A.1 provides a requirement to perform SR 3.2.5.1 within 2 hours and within 2 hours after each APSR movement. The performance of this SR is intended to assure that power peaking factors are within the appropriate limits with a misaligned or inoperable APSR. Requiring the performance of this SR within the specified Completion Time is considered to be a more restrictive requirement in that although the CTS would require a power reduction, there is no time limit specified for the completion of this action. This change is consistent with NUREG-1430, as modified by TSTF-220, as modified by a generic change currently be tracked as ANO-1-063.

# CTS DISCUSSION OF CHANGES

## TECHNICAL CHANGE -- LESS RESTRICTIVE

L1 The ITS SR 3.1.4.2 required Frequency is less restrictive than the CTS. CTS Table 4.1-2 Item 2 requires movement of CONTROL RODS on a frequency of every two (2) weeks. The ITS Frequency will be 92 days. Based on the historical operating reliability of the CONTROL RODS, this change in Frequency from 14 days to 92 days is not considered to represent a significant reduction in the ability to verify system reliability. This position is supported by Generic Letter 93-05, "Line-Item Technical Specifications Improvements to Reduce Surveillance Requirements for Testing During Power Operation." The reduction in Frequency of CONTROL ROD freedom of movement verification lessens the overall number of CONTROL ROD drive system manipulations (power supply transfers, safety rod movement, etc.) and thereby tends to lessen the overall likelihood of dropped CONTROL RODS which can occur due to failures of portions of the control rod drive system. Though not easily quantifiable, the reduction in the overall likelihood of producing a dropped CONTROL ROD, specifically those caused by a system failure during testing, represents an overall increase in the safety of the unit. This change is consistent with NUREG-1430.

L2 ITS SR 3.1.4.3 will be adopted in place of CTS 4.7.1.1. The adoption of ITS SR 3.1.4.3, including its NOTE, provides ANO-1 with the additional flexibility of testing CONTROL ROD drop times with reactor coolant flow conditions other than full flow and no flow. By restricting operation of the unit to the reactor coolant pump combination used during rod drop testing, reactor coolant flow conditions, in the event of a reactor trip, are assured to be similar to those during CONTROL ROD drop time testing and thereby the testing is bounding. This change is consistent with NUREG-1430.

L3 Testing to insure freedom of movement of "Each Rod" is required above Cold Shutdown by CTS Table 4.1-2, Item 2. This testing is currently applied to both the CONTROL RODS and APSRs. Similar testing of the CONTROL RODS only, will be required by ITS SR 3.1.4.2 and will be applicable only in MODES 1 and 2. The adoption of the NUREG-1430 SR will result in less restrictive requirements. Specifically, the adoption of ITS SR 3.1.4.2 will remove the CTS requirement to perform freedom of movement testing on the APSRs. The purpose of this testing is to ensure that CONTROL RODS are not mechanically bound and will therefore insert upon a reactor trip. Because the APSRs, by design, do not insert upon a reactor trip, this testing is not required on the APSRs. Further, the APSRs are not credited as providing any of the required SDM on a reactor trip. This change is consistent with NUREG-1430.

L4 The CTS 3.5.2.2.2 and 3.5.2.2.4 requirements to exercise the remaining CONTROL RODS, in the event that a CONTROL ROD is declared inoperable, have been removed to improve the consistency between NUREG-1430 and ITS. The intent of these requirements was to provide for testing which could detect if additional CONTROL ROD(S) were immovable. Industry experience indicates that CONTROL ROD movement testing has in only a limited number of cases, led to the determination that a CONTROL ROD was mechanically immovable. This determination that a CONTROL

ROD is mechanically immovable is instead much more likely to be made during initial CONTROL ROD withdrawal or during drop time testing. By design, electrical problems which prevent movement of CONTROL RODS, generally, do not prevent the insertion of CONTROL RODS in the event of a reactor trip. Additionally, industry experience indicates that this testing can and has resulted in reactor trips and dropped rods. The relatively low likelihood that this testing will actually reveal the inability of a CONTROL ROD to insert upon a reactor trip, coupled with the unnecessary challenges to safety systems caused by reactor trips or dropped rods which can occur as a result of this testing supports its removal from CTS. This change is consistent with NUREG-1430.

Note: This change will not remove the requirement to perform routine freedom of movement verification of the CONTROL RODS on a Frequency of every 92 days in accordance with ITS SR 3.1.4.2.
(Note--This DOC was written with significant reliance on information presented in Section 4 of NUREG-1366 published 12/92)

L5 CTS 3.5.2.2.3 has been modified to be consistent with the requirements of ITS 3.1.4 Required Action B.1. CTS 3.5.2.2.3 requires the unit be placed in Hot Standby (i.e., reactor critical but THERMAL POWER < 2% RTP) if, after one hour, SDM had not been verified to be greater than or equal to that required by the COLR. This CTS action is required regardless of whether or not boration is in progress to establish the required SDM. ITS 3.1.4 allows continued operation after one hour, even if the required SDM has not been verified, provided boration to establish SDM has been initiated. The adoption of the ITS 3.1.4 requirements allow the unit staff to focus on the restoration of required SDM without the additional operator burden of performing a unit shutdown. The initiation of boration to establish SDM will, in most cases, result in a reduction in power level which requires significant attention from the operating staff. This reduction of power level, when further complicated by the existence of an inoperable or misaligned CONTROL ROD, significantly complicates the operation of the Control Rod Drive System. These complications require even more attention from the operating staff. In light of these complicating factors, the requirement to shutdown the unit within one hour while less than adequate SDM exists, provided boration has been initiated to establish SDM, is not in the best interest of safety; and therefore, is not being retained. This change is consistent with NUREG-1430.

L6 CTS 3.1.3.5 requires that all safety rod groups be fully withdrawn prior to and during the approach to criticality. CTS 3.1.3.7 provides the action requirements if CTS 3.1.3.5 is not met, unless otherwise excepted. CTS 3.1.3.7 requires the inserted safety rod group be withdrawn within 15 minutes or the reactor be placed in at least Hot Shutdown (MODE 3) within the next 15 minutes. These CTS actions are predicated on entire "group" being out-of-position while the unit is in its approach to criticality. Individual safety rod and multiple rod inoperability (due to misalignment, loss of position indication, or slow drop time) is addressed by the CTS 3.5.2 and CTS 4.7.1 series of Specifications.

NUREG-1430 and ITS LCO 3.1.5 require that each safety rod be fully withdrawn during MODES 1 and 2. The NUREG and ITS are predicated on an "individual" rod basis and not a group position basis. Although this translates into an identical requirement to have all safety rods fully withdrawn in MODES 1 and 2, there will be no safety rod group position requirements or actions in the ITS, only individual safety rod requirements and actions. Thus, the ITS will not include actions comparable to CTS 3.1.3.7 requirements. This results in the ITS providing less restrictive requirements than the CTS.

As an effort to highlight these changes, CTS 3.1.3.7 was marked to show ITS 3.1.5 Required Action A.2, which declares inoperable within 1 hour, a safety rod that is not fully withdrawn. This declaration results in the performance of ITS 3.1.4 Required Actions which also preserve shutdown margin while addressing the potential operational concerns associated with a misaligned rod.

The removal CTS 3.1.3.7 group action requirement is acceptable because the ITS will continue to provide safety rod positioning requirements consistent with accident analysis assumptions. Operation with multiple safety rods misaligned or not fully withdrawn will not be allowed in the ITS; just as it is not allowed in the CTS. ITS 3.1.5 Required Action B.2 will require unit to be placed in MODE 3 within 6 hours of entry into Condition B (more than one safety rod not fully withdrawn). This time is reasonable and is based on the time required for the operator to reduce THERMAL POWER from RTP to MODE 3 without challenging unit systems. It must be noted that the CTS 3.1.3.7 time frames to be in Hot Shutdown were based on the reactor being subcritical during the approach to criticality. This change is consistent with NUREG-1430.

L7 During Power Operation (MODE 1), CTS 3.5.2.1 provides the "available shutdown margin" requirement and the action requirements in the event that SHUTDOWN MARGIN (SDM) is not adequate. In the ITS, the combination of LCO 3.1.5, "Safety Rod Insertion Limits," LCO 3.2.1, "Regulating Rod Insertion Limits," and the individual CONTROL ROD OPERABILITY requirements of LCO 3.1.4, "CONTROL ROD Group Alignment Limits," preserve the SDM requirements while in MODES 1 and 2. Maintaining CONTROL RODS within these limits will provide assurance that sufficient negative reactivity is available for insertion upon a reactor trip. During unit operation with an inoperable CONTROL ROD, CTS 3.5.2.2.2 provides a requirement to verify adequate SDM and initiate boration if SDM requirements were not met. Similarly, in the ITS, LCO 3.1.4, "CONTROL ROD Group Alignment Limits," will provide Required Actions that preserve the SDM requirements. [The relationship of ITS 3.2.1, "Regulating Rod Insertion Limits," to CTS 3.5.2.1 will be discussed, as appropriate, as a part of the discussion of ITS 3.2.1.]

In the CTS, if the "available shutdown margin" is less than required, CTS 3.5.2.1 directs the operator to "immediately initiate and continue boration injection until the required shutdown margin is restored," and CTS 3.5.2.2.2 directs that 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." In the ITS, if the LCO 3.1.4

and 3.1.5 requirements are not met, LCO 3.1.4 and 3.1.5 Required Actions A.1.1 and A.1.2; LCO 3.1.5 Required Actions B.1.1 and B.1.2; and LCO 3.1.4 Required Actions C.1.1 and C.1.2 require verification of adequate SDM and initiation of boration to restore adequate SDM within 1 hour of entry into the Condition. The adoption of ITS 3.1.4 and 3.1.5 Actions will represent a relaxation of the requirement to "immediately" initiate an action such as boration. This less restrictive requirement is acceptable because the 1 hour Completion Time is adequate for determining the SDM, and if necessary, allows the operator sufficient time to align the required valves and start the necessary pumps without unduly challenging the operator's ability to safely operate the unit. This change is consistent with NUREG-1430.

L8 CTS 3.5.2.2.3 requirements for determining SHUTDOWN MARGIN (SDM) have been modified by the adoption of the SDM definition in Section 1.1 of the ITS and its application in ITS 3.1.4 and 3.1.5. By CTS requirements, the reactivity worth of any inoperable rod, regardless of the reason for inoperability, has to be accounted for as if it will not insert into the core upon a reactor trip. The ITS will require that only the reactivity worth of CONTROL RODS which are not capable of being fully inserted into the core need be considered as penalties to SDM. The intent of the CTS requirement to consider the reactivity of an inoperable CONTROL ROD in the SDM calculation is to insure that the reactor is in fact subcritical, by the amount specified in the COLR, following the insertion of the CONTROL RODS upon a reactor trip. Provided the inoperability of a CONTROL ROD is not due to the fact that the rod is not capable of fully inserting into the core upon a reactor trip, the requirement to consider that rod incapable of inserting its negative reactivity upon a reactor trip is overly conservative. This change is consistent with NUREG-1430.

L9 The CTS markup was annotated to reflect that the Moderator Temperature Coefficient (MTC) requirements of ITS LCO 3.1.3 may be excepted during PHYSICS TESTS pursuant to the requirements of ITS LCO 3.1.9, "PHYSICS TESTS Exceptions - MODE 2." To satisfactorily determine the operational behavior and characteristics of the reactor following startup, it may be necessary to significantly increase RCS boron concentration to maintain required critical conditions. During the limited period of time that the elevated RCS boron concentrations may exist at higher than normal concentrations, the MTC may be more positive than that allowed by ITS LCO 3.1.3. It is acceptable to suspend the MTC LCO during PHYSICS TESTS in MODE 2 based on the usage of approved written procedures, administrative controls, the requirements of 10CFR50.59, and the ITS LCO 3.1.9 provisions in effect during the conduct of the PHYSICS TESTS. These exceptions accommodate LCO suspension 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 is consistent with NUREG-1430.

L10 The CTS markup was annotated to show the adoption of ITS LCO 3.1.8, "PHYSICS TEST Exceptions-MODE 1," and LCO 3.1.9, "PHYSICS TESTS Exceptions - MODE 2," allowances to suspend the requirements of ITS LCO 3.1.4, "CONTROL ROD Group Alignment Limits," and LCO 3.1.6, "APSR Alignment Limits," during the conduct of PHYSICS TESTS. These exceptions suspend certain ITS LCO requirements that did not have PHYSICS TESTS exceptions in the CTS. The adoption of these exceptions 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. These exceptions accommodate LCO suspension 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 is consistent with NUREG-1430.

L11 CTS requirements for CONTROL ROD and APSR position indication instrumentation are presented in CTS 4.7.1.3 and in CTS Table 4.1-1, Items 23 and 24. CTS 4.7.1.3 requires that for a CONTROL ROD or APSR to be considered OPERABLE, it must be located with one of three specified channels of indication. CTS Table 4.1-1 Items 23 and 24 require shiftly (12 hour) channel checks of only two of the three channels of indication specified in CTS 4.7.1.3. Additionally, refueling frequency calibrations of only these two channels are required.

Adoption of ITS LCO 3.1.7 establishes a requirement that maintains the CTS requirement that each CONTROL ROD and APSR have one OPERABLE channel of position indication. Further, ITS SR 3.1.7.1 and SR 3.1.7.2, in lieu of CTS Table 4.1-1, Items 23 and 24, provide testing requirements that establish appropriate assurance that the instrumentation required by ITS LCO 3.1.7 is OPERABLE. The potentially confusing cross-channel comparison of the CHANNEL CHECK located in CTS 4.1-1 was removed to ensure that any one OPERABLE indication channel, which can be adequately surveilled, will satisfy the LCO. The removal of this CTS cross-channel comparison detail results in the ITS being less restrictive. This is acceptable because the requirement to perform a CHANNEL CHECK of the instrumentation used to satisfy the LCO requirement is present in the ITS as SR 3.1.7.1.

L12 Testing to insure freedom of movement of "Each Rod" is required above Cold Shutdown by CTS Table 4.1-2, Item 2. Similar testing of the CONTROL RODS will be required by ITS SR 3.1.4.2 and will be applicable only in MODES 1 and 2. The adoption of the NUREG-1430 SR will result in less restrictive requirements. Specifically, the adoption of ITS SR 3.1.4.2 will remove the CTS requirement to perform this testing on CONTROL RODS while in MODES 3 and 4. This change actually only removes the requirement to test the CONTROL RODS while in operational MODES in which OPERABILITY of the CONTROL RODS is <u>not</u> required. This change provides for the application of Surveillance Requirements consistent with the MODES of Applicability for the tested components and is consistent with NUREG-1430.

L13 Not used.

L14 The shutdown actions in CTS 3.1.9.3 are proposed for deletion. CTS 3.1.9.1 and CTS 3.1.9.2 establish limits for the concentration of dissolved gases in the reactor coolant. These dissolved gas limits are intended to prevent possible control rod drive

3.1-02

and/or control rod damage during a trip by ensuring that the control rod drive pressure housing is filled with water. CTS 3.1.9.3 specified an action to check the vessel level instrument vent for the accumulation of undissolved gases should the limits be exceeded. This action would be performed with the reactor shutdown because of the vent's location on the reactor vessel head. The dissolved gas limits will be relocated to the Technical Requirements Manual (TRM). The purpose of the dissolved gas limits is to protect the control rods from damage due to a loss of hydraulic buffering upon insertion due to a trip. The control rods are still capable of inserting into the core even with dissolved gases not within limits. However, the control rods may not be able to be withdrawn following such a trip. The TRM will contain actions which, in the event dissolved gas concentrations not within limits, will require the reactor vessel level instrument to be checked for the accumulation of undissolved gases within 24 hours, and the restoration of the concentration of dissolved gases to within limits within 24 hours. In the event these Required Actions and Completion Times are not met, the TRM actions also ensure that the conditions are evaluated under the ANO Corrective Actions Program, allowing site management to determine any limitations on continued operation of the unit. The deletion of the CTS 3.1.9.3 actions is acceptable since the presence of dissolved gases beyond limits will not affect the safety function of the control rods to insert into the core. Adequate guidance for ensuring appropriate corrective measures will be taken will be included in the TRM. Since the TRM is considered to be a part of the SAR by reference, changes to the TRM are controlled under the ANO 10 CFR 50.59 program.

# CTS DISCUSSION OF CHANGES

## LESS RESTRICTIVE -- ADMINISTRATIVE DELETION OF REQUIREMENTS

LA1 This information has been moved to the Bases, SAR, TRM or COLR. This information provides details of design or process which are not directly pertinent to the actual requirement, i.e., Definition, Limiting Condition for Operation or Surveillance Requirement, but rather describe an acceptable method of compliance. Since these details are not necessary to adequately describe the actual regulatory requirement, they can be moved to a licensee controlled document without a significant impact on safety. Placing these details in controlled documents provides adequate assurance that they will be maintained. The Bases will be controlled by the Bases Control Process in Chapter 5 of the proposed Technical Specifications. The details of performance of the surveillances have generally been relocated to the TRM. The TRM and COLR are considered to be part of the SAR. Changes to the SAR, TRM, and COLR will be controlled by 10 CFR 50.59. This change is consistent with NUREG-1430.

3.1-01

| CTS Location | New Location |
|---|---|
| 3.1.7.2 | Bases - SR 3.1.3.1 |
| 3.1.9.1 | TRM |
| 3.1.9.2 | TRM |
| 3.1.9.3 | TRM |
| Figure 3.1.9-1 | TRM |
| Table 4.1-3, Item 1.d | TRM |
| Table 4.1-3, Note 7 | TRM |
| 4.7.1.1 | SAR - Section 7.2.2.2.1 |
| 4.7.1.2 | Bases - B 3.1.4 LCO |
| 4.7.1.3 | Bases – B 3.1.7 Background |

3.1.5
3.1.8
3.1.9

<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. (A7)

3.1.9 LCO

<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

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. (M6)

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

<LATER> (3.2)

3.1.8, 3.1.9 LCO

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 and 2 — LATER — (A10) — LATER — (A10)

3.1.5 LCO

<LATER> (3.2)

3.1.5 Appl

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

3.1.5 RA A.2

<LATER> (3.4A&B) 3.1.3.7 With any Safety rod not fully withdrawn ~~of the above limits violated~~, restore the safety rod ~~reactor~~ to within the limit in 1 hour ~~15 minutes~~ or declare the safety rod inoperable ~~or be in at least Hot Shutdown within the next 15 minutes.~~ — L6 + LATER

Bases

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

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

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

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

3.1.1
3.1.5

A2

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

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

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

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

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

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


REFERENCES

(1) FSAR, Section 3

(2) FSAR, Section 3.2.2.1.5

3.1.1
3.1.3
3.1.9

MTC — A1

3.1.7 Moderator Temperature Coefficient MTC of Reactivity
Specification

3.1.3 LCO — 3.1.7.1 The moderator temperature coefficient (MTC) shall be non-positive whenever thermal power is ≥95% of rated thermal power and shall be less positive than $0.9 \times 10^{-4}$ Δk/k/°F whenever thermal power is <95% of rated thermal power ~~and the reactor is not shutdown~~ MODES 1 and 2 — M4

3.1.3 Appl.

SR 3.1.3.1 3.1.7.2 The MTC shall be determined to be within its limits by confirmatory measurements prior to initial operation in MODE 1 — A1 ~~above 5% of rated thermal power~~ after each fuel loading. ~~MTC measured values shall be extrapolated and/or compensated to permit direct comparison with the limits in 3.1.7.1 above.~~ — LA1 BASES

3.1.3 RA A.1 3.1.7.3 With the MTC outside any one of the above limits, be in at least ~~HOT STANDBY~~ MODE 3 within 6 hours. — M12

~~Bases~~

~~A non-positive moderator coefficient at power levels above 95% of rated power is specified such that the maximum clad temperatures will not exceed the Final Acceptance Criteria based on LOCA analyses. Below 95% of rated power, the Final Acceptance Criteria will not be exceeded with a positive moderator temperature coefficient of $+0.9 \times 10^{-4}$ Δk/k/°F corrected to 95% of rated power. The most limiting event for positive MTC, the Startup Accident, has been analyzed for a range of moderator temperature coefficients including $+0.9 \times 10^{-4}$ Δk/k/°F.~~ — A2

<Add 3.1.9 PHYSICS TESTS exception to LCO 3.1.3> — L9

<Add 3.1.1 SHUTDOWN MARGIN (SDM)> — M6

3.1.8 ~~Low Power~~ Physics ~~Testing Restrictions~~ Exceptions — MODE 2 (A1)

~~Specification~~

~~The following special limitations are placed on low power physics testing.~~ (A1)

3.1.8.1 Reactor Protective System Requirements (A8)

3.1.9.a LCO — A. Below 1720 psig, shutdown bypass trip setting limits shall apply in accordance with Table 2.3-1.

3.1.9.b LCO — B. Above 1800 psig, nuclear overpower trip shall be set at less than 5.0 percent. Other settings shall be in accordance with Table 2.3-1.

3.1.9.c LCO — 3.1.8.2 Startup rate rod withdrawal hold (1) shall be in effect at all times.

3.1.8.3 ~~During low power physics testing the minimum reactor coolant temperature for criticality shall be to the right of the criticality limit of Figure 3.1.2-2.~~ (A7) The shutdown margin shall be maintained greater than or equal to that specified in the COLR with the highest worth control rod fully withdrawn.

3.1.8.d LCO
3.1.9.d LCO

~~Bases~~

~~The above specification provides additional safety margins during low power physics testing.~~ (A2)

~~REFERENCES~~

~~(1) FSAR, Section 7.2.2.1.3.~~

<ADD 3.1.8 LCO a, b, c with Note; Appl.; Actions SR 3.1.8.1; SR 3.1.8.2 with Note; SR 3.1.8.3; SR 3.1.8.4> (M14)

<Add 3.1.9 Appl> (A8)

<Add 3.1.9 ACTIONS and SRs> (M14)

<Add LCO 3.1.8 & LCO 3.1.9 PHYSICS TESTS exceptions to LCO 3.1.4 and LCO 3.1.6.> (L10)

3.1

TS1-313-07-00 A1

3.1.9 Control Rod Operation

Specification

3.1.9.1 The concentration of dissolved gases in the reactor coolant shall be limited to 100 std. cc/liter of water at the reactor vessel outlet temperature.

3.1.9.2 Allowable combinations of pressure and temperature for control rod operation shall be to the left of and above the limiting pressure versus temperature curve for a dissolved gas concentration of 100 std. cc/liter of water as shown in Figure 3.1.9-1.

LA1

TRM

3.1.9.3 In the event the limits of Specifications 3.1.9.1 or 3.1.9.2 are exceeded, the vessel level instrument vent shall be checked for accumulation of undissolved gases. The temperature, pressure and dissolved gas concentration shall be restored to within their limit within 24 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

L14

31-02

Bases

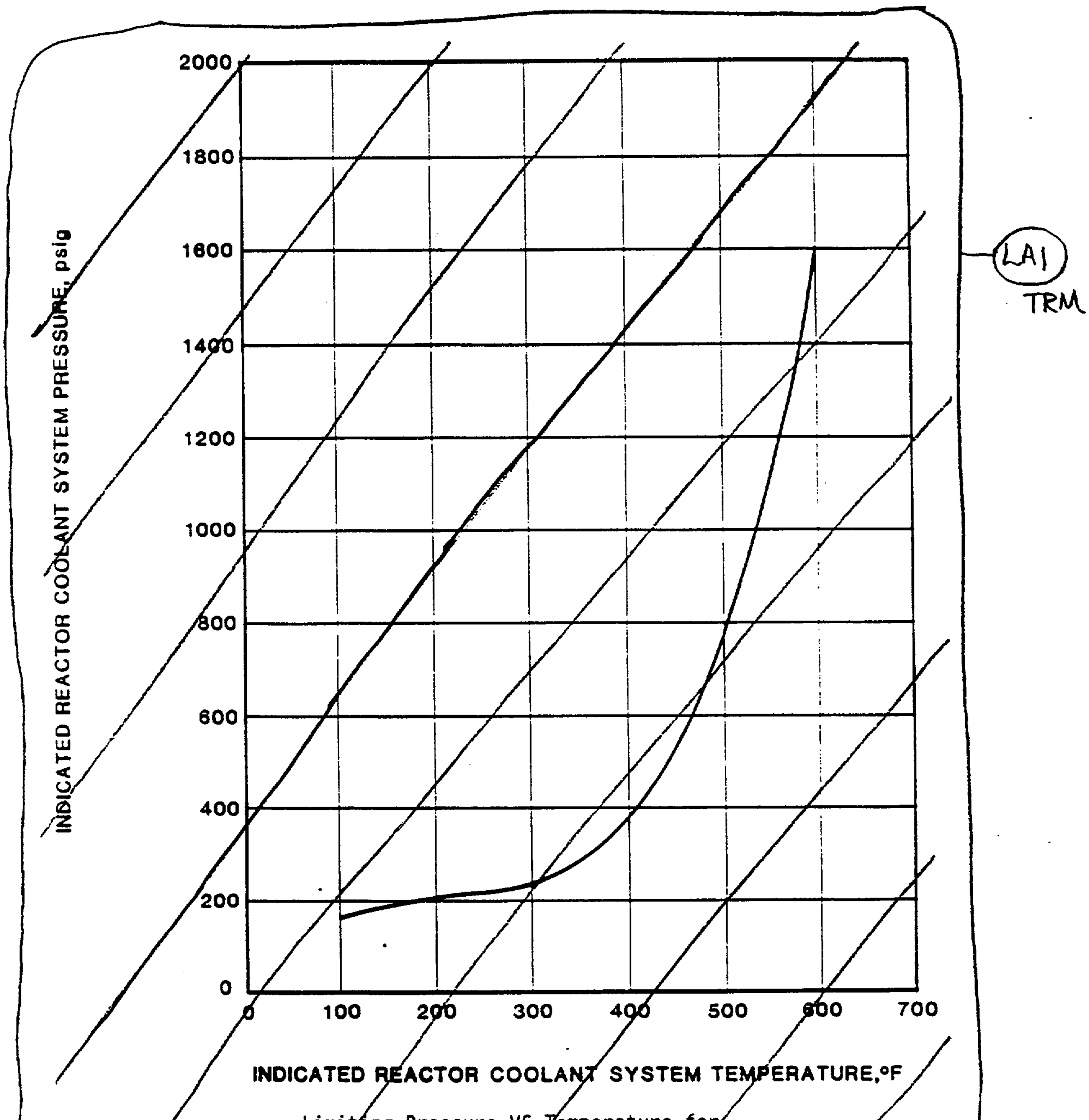
By maintaining the reactor coolant temperature and pressure as specified above, any dissolved gases in the reactor coolant system are maintained in solution.

Although the dissolved gas concentration is expected to be approximately 20-40 std. cc/liter of water, the dissolved gas concentration is conservatively assumed to be 100 std. cc/liter of water at the reactor vessel outlet temperature.

The limiting pressure versus temperature curve for dissolved gases is determined by the equilibrium pressure versus temperature curve for the equilibrium dissolved gas concentration of 100 std. cc/liter of water. The equilibrium total pressure is the sum of the partial pressure of the dissolved gases plus the partial pressure of water at a given temperature. The margin of error consists of the maximum pressure difference between the pressure sensing tap and lowest pressure point in the system, the maximum pressure gage error, and the pressure difference due to the maximum temperature gage error.

A2

If either the maximum dissolved gas concentration (100 std. cc/liter of water) is exceeded or the operating pressure falls below the limiting pressure versus temperature curve, the vessel level instrument vent should be checked for accumulation of undissolved gases.




Limiting Pressure VS Temperature for
Control Rod Drive Operation With
100 STD CC/Liter $H_2O$

Figure 3.1.9-1

3.1.4
3.1.5
3.1.6

< Add 3.1.4 RA A.1.1 second Completion Time > M10

< Add 3.1.4 RA B.1 Completion Time > M22

3.5.2 Control Rod Group and Power Distribution Limits

3.1.4 Appl
3.1.5 Appl
3.1.6 Appl
& <LATER> (3.2)

**Applicability**

This specification applies to power distribution and operation of control rods during ~~power operation.~~ MODES 1 and 2. — M5 — & LATER

~~**Objective**~~

~~To assure an acceptable core power distribution during power operation, to set a limit on potential reactivity insertion from a hypothetical control rod ejection, and to assure core subcriticality after a reactor trip.~~ — A1

**Specification**

3.1.5 LCO
3.1.5 RA A.1.1 & A.1.2
3.1.5 RA B.1.1 & B.1.2
& <LATER> (3.2)

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.~~ within 1 hr — & LATER — L7

3.5.2.2 Operation with inoperable rods:

3.1.4 R.A. C.2
3.1.5 R.A. B.2

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. Be in MODE 3 in 6 hours. — M7

3.1.4 RA A.1.1
3.1.4 RA A.1.2
3.1.4 RA C.1.1
3.1.4 RA C.1.2

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.~~ — L7 — M20 — L4 — L8

3.1.4 RA A.1.1, A.1.2, C.1.1, & C.1.2 Completion Time

any CONTROL ROD not capable of being fully inserted

3.1.4 RA B.1

and boration to restore SDM has not been initiated

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. MODE 3 in 6 hours — M21 — L5

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.~~ — L4

3.1.4 RA A.2.2.1

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.~~ cap — A1

< Add 3.1.4 RA A.2.2.1 Completion Time > M9

< Add 3.1.4 RA A.2.2.3 with Note > M19

3.1.4
3.1.6
3.1.8

3.1-10

<Add 3.1.6 RA A.1 and Completion Time> M27

<Add 3.1.4 RA A.2.1 Completion Time> M23

3.1.4 RA A.2.1

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. M27

3.1.4 RA A.2.2.2

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. A9

LATER (3.2)

3.5.2.4 Quadrant Power Tilt:

3.1.8 LCO

LATER (3.2)

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 power level for the existing reactor coolant pump combination less at least 2% for each 1% tilt in excess of the tilt limit. LATER

3.1.8 LCO

2. Within a period of 4 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:

LATER (3.2)

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.

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

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

3.1.8 LCO

LATER (3.2)

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

4. Quadrant power tilt shall be monitored on a minimum frequency of once every two hours during power operation above 15% of rated power.

<Add 3.1.4 RA A.2.2.2 Completion Time> M16

<Add 3.1.6 RA A.1 Completion Time> M24

<Add 3.1.6 RA A.2 + Completion Time> M25

<Add 3.1.6 Condition B> M26

3.1-10

3.1.5
3.1.8
3.1.9

3.5.2.5 Control rod positions:

3.1.5 LCO NOTE

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. A5

<LATER> (3.2)

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

3.1.8 LCO
3.1.9 LCO

3.1.8
3.1.9

3.1.8 LCO
3.1.9 LCO

⟨LATER⟩ (3.2)

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. 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. —LATER

3.1.8 LCO
3.1.9 LCO

⟨LATER⟩ (3.2)

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.

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 4 hours. —LATER

⟨LATER⟩ (3.2)

3.5.2.6 Reactor Power Imbalance:

1. Reactor power imbalance shall be monitored on a frequency not to exceed 2 hours during power operation above 40% rated power. —LATER

3.1.8 LCO

2. Except for physics tests, reactor power imbalance shall be maintained within the envelope defined by the CORE OPERATING LIMITS REPORT.

3. If the 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.

4. If an acceptable reactor power imbalance is not achieved within 4 hours, reactor power shall be reduced until reactor power imbalance setpoints are met.

⟨LATER⟩ (3.2)

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. —LATER

**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 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)

3.1.8
3.1.9

simultaneously all other engineering and uncertainty factors are also at their limits.* Conservatism is introduced by application of:

a. Nuclear uncertainty factors.
b. Thermal calibration.
c. Fuel densification effects.
d. Hot rod manufacturing tolerance factors.
e. Fuel rod bowing.

The 20 ±5 percent overlap between successive control rod groups is allowed since the worth of a rod is lower at the upper and lower parts of the stroke. Control rods are arranged in groups or banks defined as follows:

| Group | Function |
|---|---|
| 1 | Safety |
| 2 | Safety |
| 3 | Safety |
| 4 | Safety |
| 5 | Regulating |
| 6 | Regulating |
| 7 | Regulating |
| 8 | APSR (axial power shaping bank) |

The rod position limits are based on the most limiting of the following three criteria: ECCS power peaking, shutdown margin, and potential ejected rod worth. As discussed above, compliance with the ECCS power peaking criterion is ensured by the rod position limits. The minimum available rod worth, consistent with the rod position limits, provides for achieving hot shutdown by reactor trip at any time, assuming the highest worth control rod that is withdrawn remains in the full-out position (1). The rod position limits also ensure that inserted rod groups will not contain single rod worths greater than 0.65% $\Delta k/k$ at rated power. These values have been shown to be safe by the safety analysis (2) of the hypothetical rod ejection accident. A maximum single inserted control rod worth of 1.0% $\Delta k/k$ is allowed by the rod position limits at hot zero power. A single inserted control rod worth of 1.0% $\Delta k/k$ at beginning of life, hot zero power, would result in a lower transient peak thermal power and therefore less severe environmental consequences than a 0.65% $\Delta k/k$ ejected rod worth at rated power.

Control rod Groups are withdrawn in sequence beginning with Group 1. Groups 5, 6, and 7 are overlapped 20%. The normal position at power is for Groups 6 and 7 to be partially inserted.

*Actual operating limits depend on whether or not incore or excore detectors are used and their respective instrument and calibration errors. The method used to define the operating limits is defined in plant operating procedures.

A2

3.1.8
3.1.9

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.

A2

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 | [illegible] |
| 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

## Table 4.1-1 (Cont'd)

| | Channel Description | Check | Test | Calibrate | Remarks | |
|---|---|---|---|---|---|---|
| ⟨LATER⟩ (3.3 B) | 20. Reactor Building Spray System System Logic Channels | NA | M(1) | NA | (1) Including RB spray pump, spray valves, and chem. add. valve logic channels. | LATER |
| | 21. Reactor Building Spray System Analog Channels | | | | | |
| | a. Reactor Building Pressure Channels | NA | M | R | | |
| ⟨LATER⟩ (3.3 D) | 22. Pressurizer Temperature Channels | S | NA | R | | LATER |
| | 23. Control Rod Absolute Position | S(1) SR 3.1.7.1 | NA | R SR 3.1.7.2 | (1) Compare with Relative Position Indicator | L11 |
| | 24. Control Rod Relative Position | S(1) SR 3.1.7.1 | NA | R SR 3.1.7.2 | (1) Check with Absolute Position Indicator | |
| ⟨LATER⟩ (3.5) | 25. Core Flooding Tanks | | | | | |
| | a. Pressure Channels | S | NA | R | | LATER |
| | b. Level Channels | S | NA | R | | |
| ⟨LATER⟩ (3.3 D) | 26. Pressurizer Level Channels | S | NA | R | | LATER |
| | 27. Makeup Tank Level Channels | D | NA | R | | |
| ⟨LATER⟩ (3.3D & 3.4B) | 28. Radiation Monitoring Systems other than containment high range monitors (item 57) | | | | (1) Check functioning of self-checking feature on each detector. | LATER |
| | a. Process Monitoring System | S | Q | R | | |
| ⟨LATER⟩ (3.3D) | b. Area Monitoring System | S | M(1) | R | | LATER |
| | c. Main Steam Line Radiation Monitors | S | M | R | | |

3.1.7

3.1.4
3.1.5
3.1.6

<Add SR 3.1.4.1> M11
<Add SR 3.1.5.1> M11
<Add SR 3.1.6.1> M11

Table 4.1-2
~~Minimum Equipment Test Frequency~~ A1

| Item | Test | Frequency | |
|---|---|---|---|
| SR 3.1.4.3 — 1. Control Rods CAP | Rod Drop Times of all Full Length Rods 1/ | ~~Each Refueling Shutdown~~ Following Reactor Vessel Head Removal | M8 |
| SR 3.1.4.2 — 2. Control Rod CAP Movement | Movement of Each CONTROL ROD ~~Rod~~ In MODES 1 and 2. | Every ~~two weeks~~ 92 days ~~Above Cold Shutdown conditions~~ | L1, L3, L12 |
| <LATER> (3.4B) — ~~3. Pressurizer Code Safety Valves~~ | ~~Setpoint~~ | ~~One Valve Every 18 Month~~ | LATER |
| <LATER> (3.7) — ~~4. Main Steam Safety Valves~~ | ~~Setpoint~~ | ~~Four Valves Every 18 Mon~~ | LATER |
| ~~5. Refueling System Interlocks~~ | ~~Functioning~~ | ~~Start of Each Refueling Shutdown~~ | R TRM |
| <LATER> (3.4B) — ~~6a. Reactor Coolant System Leakage~~ | ~~Evaluate~~ | ~~Daily~~ | LATER |
| ~~b. Reactor Coolant System Pressure Isolation Valves~~ | ~~Leakage Test Per Table 3.1.6.9~~ | ~~See Notes 1 & 2~~ | |
| ~~7. Emergency-powered Pressurizer Heaters~~ | ~~Power availability~~ | ~~Daily~~ | |
| | ~~Heater capacity functional test~~ | ~~Every 18 Months~~ | |
| <LATER> (3.6) — ~~8. Reactor Building Isolation Trip~~ | ~~Functioning~~ | ~~Every 18 Months~~ | LATER |
| <LATER> (3.7) — ~~9. Service Water Systems~~ | ~~Functioning~~ | ~~Every 18 Months~~ | LATER |
| ~~10. Spent Fuel Cooling System~~ | ~~Functioning~~ | ~~Every 18 Months when irradiated fuel is in the pool~~ | R TRM |

~~1/ Same as tests listed in Section 4.7~~ A1

<LATER> (3.4B) — LATER

~~Notes:~~

~~(1) Leak testing for each valve shall be individually accomplished to demonstrate operability following each refueling, following each time the plant is placed in a cold shutdown condition if testing has not been accomplished in the preceding 9 months, and prior to returning the valve to service after maintenance, repair or replacement.~~

~~(2) Whenever integrity of a pressure isolation valve listed in Table 3.1.6.9 cannot be demonstrated the integrity of the remaining valve in each high pressure line having a leaking valve shall be determined and recorded daily. In addition, the position of one other valve located in the high pressure piping shall be recorded daily.~~

Table 4.1-3

MINIMUM SAMPLING AND ANALYSIS FREQUENCY

| Item | Test | Frequency |
|---|---|---|
| ⟨LATER⟩ (3.4B) 1. Reactor Coolant Samples | a. Gamma Isotopic Analysis | a. Bi-weekly (7) |
| | b. Gross Activity Determination | b. 3 times/week and at least every third day (1)(6)(7) LATER |
| | c. Gross Radioiodine Determination | c. Weekly (3)(6)(7) |
| | d. Dissolved Gases | d. Weekly (7) LA1 TRM |
| ⟨LATER⟩ (3.4A) | e. Chemistry (Cl, F, and $O_2$) | e. 3 times/week (8) LATER |
| ⟨LATER⟩ (3.9) | f. Boron Concentration | f. 3 times/week LATER & R TRM |
| ⟨LATER⟩ (3.4B) | g. Radiochemical Analysis for Ē Determination (2) (4) | g. Monthly (7) LATER |
| ⟨LATER⟩ (3.5) 2. Borated Water Storage Tank Water Sample | Boron Concentration | Weekly and after each makeup LATER |
| 3. Core Flooding Tank Sample | Boron Concentration | Monthly and after each makeup |
| 4. Spent Fuel Pool Water Sample | Boron Concentration | Monthly and after each makeup (9) LATER |
| ⟨LATER⟩ (3.7) 5. Secondary Coolant Samples | a. Gross Radioiodine Concentration | a. Weekly (5)(7)(10) R TRM |
| | b. Isotopic Radioiodine Concentration (4) | b. Monthly (7)(10) LATER |
| ⟨LATER⟩ (3.6) 6. Sodium Hydroxide Tank Sample | Sodium Hydroxide Concentration | Quarterly and after each makeup LATER |

Notes:

⟨LATER⟩ (3.4B) (1) A gross radioactivity analysis shall consist of the quantitative measurement of the total radioactivity of the primary coolant in units µCi/gm. The total primary coolant activity shall be the sum of the degassed beta-gamma activity and the total of all identified gaseous activities 15 minutes after the primary system is sampled. Whenever the gross radioactivity concentration exceeds 10% of the limit specified in the Specification 3.1.4.1 or increases by 10 µCi/gm from the previous measured level, the frequency of sampling and analyzing shall be increased to a minimum of once/day until a steady activity level is established. LATER

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

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

⟨LATER⟩ (3.4B) — LATER

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

⟨LATER⟩ (3.4B, 3.7) — LATER

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

— R TRM

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

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

⟨LATER⟩ (3.4B) — LATER

(7) Not required when plant is in the cold shutdown condition or refueling shutdown condition.

& ⟨LATER⟩ (3.4B, 3.7) — & R TRM & LATER

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

⟨LATER⟩ (3.4A) — LA1 TRM — LATER

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

⟨LATER⟩ (3.7) — LATER

(10) Not required when not generating steam in the steam generators.

& ⟨LATER⟩ (3.7) — R TRM & LATER

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

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

⟨LATER⟩ (3.4B) — LATER

3.1.4
3.1.6
3.1.7

<Add SR 3.1.4.3 Note> L2

~~4.7 REACTOR CONTROL ROD SYSTEM TESTS~~

~~4.7.1 Control Rod Drive System Functional Tests~~

~~Applicability~~

~~Applies to the surveillance of the control rod system.~~

~~Objective~~

~~To assure operability of the control rod system.~~

~~Specification~~

A1

criticality L2

reactor vessel head removal M8

M17

SR 3.1.4.3

4.7.1.1 The control rod trip insertion time shall be measured for each control rod at either full flow ~~or no flow~~ conditions following each ~~refueling outage~~ prior to ~~return to power~~. The maximum control rod trip insertion time for an operable control rod drive mechanism, except for the Axial Power Shaping Rods (APSRs), from the fully withdrawn position to 3/4 insertion (104 inches travel) shall not exceed 1.66 seconds at reactor coolant ~~full~~ flow conditions ~~of 1.20 seconds for no flow conditions. For the APSRs it shall be demonstrated that loss of power will not cause rod movement.~~ If the trip insertion time above is not met, the rod shall be declared inoperable.

LA1 SAR

6.5%

A3

A11

3.1.4 LCO
3.1.6 LCO

3.1-07

4.7.1.2 If a control rod is misaligned with its group average by more than an indicated nine (9) inches, ~~the rod shall be declared inoperable and the limits of Specification 3.5.2.2 shall apply. The rod with the greatest misalignment shall be evaluated first. The position of a rod declared inoperable due to misalignment shall not be included in computing the average position of the group for determining the operability of rods with lesser misalignments.~~

LA1 BASES

A13

A6

3.1.7 RA A.1

4.7.1.3 If a control rod cannot be exercised, ~~or if it cannot be located with absolute or relative position indications or in or out limit lights~~ the rod shall be declared to be inoperable.

LA1 BASES

~~Bases~~

~~The control rod trip insertion time is the total elapsed time from power interruption at the control rod drive breakers until the control rod has completed 104 inches of travel from the fully withdrawn position. The specified trip time is based upon the safety analysis in FSAR, Section 14, whose calculations are based on a rod drop from fully withdrawn to 2/3 inserted. Since the most accurate position indication is obtained from the zone reference switch at the 3/4 inserted position, this position is used instead of the 2/3 inserted position for data gathering.~~

A2

~~Each control rod drive mechanism shall be exercised by a movement approximately two (2) inches of travel every two (2) weeks. This requirement shall apply to either a partial or fully withdrawn control rod at reactor operating conditions. Exercising the drive mechanisms in this manner provides assurance of reliability of the mechanisms.~~

~~A rod is considered inoperable if it cannot be exercised, if the trip insertion time is greater than the specified allowable time, or if the rod~~

<Add 3.1.7 LCO> L11

<Add 3.1.7 Appl.> M13

<Add 3.1.7 Actions Note> A12

~~deviates from its group average position by more than nine (9) inches. Conditions for operation with an inoperable rod are specified in Technical Specification 3.5.2.~~

~~REFERENCE~~

~~(1) FSAR, Section 14~~

A2

3.1.2 — 4.9 REACTIVITY ~~ANOMALIES~~ BALANCE A1

3.1.2 Appl

~~Applicability~~

~~Applies to potential reactivity anomalies.~~

~~Objective~~ A1

~~To require the evaluation of reactivity anomalies of a specified magnitude occurring~~ ~~during the operation of the unit.~~ MODES 1 and 2 A1

Specification

SR 3.1.2.1 Following a normalization of the computed boron concentration as a function of burnup, the actual boron concentration of the coolant shall be periodically compared with the predicted value. M1

3.1.2 LCO If the difference between the observed and predicted steady-state concentrations reaches the

3.1.2 RA A.1 equivalent of one percent in reactivity, an evaluation of this abnormal occurrance will be made within 7 days to determine the cause of the discrepancy. M2

~~Bases~~

~~To eliminate possible errors in the calculations of the initial reactivity of the core and the reactivity depletion rate, the predicted relation between fuel burnup and the boron concentration, necessary to maintain adequate control characteristics, must be adjusted (normalized) to accurately reflect actual core conditions. When full power is reached initially, and with the control rod groups in the desired positions, the boron concentration is measured and the predicted curve is adjusted to this point. As power operation proceeds, the measured boron concentration is compared with the predicted concentration and the slope of the curve relating burnup and reactivity is compared with that predicted. This process of normalization should be completed after about 10 percent of the total core burnup. Thereafter, actual boron concentration can be compared with prediction, and the reactivity status of the core can be continuously evaluated. Any reactivity anomaly greater than 1 percent Δk/k would be unexpected, and its occurrence would be thoroughly investigated and evaluated.~~

~~The value of 1 percent Δk/k is considered a safe limit since a shutdown margin of at least 1 percent Δk/k with the most reactive rod in the fully withdrawn position is always maintained.~~ A2

<Add 3.1.2 RA A.2> M15

<Add 3.1.2 RA B.1> M15

<Add SR 3.1.2.1 NOTE and Frequency with NOTE> M15

# NO SIGNIFICANT HAZARDS CONSIDERATIONS
# GENERIC EVALUATIONS

**"R" - Relocation of requirements:**

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

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

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

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

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

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

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

## 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.

# NO SIGNIFICANT HAZARDS CONSIDERATIONS
# GENERIC EVALUATIONS

**"M" - More restrictive changes to requirements:**

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

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

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

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

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

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

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

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

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

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

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

# NO SIGNIFICANT HAZARDS CONSIDERATIONS STATEMENTS

## ITS Section 3.1: Reactivity Control Systems

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

### 3.1 L1

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

A less frequent performance of a Surveillance Requirement does not result in any hardware changes. The Frequency of performance also does not significantly increase the probability of occurrence for initiation of any analyzed event since the function of the equipment does not change (and therefore any initiation scenarios are not changed) and the proposed Frequency has been determined to be adequate to demonstrate reliable operation of the equipment. This position is supported by Generic Letter 93-05, "Line-Item Technical Specifications Improvements to Reduce Surveillance Requirements for Testing During Power Operation." Further, the Frequency of performance of a surveillance does not significantly increase the consequences of an accident because a change in Frequency does not change the assumed response of the equipment in performing its specified mitigation functions from that considered with the original Frequency. The control rods are used to support mitigation of the consequences of an accident; however, industry experience has shown a Frequency of 92 days is sufficient to detect failures in the Rod Control System and other information is available to the operator, e.g., individual rod position, to identify abnormalities. Therefore, this change does not involve a significant increase in the consequences of any accident previously evaluated.

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

The proposed change does not necessitate a physical alteration of the plant (no new or different type of equipment will be installed) or changes in parameters governing normal plant operation. The proposed change will still ensure proper surveillances are required for all equipment considered in the safety analysis. Thus, this change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

Changes in the surveilled parameter have been determined to be relatively slow during the proposed intervals, and the proposed Frequency has been determined to be sufficient to identify significant impact on compliance with the assumed conditions of the safety analysis. In addition, other indications continue to be available to indicate potential noncompliance. Therefore, an extended surveillance interval does not involve a significant reduction in the margin of safety.

# NO SIGNIFICANT HAZARDS CONSIDERATIONS STATEMENTS

## 3.1 L2

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

The control rods are used to support mitigation of the consequences of an accident; however, the reactor coolant system (RCS) flow conditions during control rod drop time verification are not considered an initiator of any previously analyzed accident. As such, the proposed change in the allowed RCS flow conditions will not significantly increase the probability of any accident previously evaluated. The proposed changes allow for testing the control rod drop times with less than a full complement of reactor coolant pumps operating. However, the operation of the plant is restricted to the pump combinations providing maximum flow less than or equal to the pump flow used for the testing. Therefore, the drop times verified during testing will remain valid for mitigating the consequences of any accident previously evaluated. 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 continue to ensure that the control rods are available for insertion of reactivity in the time frames consistent with the safety analysis. Thus, this change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

The margin of safety provided in the acceptable control rod drop times continues to be provided since these drop times have not been changed. The surveillance methodology is revised to allow testing with one, two, or three pumps operating. However, the operation of the plant is restricted to the reactor coolant pump combinations which maintain the margin of safety, i.e., those pump combinations providing maximum flow less than or equal to the pump flow used for the testing. Therefore, this change does not involve a significant reduction in a margin of safety.

# NO SIGNIFICANT HAZARDS CONSIDERATIONS STATEMENTS

## 3.1 L3

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

This change does not result in any hardware changes. Because the APSRs are not designed to insert on a reactor trip and are not credited toward the required shutdown margin, the removal of the requirement to verify Axial Power Shaping Rod (APSR) freedom of movement does not alter the functional performance characteristics of the control rods in performing their assumed safety analysis function.. As such the proposed change will not significantly increase the probability of any accident previously evaluated. Neither will the change alter the assumed function of the control rods in providing their assumed safety analysis function. Nor will this change alter the requirement to perform a freedom of movement verification of the control rods. Therefore, the proposed change does not involve a significant increase to the consequences of any accident previously evaluated.

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

The proposed change does not 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 require proper demonstration of control rod OPERABILITY, consistent with applicable safety analysis assumptions and regulatory guidance. 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 removes the required periodic verification that the APSR is moveable. The change does not alter the assumed function of the APSR or the operational restrictions and the administrative controls associated with the APSRs. Nor does the change alter the ability of the control rods to satisfy the safety analyses assumed function. As such, this change does not involve a significant reduction in a margin of safety.

# NO SIGNIFICANT HAZARDS CONSIDERATIONS STATEMENTS

## 3.1 L4

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

This change does not result in any hardware changes or changes in operating practice. The change removes an unnecessary additional performance of a surveillance which has been performed within its normally required Frequency. Not performing the surveillance would not affect any equipment which is assumed to be an initiator of any analyzed event. Further, since the surveillance continues to be performed on its normal Frequency there is no impact on the capability of the system to perform its required safety function. Therefore, the proposed change does not involve a significant increase in the probability or consequences of any accident previously evaluated.

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

The proposed change does not necessitate a physical alteration of the plant (no new or different type of equipment will be installed) or changes in parameters governing normal plant operation. The proposed change will still ensure adequate surveillance is performed to identify any degradation of the control rod freedom of movement. 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 normal surveillance Frequency has been shown, based on operating experience, to be adequate for assuring the equipment is available and capable of performing its intended function. Additionally, the requirements of SR 3.0.4 (CTS 4.0.4) provide assurance the equipment is OPERABLE prior to beginning the functions for which it is required. Therefore, the proposed change does not involve a significant reduction in the margin of safety.

## 3.1 L5

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

An immediate shutdown of the unit is not considered the most appropriate action for a loss of shutdown margin since such an action may result in diminished control capability. Rather, ACTIONS are proposed which will allow boration to restore the required shutdown margin to continue beyond the one hour time frame specified in the CTS. Neither an inoperable control rod nor inadequate shutdown margin have been considered as initiators for any accident previously evaluated. Therefore, an extended time frame in these conditions will not involve a significant increase in the probability of any previously evaluated accident. An extension of the Completion Time for the performance of the Required Action will not alter the capability of the mitigatory structures, systems or components from that assumed in establishing the Completion Time in the current Technical Specifications (CTS). Therefore, any consequences considered in the acceptance of the CTS Completion Time will not be significantly increased as a result of the adoption of the ITS 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 proper ACTIONS are taken for an inoperable control rod resulting in a loss of shutdown margin. 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?**

If an inoperable control rod results in a loss of shutdown margin, forcing a shutdown of the reactor may diminish the remaining control capability. However, allowing a short period to restore the required shutdown margin will, with high probability, result in restoration of the lost shutdown margin. This alternate action will also minimize the potential for plant transients that can occur during unit shutdown. As such, any perceived reduction in a margin to safety associated with the extended operating period will be offset with the benefit gained in avoiding a potential plant transient. Therefore, this change does not involve a significant reduction in a margin of safety.

## 3.1 L6

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

An extension of the Completion Time for a Required Action does not result in any hardware changes. The Completion Time for performance of the Required Action 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. Further, the extension of the Completion Time is not associated with the assumed initiation of any evaluated event. 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. 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. Nor does the extension in the Completion Time significantly change the response of the core parameters to assumed scenarios, from that considered during the original Completion Time. Thus, the extension in Completion Time will not result in either a significant increase in probability or consequences of any evaluated accident.

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

The proposed extension of the Completion Time 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 extension in 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 extension of the Completion Time interval does not involve a significant reduction in the margin of safety.

# NO SIGNIFICANT HAZARDS CONSIDERATIONS STATEMENTS

3.1 L7

**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 Required Actions to verify adequate SHUTDOWN MARGIN (SDM), or completion of actions necessary to establish boration used to restore adequate SDM, are themselves not initiators of any evaluated accident. This change does not result in any hardware changes or physical alteration of the unit. Thus, the Completion Time for performance of the Required Action does not significantly increase the probability of occurrence of any analyzed event since the function of the limit on SDM does not change. The Completion Time for performance of Required Actions does not significantly increase the consequences considered while establishing the CTS Completion Time because the extension in the Completion Time does not change the assumed response of any structure, system or component in performing its specified mitigatory function from that considered during approval of the original CTS 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 extension of the Completion Time interval does not involve a significant reduction in the margin of safety.

## 3.1 L8

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

The CONTROL RODS are used to support mitigation of the consequences of an accident; however, the mitigation is supported by all CONTROL RODS which are capable of inserting into the core when required. A method of determining SHUTDOWN MARGIN which does not consider the availability of all such rods, except the assumed stuck rod, is overly conservative. Further, such inoperable CONTROL RODS are not considered an initiator of any previously analyzed accident. As such the proposed change in the method of determining the SHUTDOWN MARGIN with inoperable, but trippable, CONTROL RODS will not significantly increase the probability of any accident previously evaluated. The proposed change allows for consideration of all trippable CONTROL RODS, except one assumed stuck rod, in the determination of SHUTDOWN MARGIN. This is consistent with the analysis for determining the consequences of previously analyzed accidents. 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 continue to provide adequate SHUTDOWN MARGIN to assure the reactor is subcritical following a reactor trip. Thus, this change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

The margin of safety provided by the defined SHUTDOWN MARGIN continues to be provided consistent with the safety analyses when considering all trippable CONTROL RODS. Therefore, this change does not involve a significant reduction in a margin of safety.

# NO SIGNIFICANT HAZARDS CONSIDERATIONS STATEMENTS

## 3.1 L9

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

The parameter of moderator temperature coefficient (MTC) is an initial assumption of the safety analyses, but it is not considered as an initiator of any previously analyzed accidents. As such, the allowed increase in MTC during MODE 2 physics testing will not involve a significant increase in the probability of any previously evaluated accident. Because of the impact of MTC on reactivity control during an event, a change in MTC alone may significantly impact analyzed consequences of accidents. However, the preservation of SHUTDOWN MARGIN requirements, limitations on THERMAL POWER generation, and adherence to approved, written procedures whose requirements were evaluated under 10CFR50.59, compensate for the potential increase in MTC above its limits for the short duration of physics testing. Therefore, the allowed increase in MTC during MODE 2 physics testing will 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 continue to preserve the reactor protection criteria. 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 allowed increase in MTC during MODE 2 physics testing may result in a small reduction of the margin of safety for this specific parameter; however, the other parameters controlled by the physics testing exception LCO, along with the other unchanged LCO requirements, the preservation of SHUTDOWN MARGIN requirements, limitations on THERMAL POWER generation, and adherence to approved, written procedures whose requirements were evaluated under 10CFR50.59, are sufficient to prevent a significant decrease in a margin of safety.

# NO SIGNIFICANT HAZARDS CONSIDERATIONS STATEMENTS

## 3.1 L10

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

The requirement for individual control rod and axial power shaping rod alignment with their respective group average position is an initial assumption of the safety analyses. Further, individual control rod and axial power shaping rod misalignments are evaluated events in the accident analyses. However, the purpose of these physics testing exceptions is to specifically allow the measure and verification of fundamental core operating characteristics under careful, administratively controlled conditions, so as to confirm the adequacy of the design methods and models used to establish the operating limits for the unit. Because of the impact the control rod and axial power shaping rods potentially have on core reactivity conditions and core power distribution conditions, specific requirements on SDM and core THERMAL POWER levels are established. Specific LCO provisions and Required Actions have been established should the physics testing LCO provisions not be met. Thus, the allowed exceptions to the control rod and axial power shaping rod alignment limits during physics testing will not involve a significant increase in the probability of any accident previously evaluated. During the conduct of the physics testing with the control rod and axial power shaping rod alignment limits not met, adverse conditions may exist such that reactivity control and power distribution would adversely affect the consequences of certain postulated accidents. However, other parameters are additionally limited during the proposed physics testing and specific THERMAL POWER limitations are imposed to compensate for the potential increase adverse consequences. Therefore, the allowed exceptions to control rod and axial power shaping rod alignment requirements during physics testing will 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 continue to preserve the reactor protection criteria. 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 allowed exceptions to control rod and axial power shaping rod alignment limits during physics testing may result in a small reduction of the margin of safety for specific parameters; however, the other parameters controlled by the physics test exception LCO along with the other unchanged LCO requirements, are sufficient to preserve the available margins of safety before exceeding the reactor protection criteria. Therefore, this change does not involve a significant reduction in a margin of safety.

# NO SIGNIFICANT HAZARDS CONSIDERATIONS STATEMENTS

## 3.1 L11

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

The detail concerning performance of a CHANNEL CHECK on the required channel of position indication is not associated with the initiation of any evaluated accident. Thus, the removal of this detail will not alter the assumed frequency of initiation of an evaluated accident. In addition, the removal of this detail will not allow unit operation in a manner other than that presently allowed. Further, no reduction in requirements will exist with regard to the requirement to determine rod position. Thus, the removal of detail concerning the performance of a CHANNEL CHECK will not result in a significant increase in the probability of any evaluated accident. The detail associated with the performance of the CHANNEL CHECK does not serve a mitigatory function and does not alter the assumed ability to verify OPERABILITY of the required position indication channel. As long as the rod positions are determined to be within limits using OPERABLE position indication channels, the detail of the performance of the CHANNEL CHECK does not impact analyzed consequences of accidents. Therefore, the removal of the detail regarding performance of the CHANNEL CHECK 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 continue to preserve the reactor protection criteria. Thus, this change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

The margin of safety for control rods is provided by the position of the rods, not the position indication. As long as the position of the rod can be accurately determined, the reactor protection criteria are preserved. The removal of the CHANNEL CHECK detail will not alter the ability to determine the rod position. Therefore, this change does not involve a significant reduction in a margin of safety.

## 3.1 L12

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

The removal of the requirement to perform a freedom of movement verification on CONTROL RODS while in MODES 3 and 4 does not result in any hardware changes, result in a physical alteration of the plant, or involve a change in the controls governing normal operation. The deletion of this requirement in MODES 3 and 4 removes a surveillance requirement applied to components that are not required to be OPERABLE in MODES 3 and 4. Thus, the removal of this requirement in these operational MODES does not alter the assumed initiation of any evaluated accident. Hence, this change does not involve a significant increase in probability. Further, the CONTROL ROD freedom of movement in MODES 3 and 4 is not associated with the mitigatory actions established in any analyzed accident in these MODES. Therefore, the removal of the freedom of movement verification in MODES 3 and 4 will not result in a significant increase in consequences of a previously evaluated accident.

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

The proposed change does not necessitate a physical alteration of the plant (no new or different type of equipment will be installed) or changes in parameters governing normal plant operation. The proposed change will continue to preserve the reactor protection criteria in those MODES in which the control rods were assumed to provide a mitigatory function. Thus, this change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

The margin of safety established by the control rods is provided by the ability to fully insert the control rods on a reactor trip. This feature will be retained in those MODES in which the control rods are assumed to serve a mitigatory function. However, in those MODES where the control rods are not assumed to provide a mitigatory function, the deletion of the freedom of movement surveillance requirement does not result in a degradation of any margin of safety that may be afforded by the control rods. Thus, in MODES 3 and 4, the removal of this surveillance requirement does not involve a significant reduction in a margin of safety.

## 3.1 L13

**NOT USED**

# NO SIGNIFICANT HAZARDS CONSIDERATIONS STATEMENTS

## 3.1 L14

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

The removal of the shutdown action statements associated with non-compliance with the dissolved gas concentration limits does not result in any hardware changes, result in a physical alteration of the plant, or involve a change in the controls governing normal operation. Thus, the removal of these statements does not alter the assumed initiator of any evaluated accident, and hence, does not involve a significant increase in the probability of any previously evaluated accident. The relocation of the dissolved gas concentration requirements to the TRM will continue to ensure that appropriate limits and associated actions are established for proper operation of the control rod drive(s) and/or the control rod(s). Further, the plausible consequences associated with a failure to comply with the concentration limits will not result in the failure of the control rods to perform their intended safety function. Therefore, the removal of the shutdown action statements will not result in a significant increase in consequences of a previously evaluated accident.

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

The proposed change does not necessitate a physical alteration of the plant (no new or different type of equipment will be installed) or changes in parameters governing normal plant operation. The proposed change will continue to preserve the operating restrictions on reactor coolant dissolved gas concentrations. The proposed change will continue to impose actions to mitigate the consequences of the out-of-limit condition. Thus, this change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

The margin of safety established by the dissolved gas concentration limits will be preserved by the requirements relocated to the Technical Requirements Manual. Appropriate remedial actions will continue to be provided should an out-of-limit condition develop. The operating restrictions provide protection for the control rod drive(s) and/or control rod(s) should a reactor trip occur while the control rod drive pressure boundary housing was filled by non-condensable gas. However, the credible damage mechanisms to the control rod drive(s) and/or control rod(s) do not affect the ability of the control rods to perform their intended safety function. Thus, the removal of the shutdown actions does not involve a significant reduction in a margin of safety.