SUMMARY OF CHANGES TO ITS SECTION 3.1

SUMMARY OF CHANGES TO ITS SECTION 3.1 - REVISION D

Source of Change	Summary of Change	Affected Pages
TSTF-222, Rev. 1	The current words of SR 3.1.4.1 require each control rod to be tested if any fuel movement in the reactor pressure vessel (RPV) occurs. This effectively means that even if only one bundle is moved (e.g., replacing a leaking fuel bundle mid-cycle), all the control rods are required to be tested per the words of the SR. While a generic change to the Bases attempted to ensure that only those rods affected be tested, the current Bases words do not preclude misinterpretation of this requirement. The actual SR was not previously modified generically and continues to require each rod to be tested. In addition, SR 3.1.4.3 requires that only the affected control rods to be tested, further adding confusion. Therefore, the first frequency of SR 3.1.4.1 is moved to SR 3.1.4.4 and modified to read "associated core cell" in lieu of "reactor pressure vessel." The Bases for SR 3.1.4.4 will state that it is expected that during a routine refueling outage, all control rods will be affected. Thus, the requirement to test all the control rods remains essentially unchanged.	**Section 3.1.4** CTS mark-up, pp 1, 2 of 3 DOC M3 (DOC p 3 of 7); DOC M4 (DOCs p 3 of 7); DOC M6 (DOCs p 4 of 7) ITS mark-up, pp 3.1-12, 3.1-13 JFD TA1 (JFDs p 1 of 1), JFD X1 (JFDs p 1 of 1) ITS Bases mark-up, pp B 3.1-25, B 3.1-27 Bases JFD TA1 (JFDs p 2 of 2), Bases JFD X2 (JFDs p 2 of 2) Retyped ITS pp 3.1-12, 3.1-13 Retyped ITS Bases pp B 3.1-25, B 3.1-27
TSTF-367, Rev. 0	The majority of the ITS Bases state that the Specification satisfies Criterion 1, 2 or 3. Rev. 1 of the ISTS NUREG does not make reference to Criterion 4, but several specifications state that the specification is retained because it is risk significant. This generic change revised the Bases to make reference to Criterion 4 of the NRC Policy Statement, where appropriate, for consistency with the remainder of the specifications and with the final version of the NRC Policy Statement. Consistent with this TSTF, the JFD reference in the "Applicable Safety Analysis" Bases for the proposed change that includes Criterion 4 is revised to include TA1(a new JFD). The reference to JFD X1 is intentionally maintained.	**Section 3.1.7** ITS Bases mark-up, p B 3.1-40 Bases JFD TA1 (JFDs p 2 of 2) Retyped ITS Bases p B 3.1-40
RAI 3.1-1	DOC A4 was developed to explicitly address the replacement of the term "monitored rod density" with the term "measured rod density."	**Section 3.1.2** CTS mark-up, p 1 of 1 DOC A4 (DOCs p 1 of 4)

SUMMARY OF CHANGES TO ITS SECTION 3.1 - REVISION D

Source of Change	Summary of Change	Affected Pages
RAI 3.1-3	The proposed change by the Authority to Required Action A.3 is withdrawn. The withdrawal of the proposed change will make Required Action A.2 consistent with SR 3.1.3.2 & SR 3.1.3.3 in that each of these SRs contains wording which makes it apparent that one SR is for a fully withdrawn rod and the other SR is for a partially withdrawn rod. Accordingly, the use of the phrase "for each fully withdrawn Operable control rod" directly after SR 3.1.3.2 and the use of the phrase "partially" directly after the word "each" in Required Action A.3 is not needed. In addition, consistent with the deletion of this change, JFD PA1 is indicated as "not used."	**Section 3.1.3** ITS mark-up, p 3.1-8 JFD PA1 (JFDs p 1 of 1) Retyped ITS p 3.1-8
RAI 3.1-4	Bases Actions A.1, A.2 and A.3 of Specification 3.1.3 will be revised to remove the proposed changes associated with "cold shutdown condition" and "hot subcritical"; restoring ISTS wording to "MODE 4" and "MODE 5", respectively.	**Section 3.1.3** ITS Bases mark-up, p B 3.1-16 Retyped ITS Bases p B 3.1-17
RAI 3.1-5	Required Action C.1 of Specification 3.1.5 will be revised to remove the proposed changes and thereby restore ISTS wording. Specifically, the previous change was "Verify associated control rods are fully inserted." The Staff indicated that they were uncertain of the enhancement and recognized that the Authority's previously proposed change was more concise but not necessarily precise. Accordingly, Action C.1 is restored to the wording of the ISTS. Consistent with this revision to Action C.1 of ITS 3.1.5, JFD PA2 is revised to state "not used."	**Section 3.1.5** ITS mark-up, p 3.1-17 JFD PA2 (JFDs p 1 of 1) Retyped ITS p 3.1-16

SUMMARY OF CHANGES TO ITS SECTION 3.1 - REVISION D

Source of Change	Summary of Change	Affected Pages
RAI 3.1-6	For Actions A.1 and A.2 Bases of ITS 3.1.6, JAF previously proposed deletion of the following sentence: "When the control rod pattern is not in compliance with the prescribed sequence, all control rod movement should be stopped except for moves needed to correct the rod pattern, or scram if warranted." This previous change was made to be consistent with the Actions in the ITS. Specifically, the sentence was deleted because the Actions do not require that all control rod movement be stopped when the control rod pattern is not in compliance with the prescribed sequence. The Staff responded that in fact the Action does not require that all control rod movement be stopped. The NRC further said that Action A.1 states, "Move associated control rod(s) to correct position." Accordingly, the Staff concluded that the Bases statement proposed for deletion by JAF supports the Required Action and should be retained. JAF responded that it agrees with the Staff's determination as stated in their comments associated with this RAI that "In fact the Action does not require that all control rod movement be stopped." Consistent with the Staff's determination, the Bases sentence is viewed as a recommendation and not a requirement. Accordingly, with this mutual understanding regarding the relationship of the Bases sentence and the Technical Specification Action, JAF will revise the Bases to retain the sentence that was previously deleted.	**Section 3.1.6** ITS Bases mark-up, p B 3.1-36 Bases JFD PA3 (JFDs p 1 of 2) Retyped ITS Bases p B 3.1-36

SUMMARY OF CHANGES TO ITS SECTION 3.1 - REVISION D

Source of Change	Summary of Change	Affected Pages
RAI 3.1-7	For Actions A.1 and A.2 Bases of ITS 3.1.6, the Authority previously proposed deletion of the following sentence: "OPERABILITY of control rods is determined by compliance with LCO 3.1.3, "Control Rod OPERABILITY," LCO 3.1.4, "Control Rod Scram Times," and LCO 3.1.5, "Control Rod Scram Accumulators." This previous change was made based on the perception that the sentence was not in the correct location. The Staff responded by saying that the Authority should consider moving the sentence to the LCO Bases section where it would be more appropriate. After further consideration of this matter, the Authority has decided to retain the sentence that was previously proposed for deletion in its original location. This retained sentence is viewed in the context with the previous sentence in the Bases. Specifically, the Bases states that a control rod not in compliance with the prescribed sequence is not considered inoperable except as required by Required Action A.2. The location of the retained sentence is appropriate when viewed in relationship to this previous sentence. Accordingly, the Authority will adopt the standard wording of the ISTS with regards to this matter. Consistent with this proposed change, JFD PA4 is revised to state "not used."	**Section 3.1.6** ITS Bases mark-up, p B 3.1-36 Bases JFD PA4 (JFDs p 1 of 2) Retyped ITS Bases pp B 3.1-36, B 3.1-37

SUMMARY OF CHANGES TO ITS SECTION 3.1 - REVISION D

Source of Change	Summary of Change	Affected Pages
RAI 3.1-8	With regards to ITS SR 3.1.7.8 Frequency, CTS 4.4.A.5 and DOC M5, the Authority proposed an SR frequency change which involves going from "24 months" in the CTS to "24 months on a staggered test basis" in the ITS. This change was previously classified as more restrictive since it "adds a more prescriptive requirement." The Staff disagreed and stated that the change is mis-categorized since it in reality decreases the frequency each subsystem is tested as testing is proposed to go from 24 to 48 months. The NRC requested additional justification for this change. The Authority stated that it would revise the submittal to eliminate the CTS 4.4.A.5 reference to DOC M5 and replace it with a new L DOC (i.e., L5). DOC L5 provides the justification for extending the test interval for the SLC System Valve(s) that are not verified unblocked by other surveillances every 24 months (e.g., SR 3.1.7.9). This justification provides sufficient basis for determination that the reliability of the system will not be adversely impacted.	**Section 3.1.7** CTS mark-up, p 2 of 5 DOC M5 (DOCs p 3 of 7); DOC L5 (DOCs p 7 of 7) NHSC L5 CHANGE (NHSCs pp 9, 10 of 10)
License Amendment Number 255	The CTS MU pages were revised for ITSs 3.1.1, 3.1.3, 3.1.5, and 3.1.8 to reflect issuance of this amendment.	**Section 3.1.1** CTS mark-up, p 2 of 2 **Section 3.1.3** CTS mark-up, p 2 of 4 **Section 3.1.5** CTS mark-up, p 2 of 2 **Section 3.1.8** CTS mark-up, p 2 of 4
Typographical Correction	For DOC M3 of ITS 3.1.2, the parenthetical phrase "(control rod placement)" is replaced with "(control rod replacement)". This correction makes the DOC consistent with the markup of the CTS page.	**Section 3.1.2** DOC M3 (DOCs p 2 of 4)
Typographical Correction	For item (4) of DOC A1 of ITS 3.1.3, the "less than" symbol is replaced with the "less than or equal to" symbol. This change makes the DOC consistent with the note to Condition "D" of ITS 3.1.3.	**Section 3.1.3** DOC A1 (DOCs p 1 of 9)

SUMMARY OF CHANGES TO ITS SECTION 3.1 - REVISION D

Source of Change	Summary of Change	Affected Pages
Typographical Correction	For Action A3 of ITS 3.1.3, minor punctuation change. Removed the period from the end of the Completion Time statement.	**Section 3.1.3** Retyped ITS p 3.1-8
Editorial Correction	Removed the parenthetical reference to 8 x 8 fuel from Bases markup insert ASA1 and corresponding ASA, second paragraph, last sentence.	**Section 3.1.6** ITS Bases mark-up, p Insert Page B 3.1-34 Retyped ITS Bases p B 3.1-35
ITS Figure 3.1.7-1 Editorial Corrections	Regarding ITS Figure 3.1.7-1, this Figure which is part of the ITS MU did not match the version that was in the clean typed ITS. In addition to editorial changes associated with the reconciliation of the differences between both versions of this Figure, further editorial changes were made to both the markup and clean typed figures. Changes include the following: (1) The header of the Figure "Sodium Pentaborate Solution (Minimum 34.7 B-10 Atom% Enriched) Volume Concentration Requirements" has been deleted; (2) The title of the vertical axis "Weight Percent of Sodium Pentaborate in Solution (Minimum Enrichment 34.7 B-10 Atom Percent)" is replaced with "Concentration Weight Percent of Enriched Sodium Pentaborate"; (3) The coordinates listed at the four points of the trapezoid shown on the Figure have been individually re-sequenced such that the gallon number appears first and the percentage number appears second consistent with the convention of showing the "x" coordinate first (4) Typographical corrections are made to the two lower sets of coordinates for the trapezoid (i.e., a comma is deleted on the lower left corner and one of two parenthesis is deleted on the lower right corner).	**Section 3.1.7** ITS mark-up, p Insert Page 3.1-23 Retyped ITS p 3.1-23

SUMMARY OF CHANGES TO ITS SECTION 3.1 - REVISION D

Source of Change	Summary of Change	Affected Pages
ITS Figure 3.1.7-2 Editorial Corrections	Regarding ITS Figure 3.1.7-2, this Figure which is part of the ITS MU did not match the version that was in the clean typed ITS. In addition to editorial changes associated with the reconciliation of the differences between both versions of this Figure, further editorial changes were made to both Figures. All the changes include the following: (1) The parenthetical phrase "includes 10 degrees F Margin" on the vertical axis was inadvertently added without noting DOC LA4 relocated the same information to the Bases. Accordingly, this parenthetical phrase is deleted from the Figure.;(2) The horizontal axis of the Figure is revised from "Percent Enriched Sodium Pentaborate by Weight of Solution" to "Concentration (Weight Percent Enriched Sodium Pentaborate)"; and (3) The header of the Figure "Saturation Temperature of Enriched Sodium Pentaborate Solution (includes 10 degree F Margin)" has been deleted.	**Section 3.1.7** ITS mark-up, p Insert Page 3.1-24 Retyped ITS p 3.1-24
Revised SR 3.1.7.10	As presently written, SR 3.1.7.10 and the corresponding Bases could be interpreted as requiring testing the sodium pentaborate being added for Boron-10 enrichment each and every time an addition is made. (This is the interpretation given in the existing DOC.) This interpretation imposes considerable burden with no corresponding value added. The DOC is rewritten and the Bases are modified to recognize that enrichment does not change with time for a given batch of sodium pentaborate. Accordingly, a single isotopic test, including vendor certified analytical test results, may be used to satisfy SR 3.1.7.10 requirements.	**Section 3.1.7** CTS mark-up, p 3 of 5 DOC M4 (DOCs pp 2. 3 of 7) ITS mark-up, pp 3.1-22 JFD DB5 (JFDs p 1 of 2) ITS Bases mark-up, p B 3.1-46; Insert Page B 3.1-46 Bases JFD PA4 (JFDs p 1 of 2) Retyped ITS Bases p B 3.1-45

SUMMARY OF CHANGES TO ITS SECTION 3.1 - REVISION D

Source of Change	Summary of Change	Affected Pages
New SR 3.1.7.11	Retains the CTS requirement for verifying the Boron-10 enrichment of the solution in the SLC tank on a 24 month frequency. This requirement was previously eliminated on the basis of sampling and testing Boron-10 enrichment of additions to the tank every time an addition is made (as per previous DOC M4). The retention of the requirement reflects current and ongoing practice, and reflects the prudent action of periodically verifying the parameter of interest, the Boron-10 content of the SLC solution.	**Section 3.1.7** CTS mark-up, p 3 of 5 ITS mark-up, pp 3.1-22, Insert Page 3.1-22 JFD CLB3 (JFDs p 1 of 2) ITS Bases mark-up, pp B 3.1-46; Insert Page B 3.1-46 Bases JFD CLB3 (JFDs p 1 of 2) Retyped ITS p 3.1-22 Retyped ITS Bases p B 3.1-45
Modification JD-99-020	The present Scram Timing Program measures scram times to all notches but is set up to compare measured times to the notches listed with CTS acceptance criteria. ITS acceptance criteria have different numbers (i.e., times) and are based on times to different notches than CTS; accordingly, the previous deletion of the sentence in Bases SR 3.1.4.2 regarding data from inadvertent scrams. However, Modification JD-99-020, a new scram timing program, will satisfy ITS requirements. The official use of this program will not take place until implementation of the ITS. Therefore, the deleted sentence in the Bases of SR 3.1.4.2 is restored in anticipation of this implementation effort. Also, JFD DB4 was developed to document the rationale behind the restoration of this ITS Bases wording.	**Section 3.1.4** ITS Bases mark-up, p B 3.1-26 Bases JFD DB4 (Bases JFDs p 1 of 2) Retyped ITS Bases p B 3.1-26

ITS CONVERSION PACKAGE

SECTION 3.1 - REACTIVITY CONTROL SYSTEMS

JAFNPP
IMPROVED TECHNICAL SPECIFICATION (ITS)

CONVERSION PACKAGE

Section 3.1 - REACTIVITY CONTROL SYSTEMS

Table of Contents

The markup package for each Specification contains the following:

Markup of the current Technical Specifications (CTS);
Discussion of changes (DOCs) to the CTS;
No significant hazards consideration (NSHC) for each less restrictive change (Lx) to the CTS;
Markup of the corresponding NUREG-1433 Specification;
Justification of differences (JFDs) from the NUREG;
Markup of NUREG-1433 Bases;
Justification for differences (JFDs) from NUREG-1433 Bases; and
Retyped proposed Improved Technical Specifications (ITS) and Bases.

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IMPROVED STANDARD TECHNICAL SPECIFICATIONS (ISTS) CONVERSION

ITS: 3.1.1

SHUTDOWN MARGIN (SDM)

MARKUP OF CURRENT TECHNICAL SPECIFICATIONS (CTS)

DISCUSSION OF CHANGES (DOCs) TO THE CTS

NO SIGNIFICANT HAZARDS CONSIDERATION (NSHC) FOR LESS RESTRICTIVE CHANGES

MARKUP OF NUREG-1433, REVISION 1, SPECIFICATION

JUSTIFICATION FOR DIFFERENCES (JFDs) FROM NUREG-1433, REVISION 1
MARKUP OF NUREG-1433, REVISION 1, BASES

JUSTIFICATION FOR DIFFERENCES (JFDs) FROM NUREG-1433, REVISION 1, BASES

RETYPED PROPOSED IMPROVED TECHNICAL SPECIFICATIONS (ITS) AND BASES

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IMPROVED STANDARD TECHNICAL SPECIFICATIONS (ISTS) CONVERSION

ITS: 3.1.1

SHUTDOWN MARGIN (SDM)

MARKUP OF CURRENT TECHNICAL SPECIFICATIONS (CTS)

Specification 3.1.1

A1

~~3.3 LIMITING CONDITION FOR OPERATION~~

[3.1] ~~3.3~~ REACTIVITY CONTROL SYSTEMS

Applicability: MODES 1, 2, 3, 4 and 5

Applies to the operational status of the Control Rod System.

Objective:

To assure the ability of the Control Rod System to control reactivity.

Specification:

[3.1.1] ~~A.~~ Reactivity Limitations ← Shutdown Margin (SDM)

1. Reactivity margin - core loading

add LCO and limits A2

[LCO 3.1.1] A sufficient number of control rods shall be operable so that the core could be made subcritical in the most reactive conditions during the operating cycle with the strongest control rod fully withdrawn and all other operable control rods fully inserted.

LA1

add ACTIONS A, B, C, D and E

M1

~~4.3 SURVEILLANCE REQUIREMENT~~

4.3 REACTIVITY CONTROL

Applicability:

Applies to the surveillance requirements of the Control Rod System.

Objective:

To verify the ability of the Control Rod System to control reactivity.

Specification:

A. Reactivity Limitations

1. Reactivity margin - core loading

LA1

[SR 3.1.1.1] Sufficient control rods shall be withdrawn following a refueling outage when core alterations were performed to demonstrate with a margin of 0.38 percent Δk/k the core can be made subcritical at any time in the subsequent fuel cycle with the analytically determined strongest control rod fully withdrawn and all other operable rods fully inserted.

LA1

add proposed SR 3.1.1.1 first frequency

M2

add proposed Second frequency

Page 1 of 2

~~3.3.A.2 (cont'd)~~

<See ITS 3.1.3>

b. The control rod directional control valves for inoperable control rods shall be disarmed electrically.

<see ITS 3.1.3>

c. Control rods with scram times greater than those permitted by Specification 3.3.C.3 are inoperable, but if they can be inserted with control rod drive pressure they need not be disarmed electrically.

d. Control rods with inoperable accumulators or those whose position cannot be positively determined shall be considered inoperable.

<see ITS 3.1.3; 3.1.5>

[ACTION A]

e. ~~Inoperable control rods shall be positioned such that Specification 3.3.A.1 is met.~~

Restore SDM to within limits in 6 hours

M1

(1) When operating with two or more inoperable control rods in the Startup/Hot Standby or Run modes at ≤ 10% rated thermal power, control rod patterns shall be equivalent to those prescribed by the Banked Position Withdrawal Sequence (BPWS) or else the inoperable control rods shall be separated by two or more operable control rods. If this condition is not met, restore compliance with the condition within 4 hours. Otherwise be in hot shutdown within the following 12 hours.

<ADD ACTION B> M1

(2) If nine or more control rods are inoperable, be in hot shutdown within 12 hours.

<see ITS 3.1.3>

~~4.3.A.2 (cont'd)~~

<See ITS 3.1.8>

e. The scram discharge volume drain and vent valves shall be full-travel cycled at least once per quarter to verify that the valves close in less than 30 seconds and to assure proper valve stroke and operation.

f. An instrument check of control rod position indication shall be performed once/day.

<see ITS 3.1.3>

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IMPROVED STANDARD TECHNICAL SPECIFICATIONS (ISTS) CONVERSION

ITS: 3.1.1

SHUTDOWN MARGIN (SDM)

DISCUSSION OF CHANGES (DOCs) TO THE CTS

DISCUSSION OF CHANGES
ITS: 3.1.1 - SHUTDOWN MARGIN (SDM)

ADMINISTRATIVE CHANGES

A1 In the conversion of the James A. FitzPatrick Nuclear Power Plant (JAFNPP) Current Technical Specification (CTS) to the proposed plant specific Improved Technical Specifications (ITS) certain wording preferences or conventions are adopted which do not result in technical changes. Editorial changes, reformatting, and revised numbering are adopted to make the ITS consistent with the conventions in NUREG-1433, "Standard Technical Specifications, General Electric Plants, BWR/4", Revision 1 (i.e., Improved Standard Technical Specifications (ISTS)).

A2 CTS 3.3.A.1 requires a sufficient number of control rods to be Operable so that the core could be made subcritical in the most reactive condition during the operating cycle with the strongest control rod fully withdrawn and all other operable control rods fully inserted. ITS LCO 3.1.1 requires the Shutdown Margin (SDM) to be ≥ 0.38% Δk/k, with the highest worth control rod analytically determined. The proposed LCO requirements (and limits) are consistent with the wording in CTS 4.3.A.1. The details that the SDM should be met in the most reactive condition during the operating cycle with the strongest control rod fully withdrawn and all other operable control rods fully inserted has been relocated to the Bases (LA1). Since the proposed LCO is consistent with CTS 4.3.A.1 this change is considered administrative. This change is consistent with NUREG-1433, Revision 1.

TECHNICAL CHANGES - MORE RESTRICTIVE

M1 CTS 3.3.A.2.e requires inoperable control rods to be positioned so that CTS 3.3.A.1(ITS LCO 3.1.1) is met. No specific time limit is provided. The following changes were made to the current Technical Specifications:

- If SDM is not met while the plant is in MODE 1 or 2, the proposed Actions (ITS 3.1.1 ACTIONS A and B) would require the SDM to be restored in 6 hours (ACTION A) or be in MODE 3 in the following 12 hours (ACTION B). Since the current requirements do not specify an explicit time to restore SDM in CTS 3.3.A.2.e, the proposed limit is considered more restrictive. The proposed time in ITS 3.1.1 ACTION B is consistent with the time to reach hot shutdown in CTS 3.3.A.2.e.(1) and CTS 3.3.A.2.e.(2). In addition, once in MODE 3, if the SDM was still not met, the Actions (ITS 3.1.1 ACTION C) would require the insertion of all insertable control rods. This action further enhances the available SDM.

TECHNICAL CHANGES - MORE RESTRICTIVE

M1 (continued)

- If SDM is not met in MODE 4 or 5, Actions (ITS 3.1.1 ACTIONS D and E) are provided to initiate action to insert all insertable control rods (in the core cells containing fuel if in MODE 5), to suspend CORE ALTERATIONS (if applicable), and to initiate actions within 1 hour to restore secondary containment, SGT System and the Secondary Containment Isolation Valves (SCIVs) to Operable status. The first two actions attempt to restore SDM, or at least to ensure SDM is not further reduced, while the last two actions provide protection from radioactive release if a SDM problem results in an inadvertent criticality.

These Actions are more restrictive since new requirements are added that currently do not exist. These changes are necessary to ensure that appropriate ACTIONS are taken when SDM is not within limits.

M2 CTS 4.3.A.1 requires that SDM be verified following a refueling outage. ITS SR 3.1.1.1 requires SDM to be verified once within 4 hours after criticality following fuel movement within the reactor pressure vessel or control rod replacement and prior to each in vessel fuel movement during the fuel loading sequence.

Therefore, a finite time (4 hours after criticality) is now provided to verify SDM following a refueling outage. In addition, a new Surveillance Frequency for SDM verification has been added to clarify the requirements necessary for assuring SDM during the refueling process. Because SDM is assumed in several refueling mode analyses in the UFSAR, some measures must be taken to ensure the intermediate fuel loading patterns during refueling have adequate SDM. This change imposes a requirement where none is explicitly provided in the CTS. This new requirement does not, however, require introducing tests or modes of operation of a new or different nature than currently exist.

As presented in the Bases corresponding to this requirement, this is best accomplished by analysis (rather than in-sequence criticals) because of the many changes in the core loading during a typical refueling. Bounding analyses may be used to demonstrate adequate SDM for the most reactive configurations during refueling thereby showing acceptability of the entire fuel movement sequence.

DISCUSSION OF CHANGES
ITS: 3.1.1 - SHUTDOWN MARGIN (SDM)

TECHNICAL CHANGES - LESS RESTRICTIVE (GENERIC)

LA1 Details of the method to perform the Surveillance in CTS 4.3.A.1 (e.g., sufficient control rods shall be withdrawn) and the details of the requirements in CTS 3.3.A.1 and 4.3.A.1 (e.g., that the shutdown margin shall be met at any time in the subsequent fuel cycle) are proposed to be relocated to the Bases. These details are not necessary to ensure the SDM is verified to be within limits. The requirement of ITS SR 3.1.1.1 to verify that SDM is within limits and the definition of Shutdown Margin in ITS Chapter 1.0 are adequate to ensure the test is performed properly. Therefore, the relocated details are not required to be in the ITS to provide adequate protection of the public health and safety. Changes to the Bases will be controlled by the provisions of the proposed Bases Control Program described in Chapter 5 of the Technical Specifications.

TECHNICAL CHANGES - LESS RESTRICTIVE (SPECIFIC)

None

TECHNICAL CHANGES - RELOCATIONS

None

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IMPROVED STANDARD TECHNICAL SPECIFICATIONS (ISTS) CONVERSION

ITS: 3.1.1

SHUTDOWN MARGIN (SDM)

NO SIGNIFICANT HAZARDS CONSIDERATION (NSHC) FOR LESS RESTRICTIVE CHANGES

NO SIGNIFICANT HAZARDS CONSIDERATION
ITS: 3.1.1 - SHUTDOWN MARGIN (SDM)

TECHNICAL CHANGES - LESS RESTRICTIVE (SPECIFIC)

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

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IMPROVED STANDARD TECHNICAL SPECIFICATIONS (ISTS) CONVERSION

ITS: 3.1.1

SHUTDOWN MARGIN (SDM)

MARKUP OF NUREG-1433, REVISION 1 SPECIFICATION

[3.3] 3.1 REACTIVITY CONTROL SYSTEMS

[3.3.A] 3.1.1 SHUTDOWN MARGIN (SDM)

TA1 not shown

[3.3.A.1] LCO 3.1.1 SDM shall be CLB1

[3.3.A.1][A2]

a. ≥ 0.38% Δk/k, with the highest worth control rod analytically determined; TOD

b. ~~≥ [0.28]% Δk/k, with the highest worth control rod determined by test.~~ CLB1

[3.3] APPLICABILITY: MODES 1, 2, 3, 4, and 5.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
[M1] [3.3.A.2.e] A. SDM not within limits in MODE 1 or 2.	A.1 Restore SDM to within limits.	6 hours
[M1] B. Required Action and associated Completion Time of Condition A not met.	B.1 Be in MODE 3.	12 hours
[M1] C. SDM not within limits in MODE 3.	C.1 Initiate action to fully insert all insertable control rods.	Immediately
D. SDM not within limits in MODE 4.	D.1 Initiate action to fully insert all insertable control rods. AND	Immediately (continued)

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
D. (continued)	D.2	Initiate action to restore (secondary) containment to OPERABLE status.	1 hour PA1
	AND		
	D.3	Initiate action to restore one standby gas treatment (SGT) subsystem to OPERABLE status.	1 hour
	AND		
	D.4	Initiate action to restore isolation capability in each required (secondary) containment penetration flow path not isolated.	1 hour PA1
E. SDM not within limits in MODE 5.	E.1	Suspend CORE ALTERATIONS except for control rod insertion and fuel assembly removal.	Immediately
	AND		
	E.2	Initiate action to fully insert all insertable control rods in core cells containing one or more fuel assemblies.	Immediately
	AND		(continued)

[M1]

[M1]

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
E. (continued)	E.3	Initiate action to restore secondary containment to OPERABLE status.	1 hour PA1
	AND		
[M1]	E.4	Initiate action to restore one SGT subsystem to OPERABLE status.	1 hour
	AND		
	E.5	Initiate action to restore isolation capability in each required secondary containment penetration flow path not isolated.	1 hour PA1

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.1.1.1	Verify SDM is ~~[~~ CLB1 a. ≥ 0.38% Δk/k with the highest worth control rod analytically determined. or b. ~~≥ [0.28]% Δk/k with the highest worth control rod determined by test~~ CLB1	Prior to each in vessel fuel movement during fuel loading sequence AND Once within 4 hours after criticality following fuel movement within the reactor pressure vessel or control rod replacement

[4.3 A.1]

[M2]

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IMPROVED STANDARD TECHNICAL SPECIFICATIONS (ISTS) CONVERSION

ITS: 3.1.1

SHUTDOWN MARGIN (SDM)

JUSTIFICATION FOR DIFFERENCES (JFDs) FROM NUREG-1433, REVISION 1

JUSTIFICATION FOR DIFFERENCES FROM NUREG-1433, REVISION 1
ITS: 3.1.1 - SHUTDOWN MARGIN (SDM)

RETENTION OF EXISTING REQUIREMENT (CLB)

CLB1 ISTS LCO 3.1.1.b and SR 3.1.1.1.b have been deleted since JAFNPP performs all shutdown margin evaluations against the current 0.38% Δk/k limit (with the highest worth control rod analytically determined). ITS LCO 3.1.1 and SR 3.1.1.1 have been modified to reflect this change.

PLANT-SPECIFIC WORDING PREFERENCE OR MINOR EDITORIAL IMPROVEMENT (PA)

PA1 The brackets have been deleted and the proper plant specific terminology included.

PLANT-SPECIFIC DIFFERENCE IN THE DESIGN (DB)

None

DIFFERENCE BASED ON AN APPROVED TRAVELER (TA)

TA1 TSTF-09, Revision 1 relocates SDM limits to the CORE OPERATING LIMITS REPORT. For JAFNPP (a BWR) these limits are not cycle-specific. Therefore, the SDM limits are maintained in the JAFNPP ITS.

DIFFERENCE BASED ON A SUBMITTED, BUT PENDING TRAVELER (TP)

None

DIFFERENCE FOR ANY REASON OTHER THAN THE ABOVE (X)

None

JAFNPP

IMPROVED STANDARD TECHNICAL SPECIFICATIONS (ISTS) CONVERSION

ITS: 3.1.1

SHUTDOWN MARGIN (SDM)

MARKUP OF NUREG-1433, REVISION 1, BASES

B 3.1 REACTIVITY CONTROL SYSTEMS

B 3.1.1 SHUTDOWN MARGIN (SDM)

BASES

BACKGROUND

SDM requirements are specified to ensure:

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

b. The reactivity transients associated with postulated accident conditions are controllable within acceptable limits; and

c. The reactor will be maintained sufficiently subcritical to preclude inadvertent criticality in the shutdown condition.

the Updated Final Safety Analysis Report (UFSAR) Section 16.6 DBI

These requirements are satisfied by the control rods, as described in ~~GDC 26~~ (Ref. 1), which can compensate for the reactivity effects of the fuel and water temperature changes experienced during all operating conditions.

APPLICABLE SAFETY ANALYSES

The control rod drop accident (CRDA) analysis (Refs. 2 and 3) assumes the core is subcritical with the highest worth control rod withdrawn. Typically, the first control rod withdrawn has a very high reactivity worth and, should the core ~~be critical~~ during the withdrawal of the first control rod, the consequences of a CRDA could exceed the fuel damage limits for a CRDA (see Bases for LCO 3.1.6, "Rod Pattern Control"). Also, SDM is assumed as an initial condition for the control rod removal error during refueling (Ref. 4) and fuel assembly insertion error during refueling (Ref. 5) accidents. The analysis of these reactivity insertion events assumes the refueling interlocks are OPERABLE when the reactor is in the refueling mode of operation. These interlocks prevent the withdrawal of more than one control rod from the core during refueling. (Special consideration and requirements for multiple control rod withdrawal during refueling are covered in Special Operations LCO 3.10.6, "Multiple Control Rod Withdrawal—Refueling.") The analysis assumes this condition is acceptable since the core will be shut down

PA1

SDM be substantially less than 0.38% Δk/k

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

with the highest worth control rod withdrawn, if adequate SDM has been demonstrated.

Prevention or mitigation of reactivity insertion events is necessary to limit energy deposition in the fuel to prevent significant fuel damage, which could result in undue release of radioactivity. Adequate SDM ensures inadvertent criticalities and potential CRDAs involving high worth control rods (namely the first control rod withdrawn) will not cause significant fuel damage.

10 CFR 50.36 (c)(2)(ii) (Ref. 5)

SDM satisfies Criterion 2 of the NRC Policy Statement. X1

LCO

analysis or by a combination of test and analysis

The specified SDM limit accounts for the uncertainty in the demonstration of SDM by testing. Separate SDM limits are provided for testing where the highest worth control rod is determined analytically or by measurement. This is due to the reduced uncertainty in the SDM test when the highest worth control rod is determined by measurement. When SDM is demonstrated by calculations not associated with a test (e.g., to confirm SDM during the fuel loading sequence), additional margin is included to account for uncertainties in the calculation. To ensure adequate SDM during the design process, a design margin is included to account for uncertainties in the design calculations (Ref. 6).

A IS CLB1

Insert LCO

APPLICABILITY

In MODES 1 and 2, SDM must be provided because subcriticality with the highest worth control rod withdrawn is assumed in the CRDA analysis (Ref. 2). In MODES 3 and 4, SDM is required to ensure the reactor will be held subcritical with margin for a single withdrawn control rod. SDM is required in MODE 5 to prevent an open vessel, inadvertent criticality during the withdrawal of a single control rod from a core cell containing one or more fuel assemblies or a fuel assembly insertion error (Ref. 8). PA2

7 PA1

(continued)

CLB1

INSERT LCO

SDM is demonstrated by analysis or by a combination of test and analysis. During refueling it is demonstrated by analysis and during a startup it is demonstrated by a combination of test and analysis.

BASES (continued)

ACTIONS

A.1

With SDM not within the limits of the LCO in MODE 1 or 2, SDM must be restored within 6 hours. Failure to meet the specified SDM may be caused by a control rod that cannot be inserted. The allowed Completion Time of 6 hours is acceptable, considering that the reactor can still be shut down, assuming no failures of additional control rods to insert, and the low probability of an event occurring during this interval.

B.1

If the SDM cannot be restored, the plant must be brought to MODE 3 in 12 hours, to prevent the potential for further reductions in available SDM (e.g., additional stuck control rods). The allowed Completion Time of 12 hours is reasonable, based on operating experience, to reach MODE 3 from full power conditions in an orderly manner and without challenging plant systems.

C.1

With SDM not within limits in MODE 3, the operator must immediately initiate action to fully insert all insertable control rods. Action must continue until all insertable control rods are fully inserted. This action results in the least reactive condition for the core.

D.1, D.2, D.3, and D.4

With SDM not within limits in MODE 4, the operator must immediately initiate action to fully insert all insertable control rods. Action must continue until all insertable control rods are fully inserted. This action results in the least reactive condition for the core. Action must also be initiated within 1 hour to provide means for control of potential radioactive releases. This includes ensuring secondary containment is OPERABLE; at least one Standby Gas Treatment (SGT) subsystem is OPERABLE; and secondary containment isolation capability (i.e., at least one secondary containment isolation valve and associated instrumentation are OPERABLE, or other acceptable

PA1

is available

PA2

move to next page (A)

PA1

(continued)

BASES

Secondary Containment PA1

ACTIONS

These administrative controls consist of stationing a dedicated operator, who is in continuous communication with the control room, at the controls of the isolation device. In this way, the penetration can be rapidly isolated when a need for secondary containment isolation is indicated). PA1

D.1, D.2, D.3, and D.4 (continued)

administrative controls to assure isolation capability in each associated penetration flow path not isolated that is assumed to be isolated to mitigate radioactivity releases. This may be performed as an administrative check, by examining logs or other information, to determine if the components are out of service for maintenance or other reasons. It is not necessary to perform the surveillances needed to demonstrate the OPERABILITY of the components. If, however, any required component is inoperable, then it must be restored to OPERABLE status. In this case, SRs may need to be performed to restore the component to OPERABLE status. Actions must continue until all required components are OPERABLE.

Insert A from previous page PA1

E.1, E.2, E.3, E.4, and E.5

With SDM not within limits in MODE 5, the operator must immediately suspend CORE ALTERATIONS that could reduce SDM (e.g., insertion of fuel in the core or the withdrawal of control rods). Suspension of these activities shall not preclude completion of movement of a component to a safe condition. Inserting control rods or removing fuel from the core will reduce the total reactivity and are therefore excluded from the suspended actions.

Action must also be immediately initiated to fully insert all insertable control rods in core cells containing one or more fuel assemblies. Action must continue until all insertable control rods in core cells containing one or more fuel assemblies have been fully inserted. Control rods in core cells containing no fuel assemblies do not affect the reactivity of the core and therefore do not have to be inserted.

PA1

Action must also be initiated within 1 hour to provide means for control of potential radioactive releases. This includes ensuring secondary containment is OPERABLE; at least one SGT subsystem is OPERABLE; and secondary containment isolation capability is available (i.e., at least one secondary containment isolation valve and associated instrumentation are OPERABLE, or other acceptable administrative controls to assure isolation capability in each associated secondary containment penetration flow path not isolated that is PA2 PA1

PA1 move to next page (A)

(continued)

BASES

PA1

PA1

Insert A from previous page

ACTIONS

E.1, E.2, E.3, E.4, and E.5 (continued)

These administrative controls consist of stationing a dedicated operator, who is in continuous communication with the control room, at the controls of the isolation device. In this way, the penetration can be rapidly isolated when a need for secondary containment isolation is indicated.

assumed to be isolated to mitigate radioactivity releases. This may be performed as an administrative check, by examining logs or other information, to determine if the components are out of service for maintenance or other reasons. It is not necessary to perform the Surveillances as needed to demonstrate the OPERABILITY of the components. If, however, any required component is inoperable, then it must be restored to OPERABLE status. In this case, SRs may need to be performed to restore the component to OPERABLE status. Action must continue until all required components are OPERABLE.

SURVEILLANCE REQUIREMENTS

SR 3.1.1.1

PA1

Verified

PA1

Adequate SDM must be ~~demonstrated~~ to ensure that the reactor can be made subcritical from any initial operating condition. Adequate SDM is demonstrated by testing before or during the first startup after fuel movement, control rod replacement, or shuffling within the reactor pressure vessel. Control rod replacement refers to the decoupling and removal of a control rod from a core location, and subsequent replacement with a new control rod or a control rod from another core location. Since core reactivity will vary during the cycle as a function of fuel depletion and poison burnup, the beginning of cycle (BOC) test must also account for changes in core reactivity during the cycle. Therefore, to obtain the SDM, the initial measured value must be increased by an adder, "R", which is the difference between the calculated value of maximum core reactivity during the operating cycle and the calculated BOC core reactivity. If the value of R is negative (that is, BOC is the most reactive point in the cycle), no correction to the BOC measured value is required (Ref. 8). For the SDM demonstrations that rely solely on calculation of the highest worth control rod, additional margin (0.10% Δk/k) must be added to the SDM limit of 0.28% Δk/k to account for uncertainties in the calculation.

PA1

This can be accomplished by a test (by withdrawing control rods), an evaluation, or a combination of the two.

with the highest reactivity worth control rod fully withdrawn and all other control rods fully inserted

CLB2

CLB1

X1

8

In both cases,

The SDM may be demonstrated during an in-sequence control rod withdrawal, in which the highest worth control rod is analytically determined, or during local criticals, where the highest worth control rod is determined by testing.

PA3

CLB1

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.1.1.1 (continued)

Local critical tests require the withdrawal of out of sequence control rods. This testing would therefore require bypassing of the rod worth minimizer to allow the out of sequence withdrawal, and therefore additional requirements must be met (see LCO 3.10.7, "Control Rod Testing—Operating") and LCO 3.10.8, "SHUTDOWN MARGIN Test — Refueling". PA1

The Frequency of 4 hours after reaching criticality is allowed to provide a reasonable amount of time to perform the required calculations and have appropriate verification.

PA1

During MODES 3 and 4, analytical calculation of SDM may be used to assure the requirements of SR 3.1.1.1 are met.

During MODE 5, adequate SDM is required to ensure that the reactor does not reach criticality during control rod withdrawals. An evaluation of each in-vessel fuel movement during fuel loading (including shuffling fuel within the core) is required to ensure adequate SDM is maintained during refueling. This evaluation ensures that the intermediate loading patterns are bounded by the safety analyses for the final core loading pattern. For example, bounding analyses that demonstrate adequate SDM for the most reactive configurations during the refueling may be performed to demonstrate acceptability of the entire fuel movement sequence. These bounding analyses include additional margins to the associated uncertainties. Spiral offload/reload sequences inherently satisfy the SR, provided the fuel assemblies are reloaded in the same configuration analyzed for the new cycle. Removing fuel from the core will always result in an increase in SDM.

REFERENCES

1. UFSAR, Section 16.6 ~~10 CFR 50, Appendix A, GDC 26~~. DB1
2. UFSAR, Section ~~[15.1.38]~~. 14.6.1.2 DB2
3. NEDE-24011-P-A-~~8~~-US, "General Electric Standard Application for Reactor Fuel," Supplement for United States, Section 2.2.3.1, ~~September 1988~~ August 1996 DB3
4. UFSAR, Section ~~[15.1.13]~~. 14.5.4.3 DB2
5. UFSAR, Section ~~[15.1.14]~~. 14.5.4.4 DB2
6. 10 CFR 50.36(c)(2)(ii) DA

PA4 PA1 7 INSERT NEXT PAGE

(continued)

INSERT FROM PAGE B 3.1-6 PA1

RA4

BASES

REFERENCES (continued)

6. UFSAR, Section [4.3.2.4.1]. 13.7.2.4 DB2

7. NEDE-24011-P-A-13, "General Electric Standard Application for Reactor Fuel," Section 3.2.4.1, ~~September 1988~~. DB3 PA1

August 1996 DB3

JAFNPP

IMPROVED STANDARD TECHNICAL SPECIFICATIONS (ISTS) CONVERSION

ITS: 3.1.1

SHUTDOWN MARGIN (SDM)

JUSTIFICATION FOR DIFFERENCES (JFDs) FROM NUREG-1433, REVISION 1, BASES

JUSTIFICATION FOR DIFFERENCES FROM NUREG-1433, REVISION 1
ITS BASES: 3.1.1 - SHUTDOWN MARGIN (SDM)

RETENTION OF EXISTING REQUIREMENT (CLB)

CLB1 The Bases have been revised to reflect the use of only one limit consistent with the Specification and current Licensing requirement in CTS 3.3.A and 4.3.A.

CLB2 The words (with the highest reactivity worth control rod fully withdrawn and all other control rods fully inserted) have been added consistent with the current requirements in CTS 3.3.A.1 and CTS 4.3.A.1.

PLANT-SPECIFIC WORDING PREFERENCE OR MINOR EDITORIAL IMPROVEMENT (PA)

PA1 Editorial changes made for enhance clarity or to be consistent with similar statements in the Specifications and/or Bases.

PA2 The brackets have been removed and the proper plant specific information included.

PA3 Typographical/grammatical error corrected.

PA4 Changes have been made (additions, deletions, and/or changes to the NUREG) to reflect the plant specific nomenclature.

PLANT-SPECIFIC DIFFERENCE IN THE DESIGN (DB)

DB1 JAFNPP was designed and under construction prior to the promulgation of Appendix A to 10 CFR 50 - General Design Criteria for Nuclear Power Plants. The JAFNPP Construction Permit was issued on May 20, 1970. The proposed General Design Criteria (GDC) were published in the Federal Register on July 11, 1967 (32 FR 10213) and became effective on February 20, 1971 (32 FR 3256). UFSAR, Section 16.6 - Conformance to AEC Design Criteria, describes the JAFNPP current licensing basis with regard to the GDC. ISTS statements concerning the GDC are modified in the ITS to reference UFSAR, Section 16.6.

DB2 The brackets have been removed from the References and the appropriate plant specific references included.

DB3 Changes have been made (additions, deletions and/or changes to the NUREG) to reflect the plant specific references.

DIFFERENCE BASED ON AN APPROVED TRAVELER (TA)

None

JUSTIFICATION FOR DIFFERENCES FROM NUREG-1433, REVISION 1
ITS BASES: 3.1.1 - SHUTDOWN MARGIN (SDM)

DIFFERENCE BASED ON A SUBMITTED, BUT PENDING TRAVELER (TP)

None

DIFFERENCE FOR ANY REASON OTHER THAN THE ABOVE (X)

X1 NUREG-1433, Revision 1, Bases reference to "the NRC Policy Statement" has been replaced with 10 CFR 50.36(c)(2)(ii), in accordance with 60 FR 36953 effective August 18, 1995. Subsequent References have been renumbered, as required.

JAFNPP

IMPROVED STANDARD TECHNICAL SPECIFICATIONS (ISTS) CONVERSION

ITS: 3.1.1

SHUTDOWN MARGIN (SDM)

RETYPED PROPOSED IMPROVED TECHNICAL SPECIFICATIONS (ITS) AND BASES

3.1 REACTIVITY CONTROL SYSTEMS

3.1.1 SHUTDOWN MARGIN (SDM)

LCO 3.1.1 SDM shall be ≥ 0.38% Δk/k, with the highest worth control rod analytically determined.

APPLICABILITY: MODES 1, 2, 3, 4, and 5.

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. SDM not within limits in MODE 1 or 2.	A.1	Restore SDM to within limits.	6 hours
B. Required Action and associated Completion Time of Condition A not met.	B.1	Be in MODE 3.	12 hours
C. SDM not within limits in MODE 3.	C.1	Initiate action to fully insert all insertable control rods.	Immediately
D. SDM not within limits in MODE 4.	D.1	Initiate action to fully insert all insertable control rods.	Immediately
	AND		(continued)

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
D. (continued)	D.2	Initiate action to restore secondary containment to OPERABLE status.	1 hour
	AND		
	D.3	Initiate action to restore one standby gas treatment (SGT) subsystem to OPERABLE status.	1 hour
	AND		
	D.4	Initiate action to restore isolation capability in each required secondary containment penetration flow path not isolated.	1 hour
E. SDM not within limits in MODE 5.	E.1	Suspend CORE ALTERATIONS except for control rod insertion and fuel assembly removal.	Immediately
	AND		
	E.2	Initiate action to fully insert all insertable control rods in core cells containing one or more fuel assemblies.	Immediately
	AND		
			(continued)

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
E. (continued)	E.3	Initiate action to restore secondary containment to OPERABLE status.	1 hour
	AND		
	E.4	Initiate action to restore one SGT subsystem to OPERABLE status.	1 hour
	AND		
	E.5	Initiate action to restore isolation capability in each required secondary containment penetration flow path not isolated.	1 hour

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.1.1.1	Verify SDM is ≥ 0.38% Δk/k with the highest worth control rod analytically determined.	Prior to each in vessel fuel movement during fuel loading sequence AND Once within 4 hours after criticality following fuel movement within the reactor pressure vessel or control rod replacement

B 3.1 REACTIVITY CONTROL SYSTEMS

B 3.1.1 SHUTDOWN MARGIN (SDM)

BASES

BACKGROUND

SDM requirements are specified to ensure:

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

b. The reactivity transients associated with postulated accident conditions are controllable within acceptable limits; and

c. The reactor will be maintained sufficiently subcritical to preclude inadvertent criticality in the shutdown condition.

These requirements are satisfied by the control rods, as described in the Updated Final Safety Analysis Report (UFSAR) Section 16.6 (Ref. 1), which can compensate for the reactivity effects of the fuel and water temperature changes experienced during all operating conditions.

APPLICABLE SAFETY ANALYSES

The control rod drop accident (CRDA) analysis (Refs. 2 and 3) assumes the core is subcritical with the highest worth control rod withdrawn. Typically, the first control rod withdrawn has a very high reactivity worth and, should the core SDM be substantially less than 0.38% $\Delta k/k$ during the withdrawal of the first control rod, the consequences of a CRDA could exceed the fuel damage limits for a CRDA (see Bases for LCO 3.1.6, "Rod Pattern Control"). Also, SDM is assumed as an initial condition for the control rod removal error during refueling (Ref. 4) and fuel assembly insertion error during refueling (Ref. 5) accidents. The analysis of these reactivity insertion events assumes the refueling interlocks are OPERABLE when the reactor is in the refueling mode of operation. These interlocks prevent the withdrawal of more than one control rod from the core during refueling. (Special consideration and requirements for multiple control rod withdrawal during refueling are covered in Special Operations LCO 3.10.6, "Multiple Control Rod Withdrawal–Refueling.") The analysis assumes this condition is acceptable since the core will be shut down

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

with the highest worth control rod withdrawn, if adequate SDM has been demonstrated.

Prevention or mitigation of reactivity insertion events is necessary to limit energy deposition in the fuel to prevent significant fuel damage, which could result in undue release of radioactivity. Adequate SDM ensures inadvertent criticalities and potential CRDAs involving high worth control rods (namely the first control rod withdrawn) will not cause significant fuel damage.

SDM satisfies Criterion 2 of 10 CFR 50.36(c)(2)(ii) (Ref. 5).

LCO

The specified SDM limit accounts for the uncertainty in the demonstration of the SDM by analysis or by a combination of test and analysis. A SDM limit is provided where the highest worth control rod is determined analytically. SDM is demonstrated by analysis or by a combination of test and analysis. During refueling it is demonstrated by analysis and during a startup it is demonstrated by a combination of test and analysis. To ensure adequate SDM during the design process, a design margin is included to account for uncertainties in the design calculations (Ref. 6).

APPLICABILITY

In MODES 1 and 2, SDM must be provided because subcriticality with the highest worth control rod withdrawn is assumed in the CRDA analysis (Ref. 2). In MODES 3 and 4, SDM is required to ensure the reactor will be held subcritical with margin for a single withdrawn control rod. SDM is required in MODE 5 to prevent an open vessel, inadvertent criticality during the withdrawal of a single control rod from a core cell containing one or more fuel assemblies or a fuel assembly insertion error (Ref. 7).

(continued)

BASES (continued)

ACTIONS

A.1

With SDM not within the limits of the LCO in MODE 1 or 2, SDM must be restored within 6 hours. Failure to meet the specified SDM may be caused by a control rod that cannot be inserted. The allowed Completion Time of 6 hours is acceptable, considering that the reactor can still be shut down, assuming no failures of additional control rods to insert, and the low probability of an event occurring during this interval.

B.1

If the SDM cannot be restored, the plant must be brought to MODE 3 in 12 hours, to prevent the potential for further reductions in available SDM (e.g., additional stuck control rods). The allowed Completion Time of 12 hours is reasonable, based on operating experience, to reach MODE 3 from full power conditions in an orderly manner and without challenging plant systems.

C.1

With SDM not within limits in MODE 3, the operator must immediately initiate action to fully insert all insertable control rods. Action must continue until all insertable control rods are fully inserted. This action results in the least reactive condition for the core.

D.1, D.2, D.3, and D.4

With SDM not within limits in MODE 4, the operator must immediately initiate action to fully insert all insertable control rods. Action must continue until all insertable control rods are fully inserted. This action results in the least reactive condition for the core. Action must also be initiated within 1 hour to provide means for control of potential radioactive releases. This includes ensuring secondary containment is OPERABLE; at least one Standby Gas Treatment (SGT) subsystem is OPERABLE; and secondary containment isolation capability is available in each associated secondary containment penetration flow path not isolated that is assumed to isolate to mitigate

(continued)

BASES

ACTIONS

D.1, D.2, D.3, and D.4 (continued)

radioactivity releases (i.e., at least one secondary containment isolation valve and associated instrumentation are OPERABLE, or acceptable administrative controls assure isolation capability. These administrative controls consist of stationing a dedicated operator, who is in continuous communication with the control room, at the controls of the isolation device. In this way, the penetration can be rapidly isolated when a need for secondary containment isolation is indicated). This may be performed as an administrative check, by examining logs or other information, to determine if the components are out of service for maintenance or other reasons. It is not necessary to perform the surveillances needed to demonstrate the OPERABILITY of the components. If, however, any required component is inoperable, then it must be restored to OPERABLE status. In this case, SRs may need to be performed to restore the component to OPERABLE status. Actions must continue until all required components are OPERABLE.

E.1, E.2, E.3, E.4, and E.5

With SDM not within limits in MODE 5, the operator must immediately suspend CORE ALTERATIONS that could reduce SDM (e.g., insertion of fuel in the core or the withdrawal of control rods). Suspension of these activities shall not preclude completion of movement of a component to a safe condition. Inserting control rods or removing fuel from the core will reduce the total reactivity and are therefore excluded from the suspended actions.

Action must also be immediately initiated to fully insert all insertable control rods in core cells containing one or more fuel assemblies. Action must continue until all insertable control rods in core cells containing one or more fuel assemblies have been fully inserted. Control rods in core cells containing no fuel assemblies do not affect the reactivity of the core and therefore do not have to be inserted.

Action must also be initiated within 1 hour to provide means for control of potential radioactive releases. This includes ensuring secondary containment is OPERABLE; at

(continued)

BASES

ACTIONS

E.1, E.2, E.3, E.4, and E.5 (continued)

least one SGT subsystem is OPERABLE; and secondary containment isolation capability is available in each associated secondary containment penetration flow path not isolated that is assumed to isolate to mitigate radioactivity releases (i.e., at least one secondary containment isolation valve and associated instrumentation are OPERABLE, or acceptable administrative controls assure isolation capability. These administrative controls consist of stationing a dedicated operator, who is in continuous communication with the control room, at the controls of the isolation device. In this way, the penetration can be rapidly isolated when a need for secondary containment isolation is indicated). This may be performed as an administrative check, by examining logs or other information, to determine if the components are out of service for maintenance or other reasons. It is not necessary to perform the Surveillances as needed to demonstrate the OPERABILITY of the components. If, however, any required component is inoperable, then it must be restored to OPERABLE status. In this case, SRs may need to be performed to restore the component to OPERABLE status. Action must continue until all required components are OPERABLE.

SURVEILLANCE REQUIREMENTS

SR 3.1.1.1

Adequate SDM must be verified to ensure that the reactor can be made subcritical from any initial operating condition with the highest reactivity worth control rod fully withdrawn and all other control rods fully inserted. This can be accomplished by a test (by withdrawing control rods), an evaluation, or a combination of the two. Adequate SDM is demonstrated by testing before or during the first startup after fuel movement, or control rod replacement. Control rod replacement refers to the decoupling and removal of a control rod from a core location, and subsequent replacement with a new control rod or a control rod from another core location. Since core reactivity will vary during the cycle as a function of fuel depletion and poison burnup, the beginning of cycle (BOC) test must also account for changes in core reactivity during the cycle. Therefore, to obtain the SDM, the initial measured value must be increased by an

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.1.1.1 (continued)

adder, "R", which is the difference between the calculated value of maximum core reactivity during the operating cycle and the calculated BOC core reactivity. If the value of R is negative (that is, BOC is the most reactive point in the cycle), no correction to the BOC measured value is required (Ref. 8).

The SDM may be demonstrated during an in-sequence control rod withdrawal or during local criticals. In both cases, the highest worth control rod is analytically determined. Local critical tests require the withdrawal of out of sequence control rods. This testing would therefore require bypassing of the rod worth minimizer to allow the out of sequence withdrawal, and therefore additional requirements must be met (see LCO 3.10.7, "Control Rod Testing—Operating" and LCO 3.10.8, "SHUTDOWN MARGIN Test—Refueling").

The Frequency of 4 hours after reaching criticality is allowed to provide a reasonable amount of time to perform the required calculations and have appropriate verification.

During MODES 3 or 4, analytical calculation of SDM may be used to assure the requirements of SR 3.1.1.1 are met. During MODE 5, adequate SDM is required to ensure that the reactor does not reach criticality during control rod withdrawals. An evaluation of each in-vessel fuel movement during fuel loading (including shuffling fuel within the core) is required to ensure adequate SDM is maintained during refueling. This evaluation ensures that the intermediate loading patterns are bounded by the safety analyses for the final core loading pattern. For example, bounding analyses that demonstrate adequate SDM for the most reactive configurations during the refueling may be performed to demonstrate acceptability of the entire fuel movement sequence. These bounding analyses include additional margins to the associated uncertainties. Spiral offload/reload sequences inherently satisfy the SR, provided the fuel assemblies are reloaded in the same configuration analyzed for the new cycle. Removing fuel from the core will always result in an increase in SDM.

(continued)

BASES (continued)

REFERENCES

1. UFSAR, Section 16.6.

2. UFSAR, Section 14.6.1.2.

3. NEDE-24011-P-A-13-US, General Electric Standard Application for Reactor Fuel, Supplement for United States, Section 2.2.3.1, August 1996.

4. UFSAR, Section 14.5.4.3.

5. 10 CFR 50.36(c)(2)(ii).

6. UFSAR, Section 13.7.2.4.

7. UFSAR, Section 14.5.4.4.

8. NEDE-24011-P-A-13, General Electric Standard Application for Reactor Fuel, Section 3.2.4.1, August 1996.

JAFNPP

IMPROVED STANDARD TECHNICAL SPECIFICATIONS (ISTS) CONVERSION

ITS: 3.1.2

Reactivity Anomalies

MARKUP OF CURRENT TECHNICAL SPECIFICATIONS (CTS)

DISCUSSION OF CHANGES (DOCs) TO THE CTS

NO SIGNIFICANT HAZARDS CONSIDERATION (NSHC) FOR LESS RESTRICTIVE CHANGES

MARKUP OF NUREG-1433, REVISION 1, SPECIFICATION

JUSTIFICATION FOR DIFFERENCES (JFDs) FROM NUREG-1433, REVISION 1
MARKUP OF NUREG-1433, REVISION 1, BASES

JUSTIFICATION FOR DIFFERENCES (JFDs) FROM NUREG-1433, REVISION 1, BASES

RETYPED PROPOSED IMPROVED TECHNICAL SPECIFICATIONS (ITS) AND BASES

JAFNPP

IMPROVED STANDARD TECHNICAL SPECIFICATIONS (ISTS) CONVERSION

ITS: 3.1.2

Reactivity Anomalies

MARKUP OF CURRENT TECHNICAL SPECIFICATIONS (CTS)

Specification 3.1.2

[LCO 3.1.2]

[3.1.2]

3.3 (Cont'd)

D. Reactivity Anomalies

M1

MODES 1 and 2

The reactivity equivalent of the difference between the actual critical rod configuration and the expected configuration during power operation shall not exceed 1 percent Δ k. If this limit is [SR 3.1.2.1] exceeded, the reactor will be shut down until the cause has been determined and corrective actions have been taken as appropriate.

[ACTION A]

L1

restore core activity difference to within the limits within 72 hours

A2

See ITS 3.1.4

[LCO 3.0.4] E

[ACTION B]

If Specifications 3.3.C and D above cannot be met, an orderly shutdown shall be initiated and the reactor shall be in the cold condition within 24 hr.

L2

M2

MODE 3 within 12 hours

[3.1.2]

4.3 (Cont'd)

E. Reactivity Anomalies

A3

During the Startup test program and startup following refueling outages, the critical rod configurations will be compared to the expected configurations at selected operating conditions. These comparisons will be used as base data for reactivity monitoring during subsequent power operation throughout the fuel cycle. At specific power operating conditions, the critical rod configuration will be compared to the configuration expected based upon appropriately corrected past data. This comparison will be made at least every full power month.

LA1

A1

Once within 24 hours after reaching equilibrium conditions following

M3

or control rod replacement

M3

A4

RAI 3.1-01

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IMPROVED STANDARD TECHNICAL SPECIFICATIONS (ISTS) CONVERSION

ITS: 3.1.2

Reactivity Anomalies

DISCUSSION OF CHANGES (DOCs) TO THE CTS

DISCUSSION OF CHANGES
ITS: 3.1.2 - REACTIVITY ANOMALIES

ADMINISTRATIVE CHANGES

A1 In the conversion of the James A. FitzPatrick Nuclear Power Plant (JAFNPP) Current Technical Specification (CTS) to the proposed plant specific Improved Technical Specifications (ITS) certain wording preferences or conventions are adopted which do not result in technical changes. Editorial changes, reformatting, and revised numbering are adopted to make the ITS consistent with the conventions in NUREG-1433, "Standard Technical Specifications, General Electric Plants, BWR/4", Revision 1 (i.e., Improved Standard Technical Specifications (ISTS)).

A2 CTS 3.3.D in part requires that if Reactivity Anomalies exceed the limit, the reactor will be shutdown "until the cause has been determined, and corrective actions have been taken as appropriate." The proposed deletion of these words in the ITS will not change this requirement. ITS LCO 3.1.2 Conditions A and B require that the plant be shutdown to MODE 3 within 84 hours of finding Core Reactivity differences not within limits. In the proposed ITS presentation the ability to change MODES is generically controlled by the provisions of LCO 3.0.4 which states in part that "when an LCO is not met, entry into a MODE or other specified condition in the Applicability shall not be made except when the associated ACTIONS to be entered permit continued operation in the MODE or other specified condition in the Applicability for an unlimited period of time." Therefore LCO 3.0.4 would prevent plant startup with Core Reactivity outside of limits. Therefore, this proposed change causes no technical or actual change from present specifications. Therefore, the change is considered administrative, and is consistent with NUREG-1433, Revision 1.

A3 The Frequency for the Reactivity Anomalies Surveillance of "During the Startup test program" has been deleted from CTS 4.3.D since this test program has already occurred and will not be repeated again. As such this change is considered an administrative change consistent with NUREG-1433, Revision 1.

RAI 3.1-01

A4 CTS 4.3.D is revised to replace the term "reactivity monitoring" with "reactivity measuring." Core reactivity is a calculated value and is not displayed as a continuous readout, which is analogous to a "monitored" value. Rather core reactivity is "measured" by considering actual control rod densities and performing appropriate calculations. This change does not affect the method utilized to verify this SR. As such, the change is considered administrative.

In addition, ITS SR 3.1.2.1 allows for the use of a plant-specific term since brackets are provided in this SR. The use of the word "measured" is consistent with the plant specific terminology used.

TECHNICAL CHANGES - MORE RESTRICTIVE

M1 CTS 3.3.D Applicability has been expanded from during "power operation" to "MODES 1 and 2." The CTS 1.0.0, definition of reactor power operation, includes the requirement that the reactor be critical and above 1 percent rated thermal power, thus excluding MODE 2 operations at less than 1 percent. The ITS Table 1.1-1 definitions (see Discussion of Changes for ITS Chapter 1.0) of MODES 1 and 2 do not rely on a power level requirement and thus are more inclusive. The ITS 3.1.2 Applicability expansion of this requirement is consistent with NUREG-1433, Revision 1, and is necessary to achieve consistency with safety analysis assumptions. This change imposes additional requirements on plant operations and, therefore, is more restrictive. This change is considered to have no adverse impact on safety.

M2 CTS 3.3.E requirement that the plant be placed in cold shutdown within 24 hours if the Reactivity Anomaly requirements are not met, is being deleted (L2). ITS 3.1.2 Required Action B.1 is added to require the plant to be in MODE 3 within 12 hours if the Required Action and associated Completion Time of Condition A (L1) are not met. Since Reactivity Anomaly is a measure of the difference between the measured and predicted rod density, placing the plant in MODE 3 ensures that all insertable control rods are fully inserted thus placing the plant in a non-applicable condition. The 12 hour Completion Time is reasonable, based on operating experience, to reach MODE 3 from full power conditions in an orderly manner and without challenging plant systems. The addition of this requirement is consistent with NUREG-1433, Revision 1, and is necessary to ensure the reactor is placed in the least reactive state, in a timely manner, in the event of a reactivity anomaly. This change imposes additional time limitations on plant operations to reach the rods fully inserted condition (MODE 3) once a shutdown is initiated and, therefore, is more restrictive. This change is considered to have no adverse impact on safety.

M3 CTS 4.3.D requires a comparison of the critical rod configurations to the expected configuration during startup following refuel outages. ITS SR 3.1.2.1 requires a verification that the core reactivity difference between the measured rod density and the predicted rod density is within $\pm$ 1% $\Delta k/k$ once within 24 hours after reaching equilibrium conditions following startup after fuel movement within the reactor pressure vessel or control rod replacement (1st frequency). This change is more restrictive since the proposed surveillance is explicit on the Frequency (24 hours after reaching equilibrium conditions) and provides an additional condition for performing the surveillance (control rod replacement). The addition of this

EDIT

TECHNICAL CHANGES - MORE RESTRICTIVE

M3 (continued)

requirement is consistent with NUREG-1433, Revision 1, and is necessary to ensure that any core change that could affect reactivity is evaluated properly. This change is considered to have no adverse impact on safety.

TECHNICAL CHANGES - LESS RESTRICTIVE (GENERIC)

LA1 Details of the method to perform the Surveillance and the purposes for the Reactivity Anomalies Surveillance in CTS 4.3.D (that the comparison will be used as the base for future reactivity anomaly checks) are proposed to be relocated to the Bases. These details are not necessary to ensure the Reactivity Anomalies limit is maintained. The requirement of ITS 3.1.2 and SR 3.1.2.1 are adequate to ensure the limit is met. As such these relocated details are not required to be in the ITS to provide adequate protection of the public health and safety. Changes to the Bases will be controlled by the provisions of the Bases Control Program described in Chapter 5 of the Technical Specifications.

TECHNICAL CHANGES - LESS RESTRICTIVE (SPECIFIC)

L1 CTS 3.3.D does not provide an explicit restoration time when Reactivity Anomalies is not met. ITS 3.1.2 ACTION A provides a Completion Time of 72 hours for the core reactivity difference to be restored to within limits (normally required to perform an analysis to determine the reason for the reactivity difference). Typically, a reactivity anomaly would be indicative of incorrect analysis inputs or assumptions of fuel reactivity used in the analysis. A determination and explanation of the cause of the anomaly may involve an offsite fuel analysis department and the fuel vendor. Contacting and obtaining the necessary input may require a time period much longer than the 24 hours currently allowed by CTS 3.3.E to place the plant in a cold shutdown condition. Since SDM has typically been demonstrated by test prior to reaching the conditions at which this surveillance is performed, the safety impact of the extended time for evaluation is negligible. Given these considerations, the time is proposed to be extended to 72 hours. This is consistent with NUREG-1433, Revision 1.

TECHNICAL CHANGES - LESS RESTRICTIVE (SPECIFIC) (continued)

L2 The CTS 3.3.E requirement to be in a cold condition within 24 hours, if CTS 3.3.D cannot be met, is being deleted. This deletion is acceptable since ITS 3.1.2 ACTION B (M2) requirement to be in MODE 3 in 12 hours, if the Required Action and associated Completion Time are not met, has been added, which places the plant in a Condition outside the ITS 3.1.2 (CTS 3.3.D) Applicability. In MODE 3 all control rods are fully inserted and therefore the reactor is in the least reactive state, where measuring core reactivity is not necessary, a continuation to cold shutdown (MODE 4) will not reduce core reactivity and therefore also is not necessary. In addition, if the reactivity anomaly specification is not met and if ITS 3.1.1 SHUTDOWN MARGIN (SDM) cannot be met, entry into the appropriate ITS 3.1.1 ACTION is required. In MODE 3, primary and secondary containment OPERABILITY is required, therefore adequate protection exists if a reactivity anomaly were to occur. The requirements of ITS 3.1.2 provide adequate protection and therefore is considered acceptable. This change is consistent with NUREG-1433, Revision 1.

TECHNICAL CHANGES - RELOCATIONS

None

JAFNPP

IMPROVED STANDARD TECHNICAL SPECIFICATIONS (ISTS) CONVERSION

ITS: 3.1.2

Reactivity Anomalies

NO SIGNIFICANT HAZARDS CONSIDERATION (NSHC) FOR LESS RESTRICTIVE CHANGES

NO SIGNIFICANT HAZARDS CONSIDERATIONS
ITS: 3.1.2 - REACTIVITY ANOMALIES

TECHNICAL CHANGES - LESS RESTRICTIVE (SPECIFIC)

L1 CHANGE

New York Power Authority has evaluated the proposed Technical Specification change identified as "Technical Changes - Less Restrictive" and has determined that it does not involve a significant hazards consideration. This determination has been performed in accordance with the criteria set forth in 10 CFR 50.92. The bases for the determination that the proposed change does not involve a significant hazards consideration are discussed below.

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

   The proposed change would allow 72 hours to evaluate and determine the cause of any reactivity anomalies prior to requiring a plant shutdown. Such a reactivity anomaly is not considered an initiator of any accident previously evaluated and therefore would not affect their probability. Substantial margin exists in the analyses which predict core reactivity and in those which analyze the accidents. In addition, adequate shutdown margin is demonstrated by test during plant startup after in-vessel fuel movement or control rod replacement and is followed by a reactivity anomaly test within 24 hours of reaching equilibrium conditions at greater than 75% rated thermal power. Since the first reactivity anomaly test is typically performed within a few days following the shutdown margin demonstration the reactivity difference between the measured and predicted rod density is expected to be small. Based on experience, the reactivity differences determined by periodic performance of reactivity anomaly tests are also expected to be small, slow developing and insignificant with respect to the probability or consequences of accidents previously evaluated. Further, the consequences of an event occurring during the proposed 72 hour Completion Time are the same as the consequences of an event occurring under the current Actions. Therefore, this change will not involve a significant increase in the probability or consequences of an accident previously evaluated.

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

   The proposed change does not involve a physical alteration of the plant (no new or different type of equipment will be installed) or changes in methods governing normal plant operation. The proposed change will only provide a 72 hour Completion Time to restore the core reactivity difference to within limits before requiring a plant shutdown. Therefore, this change will not create the possibility of a new or different kind of accident from any accident previously evaluated.

TECHNICAL CHANGES - LESS RESTRICTIVE (SPECIFIC)

L1 CHANGE

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

The proposed 72 hour Completion Time to restore core reactivity difference to within limits prior to requiring a plant shutdown is acceptable based on the small probability of an event occurring during this time period. Further, reactivity anomaly conditions develop slowly so there will not be a substantial change in the anomaly during the longer allowed interval before plant shutdown. Any minor decrease in the margin of safety during the additional time is offset by minimization of the potential for plant transients which may occur while shutting down the plant. Therefore, this change does not involve a significant reduction in a margin of safety.

NO SIGNIFICANT HAZARDS CONSIDERATIONS
ITS: 3.1.2 - REACTIVITY ANOMALIES

TECHNICAL CHANGES - LESS RESTRICTIVE (SPECIFIC)

L2 CHANGE

New York Power Authority has evaluated the proposed Technical Specification change identified as "Technical Changes - Less Restrictive" and has determined that it does not involve a significant hazards consideration. This determination has been performed in accordance with the criteria set forth in 10 CFR 50.92. The bases for the determination that the proposed change does not involve a significant hazards consideration are discussed below.

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

   The proposed change deletes the requirement to be in a cold condition in 24 hours when the Reactivity Anomalies Specification of CTS 3.3.D is not met. Placing the plant in a cold condition does not place the plant in a less reactive condition. The reactor core is more reactive at colder temperatures, therefore the requirement to be in a cold condition does not decrease significance of the reactivity anomaly. The new requirement (M2) will be to be in MODE 3 in 12 hours (ITS 3.1.2 Required Action B.1). The proposed action is considered sufficient when Reactivity Anomalies is not met. The requirement to be in a cold condition within 24 hours if Reactivity Anomalies is not met is not considered in the initiation of any accident. Therefore, this change does not significantly increase the probability of any accident previously evaluated. The proposed ACTION exits the Applicability of the LCO and limits core reactivity. In Mode 3 the primary and secondary containment are required to be OPERABLE to limit the consequences of any design bases accident. Thus, the consequences of an accident will not be increased as a result of this change. Therefore, this change will not involve a significant increase in the probability or consequences of an accident previously evaluated.

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

   The proposed change will not involve a physical alteration of the plant (no new or different type of equipment will be installed) or changes in methods governing normal plant operation. The proposed change limits core reactivity and other Specifications will provide additional requirements to ensure sufficient components are OPERABLE to limit any radioactivity release if an event were to occur. Therefore, this change will not create the possibility of a new or different type of accident from any accident previously analyzed.

TECHNICAL CHANGES - LESS RESTRICTIVE (SPECIFIC)

L2 CHANGE

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

The proposed change deletes the requirement to be in a cold condition in 24 hours when the Reactivity Anomalies Specification of CTS 3.3.D is not met. Placing the plant in a cold condition does not place the plant in a less reactive condition. The reactor core is more reactive at colder temperatures, therefore the requirement to be in a cold condition does not decrease significance of the reactivity anomaly. The new requirement (M2) will be to be in MODE 3 in 12 hours (ITS 3.1.2 Required Action B.1). The proposed ACTIONs are considered sufficient when Reactivity Anomalies is not met. The proposed action limits core reactivity and exits the Applicability of the LCO. In Mode 3 the primary and secondary containment are required to be Operable to limit the consequences of any design bases accident. Thus, the consequences of an accident will not be increased as a result of this change. Deleting this requirement to be in a cold condition when Reactivity Anomalies is not met will effectively decrease the core reactivity. This change will not impact any safety analysis assumptions. As such, no question of safety is involved. Therefore, this change does not involve a significant reduction in a margin of safety.

JAFNPP

IMPROVED STANDARD TECHNICAL SPECIFICATIONS (ISTS) CONVERSION

ITS: 3.1.2

Reactivity Anomalies

MARKUP OF NUREG-1433, REVISION 1 SPECIFICATION

PA1

3.1 REACTIVITY CONTROL SYSTEMS

3.1.2 Reactivity Anomalies

[3.1.D] LCO 3.1.2 The reactivity [difference] between the [monitored] measured rod density and the predicted rod density[] shall be within ± 1% Δk/k.

[3.3.D] APPLICABILITY: MODES 1 and 2.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
[3.3.D] A. Core reactivity [difference] not within limit.	A.1 Restore core reactivity [difference] to within limit.	72 hours
[3.3.E] B. Required Action and associated Completion Time not met.	B.1 Be in MODE 3.	12 hours

PA1

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.1.2.1	Verify core reactivity difference between the ~~monitored~~ measured rod density and the predicted rod density is within ± 1% Δk/k.	Once within 24 hours after reaching equilibrium conditions following startup after fuel movement within the reactor pressure vessel or control rod replacement AND ~~1000 MWD/T~~ Each full power month thereafter during operations in MODE 1 CLB1

[4.3.D]
[M3]

JAFNPP

IMPROVED STANDARD TECHNICAL SPECIFICATIONS (ISTS) CONVERSION

ITS: 3.1.2

Reactivity Anomalies

JUSTIFICATION FOR DIFFERENCES (JFDs) FROM NUREG-1433, REVISION 1

JUSTIFICATION FOR DIFFERENCES FROM NUREG-1433, REVISION 1
ITS: 3.1.2 - REACTIVITY ANOMALIES

RETENTION OF EXISTING REQUIREMENT (CLB)

CLB1 ITS SR 3.1.2.1 Frequency of 1000 MWD/T has been revised to reflect the current licensing requirements of JAFNPP, CTS 4.3.D of every full power month.

PLANT-SPECIFIC WORDING PREFERENCE OR MINOR EDITORIAL IMPROVEMENT (PA)

PA1 Changes have been made (additions, deletions and/or changes to the NUREG) to reflect the plant specific system/structure/component nomenclature, equipment identification or description.

PLANT-SPECIFIC DIFFERENCE IN THE DESIGN (DB)

None

DIFFERENCE BASED ON AN APPROVED TRAVELER (TA)

None

DIFFERENCE BASED ON A SUBMITTED, BUT PENDING TRAVELER (TP)

None

DIFFERENCE FOR ANY REASON OTHER THAN THE ABOVE (X)

None

JAFNPP

IMPROVED STANDARD TECHNICAL SPECIFICATIONS (ISTS) CONVERSION

ITS: 3.1.2

Reactivity Anomalies

MARKUP OF NUREG-1433, REVISION 1, BASES

PA1

B 3.1 REACTIVITY CONTROL SYSTEMS

B 3.1.2 Reactivity Anomalies

unless otherwise noted

DB1 the Updated Final Safety Evaluation Report (UFSAR) Section 16.6

BASES

BACKGROUND

In accordance with GDC 26, GDC 28, and GDC 29 (Ref. 1), reactivity shall be controllable such that subcriticality is maintained under cold conditions and acceptable fuel design limits are not exceeded during normal operation and anticipated operational occurrences. Therefore, reactivity anomaly is used as a measure of the predicted versus measured core reactivity during power operation. The continual confirmation of core reactivity is necessary to ensure that the Design Basis Accident (DBA) and transient safety analyses remain valid. A large reactivity anomaly could be the result of unanticipated changes in fuel reactivity or control rod worth or operation at conditions not consistent with those assumed in the predictions of core reactivity, and could potentially result in a loss of SDM or violation of acceptable fuel design limits. Comparing predicted versus measured core reactivity validates the nuclear methods used in the safety analysis and supports the SDM demonstrations (LCO 3.1.1, "SHUTDOWN MARGIN (SDM)") in assuring the reactor can be brought safely to cold, subcritical conditions.

PA2 transients abnormal Anomalies are PA2 requirements

R

When the reactor core is critical or in normal power operation, a reactivity balance exists and the net reactivity is zero. A comparison of predicted and measured reactivity is convenient under such a balance, since parameters are being maintained relatively stable under steady state power conditions. 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 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 loaded in the previous cycles provide excess positive reactivity beyond that required to sustain steady state operation at the beginning of cycle (BOC). When the reactor is critical at RTP and operating moderator temperature, the excess positive reactivity is compensated by burnable absorbers (if any), control rods, and whatever neutron

(continued)

PA2 the Monitoring System

BASES

BACKGROUND (continued)

poisons (mainly xenon and samarium) are present in the fuel. The predicted core reactivity, as represented by control rod density, is calculated by a 3D core simulator code as a function of cycle exposure. This calculation is performed for projected operating states and conditions throughout the cycle. The core reactivity is determined from control rod densities for actual plant conditions and is then compared to the predicted value for the cycle exposure.

APPLICABLE SAFETY ANALYSES

Accurate prediction of core reactivity is either an explicit or implicit assumption in the accident analysis evaluations (Ref. 2). In particular, SDM and reactivity transients, such as control rod withdrawal accidents or rod drop accidents, are very sensitive to accurate prediction of core reactivity. These accident analysis evaluations rely on computer codes that have been qualified against available test data, operating plant data, and analytical benchmarks. Measuring PA1 Monitoring reactivity anomaly provides additional assurance that the nuclear methods provide an accurate representation of the core reactivity.

The comparison between measured and predicted initial core reactivity provides a normalization for the calculational models used to predict core reactivity. If the measured and predicted rod density for identical core conditions at BOC do not reasonably agree, then the assumptions used in the reload cycle design analysis or the calculation models used to predict rod density may not be accurate. If reasonable agreement between measured and predicted core reactivity exists at BOC, then the prediction may be normalized to the measured value. Thereafter, any significant deviations in the measured rod density from the predicted rod density that develop during fuel depletion may be an indication that the assumptions of the DBA and transient analyses are no longer valid, or that an unexpected change in core conditions has occurred.

PA1

Reactivity anomalies satisfy Criterion 2 of the NRC Policy Statement.

10 CFR 50.36 (c)(2)(ii) (Ref. 3) X1

(continued)

PA1 unless otherwise noted

BASES (continued)

LCO

The reactivity anomaly limit is established to ensure plant operation is maintained within the assumptions of the safety analyses. Large differences between ~~monitored~~ measured and predicted core reactivity may indicate that the assumptions of the DBA and transient analyses are no longer valid, or that the uncertainties in the "Nuclear Design Methodology" are larger than expected. A limit on the difference between the ~~monitored~~ measured and the predicted rod density of ± 1% Δk/k has been established based on engineering judgment. A > 1% deviation in reactivity from that predicted is larger than expected for normal operation and should therefore be evaluated.

APPLICABILITY

In MODE 1, most of the control rods are withdrawn and steady state operation is typically achieved. Under these conditions, the comparison between predicted and ~~monitored~~ measured core reactivity provides an effective measure of the reactivity anomaly. In MODE 2, control rods are typically being withdrawn during a startup. In MODES 3 and 4, all control rods are fully inserted and therefore the reactor is in the least reactive state, where ~~monitoring~~ measuring core reactivity is not necessary. In MODE 5, fuel loading results in a continually changing core reactivity. SDM requirements (LCO 3.1.1) ensure that fuel movements are performed within the bounds of the safety analysis, and an SDM demonstration is required during the first startup following operations that could have altered core reactivity (e.g., fuel movement, control rod replacement, shuffling). The SDM test, required by LCO 3.1.1, provides a direct comparison of the predicted and ~~monitored~~ measured core reactivity at cold conditions; therefore, the reactivity ~~anomaly~~ Anomalies Specification is not required during these the conditions.

ACTIONS

A.1

Should an anomaly develop between measured and predicted core reactivity, the core reactivity difference must be restored to within the limit to ensure continued operation is within the core design assumptions. Restoration to within the limit could be performed by an evaluation of the core design and safety analysis to determine the reason for the anomaly. This evaluation normally reviews the core

(continued)

PA1 unless otherwise noted

BASES

ACTIONS

A.1 (continued)

conditions to determine their consistency with input to design calculations. Measured core and process parameters are also normally evaluated to determine that they are within the bounds of the safety analysis, and safety analysis calculational models may be reviewed to verify that they are adequate for representation of the core conditions. The required Completion Time of 72 hours is based on the low probability of a DBA occurring during this period, and allows sufficient time to assess the physical condition of the reactor and complete the evaluation of the core design and safety analysis.

B.1

If the core reactivity cannot be restored to within the 1% $\Delta k/k$ limit, the plant must be brought to a MODE in which the LCO does not apply. To achieve this status, the plant must be brought to at least MODE 3 within 12 hours. The allowed Completion Time of 12 hours is reasonable, based on operating experience, to reach MODE 3 from full power conditions in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.1.2.1

measured

PA2

3D Monicore

measured

Verifying the reactivity difference between the monitored and predicted rod density is within the limits of the LCO provides added assurance that plant operation is maintained within the assumptions of the DBA and transient analyses. The Core Monitoring System calculates the rod density for the reactor conditions obtained from plant instrumentation. A comparison of the monitored rod density to the predicted rod density at the same cycle exposure is used to calculate the reactivity difference. The comparison is required when the core reactivity has potentially changed by a significant amount. This may occur following a refueling in which new fuel assemblies are loaded, fuel assemblies are shuffled within the core, or control rods are replaced or shuffled. Control rod replacement refers to the decoupling and removal of a control rod from a core location, and subsequent replacement with a new control rod or a control

(continued)

PA1

BASES

SURVEILLANCE REQUIREMENTS

SR 3.1.2.1 (continued)

measured

rod from another core location. Also, core reactivity changes during the cycle. The 24 hour interval after reaching equilibrium conditions following a startup is based on the need for equilibrium xenon concentrations in the core, such that an accurate comparison between the monitored and predicted rod density can be made. For the purposes of this SR, the reactor is assumed to be at equilibrium conditions when steady state operations (no control rod movement or core flow changes) at ≥ 75% RTP have been obtained. The 1000 MWD/T Frequency was developed, considering the relatively slow change in core reactivity with exposure and operating experience related to variations in core reactivity. This comparison requires the core to be operating at power levels which minimize the uncertainties and measurement errors, in order to obtain meaningful results. Therefore, the comparison is only done when in MODE 1.

CLB1

every full power month

DB1

REFERENCES

PA2

1. 10 CFR 50, Appendix A, GDC 26.

UFSAR, Section 16.6

4

2. FSAR, Chapter 15.

14

DB2

3. 10 CFR 50.36 (c)(2)(ii)

X1

The tests performed at this Frequency also use base data obtained during the first test of the specific cycle.

JAFNPP

IMPROVED STANDARD TECHNICAL SPECIFICATIONS (ISTS) CONVERSION

ITS: 3.1.2

Reactivity Anomalies

JUSTIFICATION FOR DIFFERENCES (JFDs) FROM NUREG-1433, REVISION 1, BASES

JUSTIFICATION FOR DIFFERENCES FROM NUREG-1433, REVISION 1
ITS BASES: 3.1.2 - REACTIVITY ANOMALIES

RETENTION OF EXISTING REQUIREMENT (CLB)

CLB1 ITS SR 3.1.2.1 Frequency of 1000 MWD/T has been revised to reflect the current licensing requirements of JAFNPP, CTS 4.3.D of every full power month.

PLANT-SPECIFIC WORDING PREFERENCE OR MINOR EDITORIAL IMPROVEMENT (PA)

PA1 Editorial changes have been made for enhanced clarity or to correct a grammatical/typographical error.

PA2 Changes have been made (additions, deletions, and/or changes to the NUREG) to reflect the plant specific system/structure/component nomenclature, equipment identification or description.

PLANT-SPECIFIC DIFFERENCE IN THE DESIGN (DB)

DB1 JAFNPP was designed and under construction prior to the promulgation of Appendix A to 10 CFR 50 - General Design Criteria for Nuclear Power Plants. The JAFNPP Construction Permit was issued on May 20, 1970. The proposed General Design Criteria (GDC) were published in the Federal Register on July 11, 1967 (32 FR 10213) and became effective on February 20, 1971 (32 FR 3256). UFSAR, Section 16.6 - Conformance to AEC Design Criteria, describes the JAFNPP current licensing basis with regard to the GDC. ISTS statements concerning the GDC are modified in the ITS to reference UFSAR, Section 16.6. The brackets have been removed from the Reference and the proper plant specific reference included.

DB2 ITS 3.1.2 has been revised to reflect the specific JAFNPP reference requirements of, UFSAR, Chapter 14.

DIFFERENCE BASED ON AN APPROVED TRAVELER (TA)

None

DIFFERENCE BASED ON A SUBMITTED, BUT PENDING TRAVELER (TP)

None

DIFFERENCE FOR ANY REASON OTHER THAN THE ABOVE (X)

X1 NUREG-1433, Revision 1, Bases references to "the NRC Policy Statement" has been replaced with 10 CFR 50.36(c)(2)(ii), in accordance with 60 FR 36953 effective August 18, 1995.

JAFNPP

IMPROVED STANDARD TECHNICAL SPECIFICATIONS (ISTS) CONVERSION

ITS: 3.1.2

Reactivity Anomalies

RETYPED PROPOSED IMPROVED TECHNICAL SPECIFICATIONS (ITS) AND BASES

3.1 REACTIVITY CONTROL SYSTEMS

3.1.2 Reactivity Anomalies

LCO 3.1.2 The reactivity difference between the measured rod density and the predicted rod density shall be within ± 1% Δk/k.

APPLICABILITY: MODES 1 and 2.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Core reactivity difference not within limit.	A.1 Restore core reactivity difference to within limit.	72 hours
B. Required Action and associated Completion Time not met.	B.1 Be in MODE 3.	12 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.1.2.1	Verify core reactivity difference between the measured rod density and the predicted rod density is within ± 1% Δk/k.	Once within 24 hours after reaching equilibrium conditions following startup after fuel movement within the reactor pressure vessel or control rod replacement AND Every full power month thereafter during operations in MODE 1

B 3.1 REACTIVITY CONTROL SYSTEMS

B 3.1.2 Reactivity Anomalies

BASES

BACKGROUND

In accordance with the Updated Final Safety Evaluation Report (UFSAR) Section 16.6 (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 abnormal operational transients. Therefore, Reactivity Anomalies are used as a measure of the predicted versus measured core reactivity during power operation. The continual confirmation of core reactivity is necessary to ensure that the Design Basis Accident (DBA) and transient safety analyses remain valid. A large reactivity anomaly could be the result of unanticipated changes in fuel reactivity or control rod worth or operation at conditions not consistent with those assumed in the predictions of core reactivity, and could potentially result in a loss of SDM or violation of acceptable fuel design limits. Comparing predicted versus measured core reactivity validates the nuclear methods used in the safety analysis and supports the SDM requirements (LCO 3.1.1, "SHUTDOWN MARGIN (SDM)") in assuring the reactor can be brought safely to cold, subcritical conditions.

When the reactor core is critical or in normal power operation, a reactivity balance exists and the net reactivity is zero. A comparison of predicted and measured reactivity is convenient under such a balance, since parameters are being maintained relatively stable under steady state power conditions. 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 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 loaded in the previous cycles provide excess positive reactivity beyond that required to sustain steady state operation at the beginning of cycle (BOC). When the reactor is critical at RTP and operating moderator temperature, the excess positive reactivity is compensated by burnable absorbers (if any), control rods, and whatever neutron

(continued)

BASES

BACKGROUND (continued)

poisons (mainly xenon and samarium) are present in the fuel. The predicted core reactivity, as represented by control rod density, is calculated by the 3D Monicore System as a function of cycle exposure. This calculation is performed for projected operating states and conditions throughout the cycle. The core reactivity is determined from control rod densities for actual plant conditions and is then compared to the predicted value for the cycle exposure.

APPLICABLE SAFETY ANALYSES

Accurate prediction of core reactivity is either an explicit or implicit assumption in the accident analysis evaluations (Ref. 2). In particular, SDM and reactivity transients, such as control rod withdrawal accidents or rod drop accidents, are very sensitive to accurate prediction of core reactivity. These accident analysis evaluations rely on computer codes that have been qualified against available test data, operating plant data, and analytical benchmarks. Measuring reactivity anomaly provides additional assurance that the nuclear methods provide an accurate representation of the core reactivity.

The comparison between measured and predicted initial core reactivity provides a normalization for the calculational models used to predict core reactivity. If the measured and predicted rod density for identical core conditions at BOC do not reasonably agree, then the assumptions used in the reload cycle design analysis or the calculation models used to predict rod density may not be accurate. If reasonable agreement between measured and predicted core reactivity exists at BOC, then the prediction may be normalized to the measured value. Thereafter, any significant deviations in the measured rod density from the predicted rod density that develop during fuel depletion may be an indication that the assumptions of the DBA and transient analyses are no longer valid, or that an unexpected change in core conditions has occurred.

Reactivity Anomalies satisfy Criterion 2 of 10 CFR 50.36(c)(2)(ii) (Ref. 3).

(continued)

BASES (continued)

LCO

The reactivity anomaly limit is established to ensure plant operation is maintained within the assumptions of the safety analyses. Large differences between measured and predicted core reactivity may indicate that the assumptions of the DBA and transient analyses are no longer valid, or that the uncertainties in the "Nuclear Design Methodology" are larger than expected. A limit on the difference between the measured and the predicted rod density of ± 1% Δk/k has been established based on engineering judgment. A > 1% deviation in reactivity from that predicted is larger than expected for normal operation and should therefore be evaluated.

APPLICABILITY

In MODE 1, most of the control rods are withdrawn and steady state operation is typically achieved. Under these conditions, the comparison between predicted and measured core reactivity provides an effective measure of the reactivity anomaly. In MODE 2, control rods are typically being withdrawn during a startup. In MODES 3 and 4, all control rods are fully inserted and therefore the reactor is in the least reactive state, where measuring core reactivity is not necessary. In MODE 5, fuel loading results in a continually changing core reactivity. SDM requirements (LCO 3.1.1) ensure that fuel movements are performed within the bounds of the safety analysis, and an SDM demonstration is required during the first startup following operations that could have altered core reactivity (e.g., fuel movement, control rod replacement, shuffling). The SDM test, required by LCO 3.1.1, provides a direct comparison of the predicted and measured core reactivity at cold conditions; therefore, the Reactivity Anomalies Specification is not required during these conditions.

ACTIONS

A.1

Should an anomaly develop between measured and predicted core reactivity, the core reactivity difference must be restored to within the limit to ensure continued operation is within the core design assumptions. Restoration to within the limit could be performed by an evaluation of the core design and safety analysis to determine the reason for the anomaly. This evaluation normally reviews the core

(continued)

BASES

ACTIONS

A.1 (continued)

conditions to determine their consistency with input to design calculations. Measured core and process parameters are also normally evaluated to determine that they are within the bounds of the safety analysis, and safety analysis calculational models may be reviewed to verify that they are adequate for representation of the core conditions. The required Completion Time of 72 hours is based on the low probability of a DBA occurring during this period, and allows sufficient time to assess the physical condition of the reactor and complete the evaluation of the core design and safety analysis.

B.1

If the core reactivity cannot be restored to within the 1% $\Delta k/k$ limit, the plant must be brought to a MODE in which the LCO does not apply. To achieve this status, the plant must be brought to at least MODE 3 within 12 hours. The allowed Completion Time of 12 hours is reasonable, based on operating experience, to reach MODE 3 from full power conditions in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.1.2.1

Verifying the reactivity difference between the measured and predicted rod density is within the limits of the LCO provides added assurance that plant operation is maintained within the assumptions of the DBA and transient analyses. The 3D Monicore System calculates the rod density for the reactor conditions obtained from plant instrumentation. A comparison of the measured rod density to the predicted rod density at the same cycle exposure is used to calculate the reactivity difference. The comparison is required when the core reactivity has potentially changed by a significant amount. This may occur following a refueling in which new fuel assemblies are loaded, fuel assemblies are shuffled within the core, or control rods are replaced or shuffled. Control rod replacement refers to the decoupling and removal of a control rod from a core location, and subsequent replacement with a new control rod or a control rod from

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.1.2.1 (continued)

another core location. Also, core reactivity changes during the cycle. The 24 hour interval after reaching equilibrium conditions following a startup is based on the need for equilibrium xenon concentrations in the core, such that an accurate comparison between the measured and predicted rod density can be made. For the purposes of this SR, the reactor is assumed to be at equilibrium conditions when steady state operations (no control rod movement or core flow changes) at ≥ 75% RTP have been obtained. The every full power month Frequency was developed, considering the relatively slow change in core reactivity with exposure and operating experience related to variations in core reactivity. This comparison requires the core to be operating at power levels which minimize the uncertainties and measurement errors, in order to obtain meaningful results. Therefore, the comparison is only done when in MODE 1. The tests performed at this Frequency also use base data obtained during the first test of the specific cycle.

REFERENCES

1. UFSAR, Section 16.6.
2. UFSAR, Chapter 14.
3. 10 CFR 50.36(c)(2)(ii).

JAFNPP

IMPROVED STANDARD TECHNICAL SPECIFICATIONS (ISTS) CONVERSION

ITS: 3.1.3

Control Rod OPERABILITY

MARKUP OF CURRENT TECHNICAL SPECIFICATIONS (CTS)

DISCUSSION OF CHANGES (DOCs) TO THE CTS

NO SIGNIFICANT HAZARDS CONSIDERATION (NSHC) FOR LESS RESTRICTIVE CHANGES

MARKUP OF NUREG-1433, REVISION 1, SPECIFICATION

JUSTIFICATION FOR DIFFERENCES (JFDs) FROM NUREG-1433, REVISION 1
MARKUP OF NUREG-1433, REVISION 1, BASES

JUSTIFICATION FOR DIFFERENCES (JFDs) FROM NUREG-1433, REVISION 1, BASES

RETYPED PROPOSED IMPROVED TECHNICAL SPECIFICATIONS (ITS) AND BASES

JAFNPP

IMPROVED STANDARD TECHNICAL SPECIFICATIONS (ISTS) CONVERSION

ITS: 3.1.3

Control Rod OPERABILITY

MARKUP OF CURRENT TECHNICAL SPECIFICATIONS (CTS)

3.1.3 Control Rod Operability

Specification 3.1.3

A1

3.3.A (cont'd)

2. Reactivity margin - inoperable control rods

a. Control rods which cannot be moved with control rod drive pressure shall be considered inoperable. If a partially or fully withdrawn control rod drive cannot be moved with drive or scram pressure, the reactor shall be brought to the Cold Shutdown condition within 24 hours and shall not be restarted unless (1) investigation has shown that the cause of the failure is not a failed control rod drive mechanism collet housing, and (2) adequate shutdown margin has been demonstrated as required by Specification 4.3.A.

If investigation shows that the cause of control rod failure is a cracked collet housing, or if this possibility cannot be ruled out, the reactor shall not be restarted until the affected control rod drive has been replaced or repaired.

L1 ACTION A

add ACTION B M2

ACTION E M5

A1 L6 M2 A2 M2

add Applicability

A1

add LCO 3.1.3 ACTIONS Note

add Required Action A Note

A1

4.3.A (cont'd)

2. Reactivity margin - inoperable control rods

L2 M3

SRs 3.1.3.2 and SR 3.1.3.3

a. Each partially or fully withdrawn operable control rod shall be exercised one notch at least once each week when operating above 30 percent power. In the event power operation is continuing with three or more inoperable control rods, this test shall be performed at least once each day, when operating above 30 percent power.

M4 L3 L4 M4

b. The scram discharge volume drain and vent valves shall be verified open at least once per 31 days (these valves may be closed intermittently for testing under administrative control).

See JTS 3.1.8

c. The status of the pressure and level alarms for each accumulator shall be checked once per week.

See JTS 3.1.5

d. When it is initially determined that a control rod is incapable of normal insertion, an attempt to fully insert the control rod shall be made. If the control rod cannot be fully inserted, shutdown margin test shall be made to demonstrate under this condition that the core can be made subcritical for any reactivity condition during the remainder of the operating cycle with the analytically determined, highest worth control rod capable of withdrawal, fully withdrawn, and all other control rods capable of insertion fully inserted. If Specification 3.3.A.1 and 4.3.A.1 are met, reactor startup may proceed.

A1

Required Action A.4

A3

Page 1 of 4

REVISION D

~~3.3.A.2 (cont'd)~~

[Required Action C.2] b. The control rod directional control valves for inoperable control rods shall be disarmed ~~electrically~~ (LA2)

(M1) ADD: Required Action C.1 within 4 hours

(A5) [CONDITION C] c. Control rods with scram times greater than those permitted by Specification 3.3.C.3 are inoperable, ~~but if they can be inserted with control rod drive pressure they need not be disarmed electrically~~ (M1)

[ADD Required Actions C.1 & C.2]

d. Control rods with inoperable accumulators or those whose position cannot be positively determined shall be considered inoperable. <See ITS 3.1.5>

[SR 3.1.3.1] (M1)

e. Inoperable control rods shall be positioned such that Specification 3.3.A.1 is met.

<ADD Required Action C.1 Note> (A1)

[Required Actions A.4, B.1, C.1] (M5)

[ACTION D] (1) When operating with two or more inoperable control rods in the Startup/Hot Standby or Run modes at ≤ 10% rated thermal power, control rod patterns shall be equivalent to those prescribed by the Banked Position Withdrawal Sequence (BPWS) or else the inoperable control rods shall be separated by two or more operable control rods. If this condition is not met, restore compliance with the condition within 4 hours. Otherwise be in hot shutdown within the following 12 hours.

<ADD ACTION E (for Condition A or C not met)> (M5)

[ACTION E]

[ACTION E] (2) If nine or more control rods are inoperable, be in hot shutdown within 12 hours.

[Condition E second part]

~~4.3.A.2 (cont'd)~~

<See ITS 3.1.8>

e. The scram discharge volume drain and vent valves shall be full-travel cycled at least once per quarter to verify that the valves close in less than 30 seconds and to assure proper valve stroke and operation.

[SR 3.1.3.1]

f. ~~An instrument check of~~ Determine the control rod position ~~indication~~ shall be performed once/day. (LA3)

(A1)

AMEND 255

Specification 3.1.3

A1

3.3 (cont'd)

B. Control Rods

A4

LA2

1. Each control rod shall be coupled to its drive or completely inserted and the control rod directional control valves disarmed electrically. This requirement does not apply in the refuel condition when the reactor is vented. Two control rod drives may be removed as long as Specification 3.3.A.1 is met.

[ACTION C]

M1

A6

4.3 (cont'd)

B. Control Rods

☑ Demonstrate that each control rod drive does not go to the overtravel position:

[SR 3.1.3.5]

a. Each time a control rod is withdrawn to the "full out" position.

b. Prior to declaring a control rod OPERABLE, after work on a control rod or the CRD System that could affect coupling.

2. The control rod drive housing support system shall be in place during reactor power operation or when the reactor coolant system is pressurized above atmospheric pressure with fuel in the reactor vessel, unless all control rods are fully inserted and Specification 3.3.A.1 is met.

2. The control rod drive housing support system shall be inspected after reassembly and the results of the inspection recorded.

L5

A1

add RA c.1 Note

Specification 3.1.3

A1

3.3.C (cont'd)

2. The average of the scram insertion times for the three fastest operable control rods of all groups of four control rods in a two-by-two array shall be no greater than:

Control Rod Notch Position Observed	Average Scram Insertion Time (Seconds)
46	0.361
38	0.977
24	2.112
04	3.764

[SR 3.1.3.4]

3. The maximum scram insertion time for 90 percent insertion of any operable control rod shall not exceed 7.00 sec.

Verify

add Surveillance Frequency

M6

4.3.C (cont'd)

2. At 16-week intervals, 10 percent of the operable control rod drives shall be scram timed above 950 psig. The same control rod drives should not be tested each interval. Whenever such scram time measurements are made, an evaluation shall be made to provide reasonable assurance that proper control rod drive performance is being maintained.

See ITS 3.1.4

See also ITS 3.1.4

3. All control rods shall be determined operable by demonstrating the scram discharge volume drain and vent valves are:

	Item	Frequency
a.	Verified Open	Once per 31 Days
b.	Cycled Fully Closed and Open	In accordance with the Inservice Testing Program
c.	Verified to close within 30 seconds after receipt of an actual or simulated scram signal and open when the actual or simulated scram signal is reset.	Once per 24 Months

See ITS 3.1.8

JAFNPP

IMPROVED STANDARD TECHNICAL SPECIFICATIONS (ISTS) CONVERSION

ITS: 3.1.3

Control Rod OPERABILITY

DISCUSSION OF CHANGES (DOCs) TO THE CTS

DISCUSSION OF CHANGES
ITS: 3.1.3 - CONTROL ROD OPERABILITY

ADMINISTRATIVE CHANGES

A1 In the conversion of the James A. FitzPatrick Nuclear Power Plant (JAFNPP) Current Technical Specification (CTS) to the proposed plant specific Improved Technical Specifications (ITS) certain wording preferences or conventions are adopted which do not result in technical changes. Editorial changes, reformatting, and revised numbering are adopted to make the ITS consistent with the conventions in NUREG-1433, "Standard Technical Specifications, General Electric Plants, BWR/4", Revision 1 (i.e., Improved Standard Technical Specifications (ISTS)).

In Addition, the proposed Control Rod Operability Specification includes all conditions that can affect the ability of the control rods to provide the necessary reactivity insertion. The proposed Specification is also simplified as follows:

1) All inoperable control rods (except stuck rods) are required to be fully inserted and disarmed.

2) A control rod is considered "inoperable" and "stuck" if it is incapable of being inserted. Requirements are retained to preserve Shutdown Margin for this situation and the control rod is required to be disarmed.

3) A control rod is considered "slow" when it is capable of providing the scram function, but may not be able to meet the assumed time limits. The scram reactivity used in the safety analysis allows for a specified number of slow rods.

4) Special considerations are provided for non-conformance to the banked position withdrawal sequence (BPWS), due to inoperable control rods, at ≤ 10% of Rated Thermal Power.

EDIT

Two Notes have also been proposed. The ITS 3.1.3 Action Table Note, "Separate Condition entry is allowed for each control rod," provides more explicit instructions for proper application of the Actions for Technical Specification compliance. In conjunction with the proposed Specification 1.3, "Completion Times," this Note provides direction consistent with the existing Actions for inoperable control rods. It is intended that each inoperable control rod is allowed a specified period of time during which compliance with certain limits is verified and, following which, the control rod is fully inserted and disarmed. The second Note is added to ITS 3.1.3 Required Action for Condition A and Required Action C.1 and allows for bypassing the RWM if necessary for continued operation. This Note is informative in that the RWM may be

ADMINISTRATIVE CHANGES

A1 (continued)

bypassed at any time provided the proper Actions of proposed LCO 3.3.2.1 (the RWM Specification) are taken. This is a human factors consideration to assure clarity of the requirement and allowance for operation.

A2 CTS 3.3.A.2.a requires in part that the plant can not be restarted after finding stuck control rods "unless (1) investigation has shown that the cause of the failure is not a failed control rod drive collet housing, and (2) adequate shutdown margin testing has been demonstrated as required by Specification 4.3.A. If investigation shows that the cause of the control rod failure is a cracked collet housing, or if this possibility cannot be ruled out, the reactor shall not be restarted until the affected control rod drive has been replaced or repaired."

In the proposed ITS, the ability to change MODES is generically controlled by the provisions of LCO 3.0.4 which states in part that "when an LCO is not met, entry into a MODE or other specified condition in the Applicability shall not be made except when the associated ACTIONS to be entered permit continued operation in the MODE or other specified condition in the Applicability for an unlimited period of time." ITS 3.1.3 ACTION B requires that the plant be shutdown to MODE 3 within 12 hours of finding two or more withdrawn stuck control rods, and therefore LCO 3.0.4 would prevent plant startup with two or more withdrawn stuck control rods. Therefore, this proposed change causes no technical or actual change from present specifications if two or more control rods are stuck (one stuck control rod is addressed in DOC L1). Therefore, the change is considered administrative, and is consistent with NUREG-1433, Revision 1.

A3 CTS 4.3.A.2.d requires in part that for stuck control rods that a SDM test be performed "to demonstrate under this condition that the core can be made subcritical for any reactivity condition during the remainder of the operating cycle with the analytically determined, highest worth control rod capable of withdrawal, fully withdrawn, and all other control rods capable of insertion, fully inserted." ITS Required Action A.4 requires SR 3.1.1.1 to be performed if a rod is withdrawn and can not be inserted (stuck). SR 3.1.1.1 is the proposed SDM test. In the proposed ITS, the definition for Shutdown Margin (SDM) requires in part that "all control rods are fully inserted except for the single control rod of highest reactivity worth, which is assumed to be fully withdrawn. With control rods not capable of being fully inserted, the reactivity worth of these control rods must be accounted for in the determination of SDM." Therefore the present requirements of CTS 4.3.A.2.d have been

ADMINISTRATIVE CHANGES

A3 (continued)

incorporated into the definition of SDM, and its removal from this Specification will result in no technical change to plant operations. Therefore, this change is considered administrative, and is consistent with NUREG-1433, Revision 1.

A4 The CTS 3.3.B.1 requirement that control rods be coupled to the drive is presented in SR 3.1.3.5, making it a requirement for control rods to be considered OPERABLE. The actions for uncoupled control rods remain in LCO 3.1.3, ACTION C. Eliminating the separate LCO for control rod coupling, by moving the surveillance and actions to another Specification (as a Surveillance Requirement), does not eliminate any requirements, or impose a new or different treatment of the requirements. Therefore, this proposed change is considered administrative.

A5 CTS 3.3.A.2.c requires that control rods with scram times greater than those permitted in CTS 3.3.C.3 be declared inoperable. The requirement that maximum control rod scram insertion time to notch position 4 be ≤ 7 seconds is presented in ITS SR 3.1.3.4, making it a requirement for control rods to be considered Operable. Eliminating the separate Specification for excessive scram time by moving the requirement to a Surveillance Requirement, does not eliminate any of the requirements, or impose a new or different treatment of the requirement, except as provided in Comment M1. Therefore, this proposed change is considered administrative.

A6 This requirement in CTS 3.3.B.1 (This requirement does not apply in the refuel condition) duplicates an identical and more appropriately placed requirement in CTS 3.10.A.5 (ITS 3.10.6). Therefore, deletion of this requirement is an administrative change.

TECHNICAL CHANGES - MORE RESTRICTIVE

M1 CTS 3.3.A.2.b requires the directional control valves for inoperable control rods to be disarmed, CTS 3.3.B.1 requires the same action for uncoupled control rods. However, CTS 3.3.A.2.c allows a control rod inoperable due to a scram time greater than 7 seconds to not be disarmed, provided it can be inserted. ITS 3.1.3 Required Actions C.1 and C.2 have been added, such that if a rod is considered inoperable for any reason (including excessive scram time), it must be fully inserted within 3 hours (unless it is stuck) and disarmed within 4 hours. This is more restrictive than current requirements and is necessary to ensure

TECHNICAL CHANGES - MORE RESTRICTIVE

M1 (continued)

timely action is taken to maintain scram reactivity assumptions.

M2 CTS 3.3.A.2.a will allow the plant to restart and continue operation with multiple stuck control rods if: 1) collet housing failure is eliminated as a potential cause; 2) sufficient control rods remain operable to make the core subcritical with the most reactive rod fully withdrawn (i.e., SDM is maintained); and 3) the stuck rod is disarmed. The proposed change will require Hot Shutdown (MODE 3) within 12 hours when more than one control rod is stuck but not fully inserted, regardless of the reasons for the stuck control rods. More than one stuck control rod (not fully inserted) will require Hot Shutdown within 12 hours (ITS 3.1.3 Required Action B.1) because the assumptions utilized in establishing the proposed scram time limits account for only a single stuck control rod.

M3 CTS 4.3.A.2.a requires that control rods be "exercised one notch." Proposed surveillances SR 3.1.3.2 and SR 3.1.3.3 require control rods to be "inserted" at least one notch, in lieu of the existing requirement for "exercising." The existing requirement could be met by control rod withdrawal. It is conceivable that a mechanism causing binding of the control rod that prevents insertion could exist and that a withdrawal test would not detect the problem. Since the purpose of the test is to assure scram insertion capability, restricting the test to control rod insertion provides an increased likelihood of this test detecting a problem that impacts insertion capability.

M4 The Surveillance condition described in CTS 4.3.A.2.a as "above 30% rated thermal power" is proposed to be changed to "Thermal Power is greater than the LPSP of the RWM," and shown in the form of a Note to proposed SRs 3.1.3.2, 3.1.3.3 and in the proposed Required Action A.3 Completion Time. Since the LPSP is set well below the 30% RTP level (the RWM must be operable equal to and less than 10% RTP), this change is more restrictive than present requirements but does not impose any safety concerns since at power levels above the LPSP notch insertions will not impact the requirements of the Banked Position Withdrawal Sequence. This change is necessary to ensure that control rod insertion capability is verified at the earliest opportunity in the applicable condition. This does not represent any change in safety and is consistent with NUREG-1433, Revision 1.

M5 CTS 3.3.A.2.e requires that inoperable (and stuck) control rods be positioned such that SDM requirements (CTS 3.3.A.1) are maintained. CTS 3.3.A.2.a requires the reactor to be in Cold Shutdown within

TECHNICAL CHANGES - MORE RESTRICTIVE

M5 (continued)

24 hours (see L6) when a control rod is first found to be stuck. ITS 3.1.3 requires that: with one stuck rod (Required Action A.4), SDM be verified within 72 hours; with more than one stuck rod (Required Action B.1), the reactor be in Hot Shutdown within 12 hours; and, with one or more inoperable rods (Required Action C.1) that each inoperable rod be fully inserted. If the requirements of Required Action A.4 and C.1 cannot be met the reactor must be placed in MODE 3 in 12 hours (ITS 3.1.3 Required Action E.1).

By allowing only one stuck rod, and by requiring that all inoperable rods be fully inserted, proposed ITS 3.1.3 Required Actions A.4, B.1, and C.1 provide greater assurance that SDM will be maintained than the current requirement for verifying SDM for multiple rods that remain withdrawn.

M6 CTS 3.3.C.3 requires that the maximum scram insertion for 90 percent insertion of any control rod be less than 7.00 sec. This requirement is included in ITS SR 3.1.3.4 however an explicit Surveillance Frequency has been added. The proposal Frequency is in accordance with SR 3.1.4.1 (CTS 4.3.4.1), SR 3.1.4.2 (CTS 4.3.C.2), SR 3.1.4.3, and SR 3.1.4.4. Since the Surveillance Frequencies for determining control rod scram times have been supplemented (see Discussion of Changes for ITS 3.1.4) this change is considered more restrictive. This change is necessary to help ensure control rod operability.

TECHNICAL CHANGES - LESS RESTRICTIVE (GENERIC)

LA1 CTS 4.3.A.2.d states that an attempt should be made to fully insert a control rod if it is initially determined to be incapable of normal insertion. This requirement for attempting control rod insertion is proposed to be relocated to plant procedures. ITS 3.1.3 ACTION A is adequate to control what to do with a control rod that is stuck. ITS 3.1.3 Required Action A.2 requires the disarming of the associated control rod drive within 3 hours. Up until this Completion Time nothing precludes the Operators from attempting to insert the control rod using plant procedures, therefore, these details are not required to be in the ITS to provide adequate protection of the public health and safety. Changes to plant procedures are controlled by the provisions of plant administrative control process.

TECHNICAL CHANGES - LESS RESTRICTIVE (GENERIC)

LA2 Details of the methods for disarming control rod drives (CRDs) in CTS 3.3.A.2.b and CTS 3.3.B.1 are proposed to be relocated to the Bases. These details are not necessary to ensure the associated CRDs of inoperable control rods are disarmed. The requirement in ITS 3.1.3 Required Actions A.2 and C.2, which require disarming the associated CRDs of inoperable control rods, are adequate for ensuring associated CRDs and inoperable control rods are disarmed, therefore these details are not required to be in the ITS to provide adequate protection of the public health and safety. Changes to the Bases will be controlled by the provisions of the proposed Bases Control Program described in Chapter 5 of the Technical Specifications.

LA3 The method used to determine the position of each control rod in CTS 4.3.A.2.f (an instrument check of control rod position indication) is proposed to be relocated to the Bases. This instrument check is performed to determine the position of each control rod. The requirement in ITS SR 3.1.3.1 to determine the position of each control rod is sufficient to ensure adequate information on control rod position is available to the operator for determining control rod OPERABILITY and controlling rod patterns. The methods used to determine the position of each control rod is not required to be in the ITS to provide adequate protection of the public health and safety. Changes to the Bases will be controlled by the provisions of the proposed Bases Control Program described in Chapter 5 of the Technical Specification.

TECHNICAL CHANGES - LESS RESTRICTIVE (SPECIFIC)

L1 Currently in CTS 3.3.A.2.a, a stuck control rod (not fully inserted) that may be stuck as a result of a collet housing failure or for some other reason requires that the reactor be in a cold shutdown condition within 24 hours. No allowance is provided for repair prior to entering the shutdown statement. The proposed Specification (ITS 3.1.3 ACTION A) allows continued operation with a stuck control rod the and the requirement to be in Cold Shutdown has been deleted. With a single withdrawn control rod stuck, the remaining Operable control rods are capable of providing the required scram and shutdown reactivity. The assumptions utilized in establishing the proposed scram time limits account for a single stuck control rod in addition to an assumed single failure during a transient. To ensure that local scram reactivity assumptions are maintained in this condition, stuck control rod separation criteria must be verified (ITS 3.1.3 Required Action A.1). ITS 3.1.3 Required Action A.2 is also added to disarm the stuck control rod within 2 hours to prevent damaging the control rod drive.

TECHNICAL CHANGES - LESS RESTRICTIVE (SPECIFIC)

L1 (continued)

Shutdown Margin must still be met, accounting for the loss of negative reactivity due to the stuck control rod (refer to the proposed definition of SDM and proposed Required Action A.4 of LCO 3.1.3). In addition, a time limit of 72 hours on the Shutdown Margin determination has been provided. The existing limitation on reactor startup based on the reason for the failure (e.g., failed collet housing) has been eliminated. The particular failure mechanism is not significant, provided all other rods are tested to ensure a similar failure has not occurred. Proposed ITS 3.1.3 Required Action A.3 performs this check within 24 hours from discovery of the stuck withdrawn control rod with Thermal Power greater than the low power setpoint of the RWM to confirm that no additional stuck control rods exist. Therefore, continued operation is proposed to be allowed.

L2 The existing surveillance (CTS 4.3.A.2.a) requires that all partially or fully withdrawn control rods be exercised at least once per week. The proposed requirements (SR 3.1.3.2 and SR 3.1.3.3) will differentiate between fully and partially withdrawn rods. Fully withdrawn rods will still be exercised once per 7 days. However, partially withdrawn rods will be exercised once per 31 days. This is in accordance with NUREG-1433, Revision 1. The reason for decreasing the frequency for exercising partially withdrawn rods from 7 to 31 days is that partially withdrawn control rods have a significantly greater effect on core flux distribution than do fully withdrawn control rods. Power reductions could conceivably be required each week to perform this test on the partially withdrawn control rods. This potential impact on plant operator is not warranted given the following considerations:

1) At full power a large percentage of control rods (typically 80-90%) are fully withdrawn and would continue to be exercised each week. This represents a significant sample size when looking for an unexpected random event or systemic problem.

2) Operating experience has shown "stuck" control rods to be a rare event while operating.

3) Should a stuck rod be discovered, all of the remaining control rods (even partially withdrawn) must be exercised within 24 hours (proposed Required Action A.3).

4) Power reduction and restoration to the pre-test power conditions may induce a thermal transient.

TECHNICAL CHANGES - LESS RESTRICTIVE (SPECIFIC)

L2 (continued)

Therefore, extending the surveillance interval for exercising partially withdrawn control rods from 7 to 31 days is justified.

L3 Currently in CTS 4.3.A.2.a, if three or more control rods are inoperable but not stuck, all operable control rods must be exercised once every 24 hours. The proposed requirement for control rods that are inoperable but not stuck, ITS 3.1.3 Required Actions C.1 and C.2, is to fully insert and disarm the inoperable rod(s), respectively. There will be no requirement to exercise the operable rods to verify their operability other than the scheduled surveillance requirements in SR 3.1.3.2 and SR 3.1.3.3. Since an inoperable rod that is not stuck can be inserted, a verification that all rods can be inserted does not contribute to the identification of a generic failure that reduces scram capability. For a stuck control rod, ITS 3.1.3 Required Action A.3 will still require that all operable rods be inserted at least one notch to verify that the stuck control rod is not caused by a generic failure that would interfere with scram capability.

L4 Currently in CTS 4.3.A.2.a, if one or more control rods are stuck, all operable control rods must be exercised "at least every 24 hours." In the proposed change, after discovery of a stuck rod, all withdrawn control rods are required to be exercised only once within 24 hours when Thermal Power is greater than the low power setpoint of the RWM (ITS 3.1.3 Required Action A.3.) This provides adequate assurance that the cause of the stuck rod is not of generic concern. Thereafter, continued testing of control rods per the normal frequency is sufficient to ensure continued operability of the remaining control rods.

L5 The CTS 3/4.3.B.2 requirement for the CRD housing support to be in place is included in the Operability requirements for control rods. Plant configuration management provides adequate controls to assure the CRD housing support is in place. The current Technical Specifications require inspections of the CRD housing support following reassembly. The current Technical Specifications requirement verifies that the CRD housing support is in place for reactor operation in MODES 1, 2 and 3. Post-maintenance inspections conducted through plant configuration management control have the same function as the current Technical Specifications requirement. Since work is not normally performed on the CRD housing support at power, and checks on its installation are not made at power, there is no current requirement to verify CRD housing support installation in power operating conditions. Therefore, the deletion of this current Technical Specification is acceptable based on housing support installation.

TECHNICAL CHANGES - LESS RESTRICTIVE (SPECIFIC)

L6 CTS 3.3.A.2.a requires the plant to be brought to Cold Shutdown within 24 hours when the requirements for "Inoperable Control Rods" cannot be met. This implies the Applicability of CTS 3.3.A.2.a to be Modes 1, 2 and 3. The Applicability in ITS 3.1.3 is Modes 1 and 2 and the default condition has been changed to Mode 3 as reflected in ITS 3.1.3 Required Action B.1 and E.1. Placing the plant in MODE 3 ensures all control rods are fully inserted and that the Applicability of the LCO is exited. Cooling down the plant does not provide any additional reactivity margin and, in some cases, could be counterproductive since positive reactivity is inserted during a cooldown. Given that the only difference between MODES 3 and 4 is the temperature requirement, the safety impact of this change as it relates to control rods and the safety analysis they affect, is negligible. The default condition is consistent with that currently allowed in CTS 3.3.A.2.e for other control rod inoperabilities. This change is consistent with NUREG-1433, Revision 1.

TECHNICAL CHANGES - RELOCATIONS

None

JAFNPP

IMPROVED STANDARD TECHNICAL SPECIFICATIONS (ISTS) CONVERSION

ITS: 3.1.3

Control Rod OPERABILITY

NO SIGNIFICANT HAZARDS CONSIDERATION (NSHC) FOR LESS RESTRICTIVE CHANGES

NO SIGNIFICANT HAZARDS CONSIDERATIONS
ITS: 3.1.3 - CONTROL ROD OPERABILITY

TECHNICAL CHANGES - LESS RESTRICTIVE (SPECIFIC)

L1 CHANGE

New York Power Authority has evaluated the proposed Technical Specification change identified as "Technical Changes - Less Restrictive" and has determined that it does not involve a significant hazards consideration. This determination has been performed in accordance with the criteria set forth in 10 CFR 50.92. The bases for the determination that the proposed change does not involve a significant hazards consideration are discussed below.

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

   The probability of an accident is not increased because the proposed change will not involve any physical changes to plant systems, structures, or components (SSC) or the manner in which these SSC are operated, maintained, modified, tested, or inspected. Elimination of the requirement to shutdown if one control rod is stuck because of a potential collet retainer tube failure is being made concurrently with another change that will require a reactor shutdown if more than one control rod is stuck for any reason. This additional restriction ensures that the reactor will be shut down as soon as it is determined that more than one control rod may fail to scram and the reactor may fall outside of the assumptions used in the analysis of those accidents and transients that depend on a scram. This differs from the existing requirement that allows operating with multiple stuck control rods that are not fully inserted. Eliminating the actions required for one particular failure mechanism (i.e., failed collet retainer tube) is not significant provided all other rods are tested to ensure a similar failure has not occurred to another control rod. This verification is performed as part of the proposed actions. Therefore, eliminating the requirement to shutdown if one control rod is stuck because of a potential collet retainer tube failure will not involve a significant increase in the probability or consequences of an accident previously evaluated.

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

   The proposed change will not involve any physical changes to plant systems, structures, or components (SSCs), or the manner in which these SSCs are operated, maintained, modified, tested, or inspected. Therefore, this change will not create the possibility of a new or different kind of accident from any accident previously evaluated.

TECHNICAL CHANGES - LESS RESTRICTIVE (SPECIFIC)

L1 CHANGE

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

This proposed change involves the elimination of a requirement to shutdown if one control rod is stuck due to a potential collet retainer tube failure and will not involve any physical changes to plant systems, structures, or components (SSCs). Additionally, elimination of a requirement to shutdown if one control rod is stuck due to potential collet retainer tube failure will not decrease a margin of safety because this change is being made concurrently with another change that will require a reactor shutdown if more than one control rod is stuck for any reason. This additional restriction ensures that the reactor will be shut down as soon as it is determined that more than one control rod may fail to scram, and the reactor falls outside of the assumptions used in the analysis of those accidents and transients that depend on a scram. This differs from the existing requirement that allows operation with multiple stuck control rods that are not fully inserted. Therefore, this change does not involve a significant reduction in a margin of safety.

NO SIGNIFICANT HAZARDS CONSIDERATIONS
ITS: 3.1.3 - CONTROL ROD OPERABILITY

TECHNICAL CHANGES - LESS RESTRICTIVE (SPECIFIC)

L2 CHANGE

New York Power Authority has evaluated the proposed Technical Specification change identified as "Technical Changes - Less Restrictive" and has determined that it does not involve a significant hazards consideration. This determination has been performed in accordance with the criteria set forth in 10 CFR 50.92. The bases for the determination that the proposed change does not involve a significant hazards consideration are discussed below.

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

   The proposed change increases the interval between performances of a surveillance designed to verify that control rods are not stuck and that scram capability is maintained. The proposed change will not involve any physical changes to plant systems, structures, or components (SSC), or the manner in which these SSC are operated, maintained, modified, tested, or inspected. The increased frequency interval does not apply to the fully withdrawn control rods, which represent a significant sample size (80-90%) at full power in evaluating this infrequent random event. The proposed frequency of the surveillance is based on engineering judgment and the accumulated industry experience with CRD performance, which shows it to be highly reliable. The proposed change will not increase the consequences of an accident because this change is being implemented concurrently with more restrictive requirements governing continued operation with stuck and inoperable control rods. Collectively, these changes provide assurance that when a scram is required, the assumptions used in the accident analysis (i.e., most reactive control rod fully withdrawn) will be met. Therefore, this change will not involve a significant increase in the probability or consequences of an accident previously evaluated.

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

   The proposed change increases the interval between performances of a surveillance designed to verify that control rods can be inserted and will not involve any physical changes to plant systems, structures, or components (SSCs). Therefore, this change will not create the possibility of a new or different kind of accident from any accident previously evaluated.

TECHNICAL CHANGES - LESS RESTRICTIVE (SPECIFIC)

L2 CHANGE

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

A margin of safety is not reduced even though the proposed increase in the interval between performances of a surveillance may increase the time before an untrippable control rod is discovered. The increased frequency interval does not apply to the fully withdrawn control rods, which represent a significant sample size in evaluating this infrequent random event. The proposed frequency of the surveillance is based on engineering judgment and the accumulated industry experience with CRD performance. Additionally, this change is being implemented concurrently with more restrictive requirements governing continued operation with stuck and inoperable control rods. Collectively, these changes provide assurance that when a scram is required, the assumptions used in the accident analysis (i.e., most reactive control rod fully withdrawn) will be met. Therefore, this change does not involve a significant reduction in a margin of safety.

TECHNICAL CHANGES - LESS RESTRICTIVE (SPECIFIC)

L3 CHANGE

New York Power Authority has evaluated the proposed Technical Specification change identified as "Technical Changes - Less Restrictive" and has determined that it does not involve a significant hazards consideration. This determination has been performed in accordance with the criteria set forth in 10 CFR 50.92. The bases for the determination that the proposed change does not involve a significant hazards consideration are discussed below.

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

   The proposed change deletes the existing requirement that all control rods be exercised once every 24 hours if 3 or more control rods are inoperable. The proposed change will not involve any physical changes to plant systems, structure, or components (SSCs), or the manner in which these SSCs are operated, maintained, modified, tested, or inspected. The proposed change is being implemented concurrently with more restrictive requirements governing continued operation with stuck and inoperable control rods. Since an inoperable control rod that is not stuck can be inserted, a verification that all rods can be inserted does not contribute to the identification of a generic failure that reduces scram capability. These more restrictive requirements include fully inserting all inoperable control rods within 3 hours and disarming these control rods within 4 hours (LCO 3.1.3 Condition C) and requiring reactor shutdown within 12 hours if more than one control rod is stuck (LCO 3.1.3 Condition B). Collectively, these changes provide assurance that when a scram is required, the assumptions used in the accident analysis (i.e., most reactive control rod fully withdrawn) will be met. Therefore, this change will not involve a significant increase in the probability or consequences of an accident previously evaluated.

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

   The proposed change will not involve any physical changes to plant systems, structures, or components (SSCs), or the manner in which these SSCs are operated, maintained, modified, tested, or inspected. Therefore, this change will not create the possibility of a new or different kind of accident from any accident previously evaluated.

TECHNICAL CHANGES - LESS RESTRICTIVE (SPECIFIC)

L3 CHANGE

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

No margins of safety are being reduced. The proposed change is being implemented concurrently with more restrictive requirements governing continued operation with stuck and inoperable control rods. Since an inoperable control rod that is not stuck can be inserted, a verification that all rods can be inserted does not contribute to the identification of a generic failure that reduces scram capability. These more restrictive requirements include fully inserting all inoperable control rods within 3 hours (LCO 3.1.3 Condition C) and requiring reactor shutdown within 12 hours if more than one control rod is stuck (LCO 3.1.3 Condition B). Collectively, these changes provide assurance that when a scram is required, the assumptions used in the accident analysis (i.e., most reactive control rod fully withdrawn) will be met. Therefore, this change does not involve a significant reduction in a margin of safety.

TECHNICAL CHANGES - LESS RESTRICTIVE (SPECIFIC)

L4 CHANGE

New York Power Authority has evaluated the proposed Technical Specification change identified as "Technical Changes - Less Restrictive" and has determined that it does not involve a significant hazards consideration. This determination has been performed in accordance with the criteria set forth in 10 CFR 50.92. The bases for the determination that the proposed change does not involve a significant hazards consideration are discussed below.

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

   The proposed change replaces the daily control rod notch test required when operating with stuck control rods, with one performed once within 24 hours when Thermal Power is greater than the low power setpoint of the RWM. The intent of the current daily test of control rods is to ensure that a generic problem does not exist and that control rod insertion capability remains. The proposed single performance provides the information to be used in determining whether a generic problem exists and control rod insertion capability remains.

   The proposed change does not affect an accident precursor and, therefore, does not involve a significant increase in the probability of an accident previously evaluated. The proposed Frequency change for the control rod notch test will still provide the operator with the necessary information to be used in determining whether control rod insertion capability remains. Therefore, the proposed change does not involve a significant increase in the consequences of an accident previously evaluated.

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

   The possibility of a new or different kind of accident from any accident previously evaluated is not created because the proposed change does not introduce a new mode of plant operation and does not involve physical modifications to the plant.

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

   The performance of the test once within 24 hours when Thermal Power is greater than the low power setpoint of the RWM, instead of the current

TECHNICAL CHANGES - LESS RESTRICTIVE (SPECIFIC)

L4 CHANGE

3. (continued)

daily test when a control rod is stuck, is an adequate indicator of system problems without having to perform additional, unnecessary testing. Therefore, the proposed change does not involve a significant reduction in a margin of safety.

TECHNICAL CHANGES - LESS RESTRICTIVE (SPECIFIC)

L5 CHANGE

New York Power Authority has evaluated the proposed Technical Specification change identified as "Technical Changes - Less Restrictive" and has determined that it does not involve a significant hazards consideration. This determination has been performed in accordance with the criteria set forth in 10 CFR 50.92. The bases for the determination that the proposed change does not involve a significant hazards consideration are discussed below.

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

   The CRD housing support restricts the outward movement of a control rod to less than 3 inches in the extremely remote event of a CRD housing failure. The CRD housing support is not an accident initiator or precursor and, as such, cannot contribute to an increase in the probability of an accident previously evaluated. The deletion of this Specification does not result in the removal of the requirement to verify proper installation of the CRD housing support. Plant configuration management controls ensure through post-maintenance testing and inspections that the proper configuration for the CRD housing supports is maintained. These controls are currently in place and are used to ensure this system and other plant systems are properly configured prior to being considered Operable for plant operation. Based on the controls that the plant has in place to ensure the CRD housing support is properly installed, the change does not involve a significant increase in the consequences of an accident previously evaluated.

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

   The proposed change does not involve physical alteration of the plant (no new or different type of equipment will be installed) or changes in the methods governing normal plant operation. The proposed change does not impose requirements different from those being used for normal post-maintenance inspections to ensure the CRD housing support is properly installed. The proposed change will rely on plant configuration management controls to ensure that this system and other plant systems are returned to their design configuration condition. Thus, this change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

TECHNICAL CHANGES - LESS RESTRICTIVE (SPECIFIC)

L5 CHANGE

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

The CRD housing support Technical Specification ensures proper installation of this system during MODES 1, 2 and 3. The installation checks are performed while the plant is shutdown and are necessary only after work has been done to alter the system configuration. These post-maintenance checks are currently performed by procedural control on this and other plant systems. The use of present plant configuration management controls will ensure that these systems meet design requirements. Therefore, the proposed change does not involve a significant reduction in a margin of safety.

TECHNICAL CHANGES - LESS RESTRICTIVE (SPECIFIC)

L6 CHANGE

New York Power Authority has evaluated the proposed Technical Specification change identified as "Technical Changes - Less Restrictive" and has determined that it does not involve a significant hazards consideration. This determination has been performed in accordance with the criteria set forth in 10 CFR 50.92. The bases for the determination that the proposed change does not involve a significant hazards consideration are discussed below.

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

   The requirement to place the plant in MODE 4 within 24 hours has been deleted and the new Applicability for control rods is MODES 1 and 2. Placing the plant in MODE 3 (see M5) ensures all control rods are inserted and that the Applicability of the LCO is exited. With the plant in MODE 3, all rods are fully inserted, and will remain inserted since the mode switch, while in the shutdown position, enforces a rod block. Therefore, a reactivity control accident related to control rods cannot occur. Cooling down the plant does not provide any additional reactivity margin and, in some cases, could be counterproductive since positive reactivity is inserted during a cooldown. Given that the only difference between MODES 3 and 4 is the temperature requirement, the safety impact of this change as it relates to control rods and the safety analysis they affect, is negligible. Shutdown Completion Times are not considered in the initiation of any accident previously analyzed. Thus, 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 introduce a new mode of plant operation and does not involve physical modification to the plant. Therefore, it does not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

   The requirement to place the plant in MODE 4 within 24 hours has been deleted and the new Applicability for control rods is MODES 1 and 2. Requiring the plant to be placed in MODE 3 within 12 hours (M5) will improve the margin of safety. This is due to the <u>positive</u> reactivity inserted due to a plant cooldown (A decrease in reactor coolant temperature results in a positive reactivity addition.) In addition,

TECHNICAL CHANGES - LESS RESTRICTIVE (SPECIFIC)

L6 CHANGE

3. (continued)

the proposed change will require the plant to have all rods to be inserted (MODE 3) within 12 hours versus the current 24 hours. The shutdown Completion Time is considered acceptable since it helps ensure a steady decrease in power and reduces the chances of a plant transient which could challenge safety systems. The requirement to be in MODE 3 also exits the Applicability of the LCO and there is no need to continue to MODE 4. Therefore, this change does not involve a significant reduction in a margin of safety.