Wisconsin Electric
A WISCONSIN ENERGY COMPANY

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NPL 2000-0260 10 CFR 50.90

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Document Control Desk
U.S. NUCLEAR REGULATORY COMMISSION
Mail Station P1-137
Washington, DC 20555

Ladies/Gentlemen:

DOCKETS 50-266 AND 50-301
SUPPLEMENT 2 TO APPLICATION FOR AMENDMENT TO
FACILITY OPERATING LICENSE APPENDIX A
TECHNICAL SPECIFICATIONS IMPROVEMENT PROJECT
RESPONSE TO RAI ON ITS SECTIONS 2.0, 3.1, 3.2, AND 3.5
POINT BEACH NUCLEAR PLANT, UNITS 1 AND 2

On November 15, 1999, Wisconsin Electric Power Company (WE), licensee for the Point Beach Nuclear Plant (PBNP), submitted an application to amend Appendix A, Technical Specifications, for Facility Operating Licenses DPR-24 and DPR-27 for Point Beach Nuclear Power Plant, Units 1 and 2, respectively (reference letter NPL 99-0669). The application proposed to convert the Point Beach Current Technical Specifications (CTS) to the Point Beach Improved Technical Specifications (ITS). That application contained documentation for ITS Chapters 1.0 and 2.0 and Sections 3.0 through 3.9. Documentation for ITS Chapters 4.0 and 5.0 was enclosed with Supplement 1 to the PBNP ITS submittal dated March 15, 2000 (reference letter NPL 2000-0142).

In a letter dated May 5, 2000, the NRC issued a Request for Additional Information (RAI) to WE on ITS sections 3.1, 3.2, and 3.5. In a letter dated May 15, 2000, the NRC issued a RAI to WE on ITS section 2.0.

Attachment 1 of this letter includes our response to the Staff's questions in the above referenced RAIs. In some instances, the response includes changes that are required to the original submittal, including changes to the Current Technical Specification (CTS) markups, Descriptions of Change (DOC), NUREG markups, proposed ITS and associated Bases, Justifications for Deviation (JFD), and No Significant Hazard Considerations (NSHC). These changes are discussed in the response to each question and are included in the attachment. Pages containing the changes required to the DOC, JFD, and NSHC are identified by "Rev. B". Pages containing the old DOCs, JFDs, and NSHCs in the original submittal should be replaced with the new pages enclosed with this letter, following the instructions of attachment 2.

NRR-057

A001

The changes required to the CTS, NUREG, and ITS markups are identified as follows (example):

B

RAI 3.6.4-2

The revision bar identifies the section that has been revised; the B in the triangle identifies revision B; and the RAI number identifies which RAI question the revision relates to.

Additional changes to the conversion package for the subject ITS Sections have been identified as a result of ITS reviews by WE staff and Amendment approvals that have occurred after the original ITS submittal. These additional changes have been included (where necessary) in response to each RAI question for completeness and are clearly identified in the new pages enclosed with this letter.

Wisconsin Electric has determined that this supplement does not involve a significant hazards consideration, authorize a significant change in the types or total amounts of effluent release, or result in any significant increase in individual or cumulative occupational radiation exposure. Therefore, Wisconsin Electric concludes that the proposed supplement meets the categorical exclusion requirements of 10 CFR 51.22(c)(9) and that an environmental impact appraisal need not be prepared.

Wisconsin Electric is notifying the State of Wisconsin of this supplement by transmitting a copy of this letter, and its attachments, to the Public Service Commission of Wisconsin.

To the best of my knowledge and belief, the statements contained in this document are true and correct. In some respects, these statements are not based entirely on my personal knowledge, but on information furnished by cognizant Wisconsin Electric employees, contractor employees, and/or consultants. Such information has been reviewed in accordance with company practice, and I believe it to be reliable.

Should you have any questions on this submittal or require additional information, please contact me.

Sincerely,

Richard L. Mende

For Mark Reddemann
Site Vice President
Point Beach Nuclear Plant

Subscribed to and sworn before me
on this 16th day of June, 2000

Tammy J. Augustt

Notary Public, State of Wisconsin

My Commission expires on 4-27-03.

Enclosures

cc: NRC Regional Administrator
NRC Project Manager
NRC Resident Inspector
PSCW

DOCKETS 50-266 AND 50-301
RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION
TECHNICAL SPECIFICATIONS IMPROVEMENT PROJECT SECTION 2.0, 3.1, 3.2, and 3.5
POINT BEACH NUCLEAR PLANT UNITS 1 AND 2

The following information is provided in response to the Nuclear Regulatory Commission staff's requests for additional information dated May 5, 2000 and May 15, 2000.

Each question is restated on the following pages with Wisconsin Electric's response following.

NRC Question 2.0-1:

2.0-1 ITS B2.0 REFERENCES
STS B2.0 REFERENCES
JFD 4

References to specific sections of the FSAR for safety analyses initial conditions are not appropriate because they are fragmented throughout the FSAR. Also, RPS setpoints are not contained in the FSAR, they are in the TS, and therefore FSAR references have been deleted. Comment: To the extent possible include appropriate references or a brief discussion where the information is contained.

WE Response:

WE has reviewed this question and the JFD and has determined that the NRC recommendation should be implemented. PBNP FSAR section 7.2 describes the Reactor Protection System (RPS). The section states that the RPS setpoints are established in the Technical Specifications, and, in some cases, identifies the applicable sections. It also states that the analytical limits established in the accident analyses (FSAR Chapter 14) determine the point at which the variables require actuation. Therefore, reference to FSAR 7.2 is appropriate here in the Bases of ITS 2.1.

As a result of this change, JFD 04 of this section was changed to "not used" and reference 2 was changed to "FSAR 7.2" in the NUREG markup and proposed ITS Bases (see attached markups).

Additional Corrections Required to ITS Section 2.0:

Additional corrections to the conversion package for this section have been identified as a result of ITS reviews and TSCR submittals and have been included in this RAI response for completeness.

Changes have been indicated in the CTS markup, NUREG markup and proposed ITS Bases as a result of TSCR 218 (COLR submittal) and Amendments 193/198, approved 2/08/00 (related to

incorporation of new fuel design). As a result of incorporating these submittals, DOC A.02 was changed to "not used," new DOC LA.02 was created, and JFD 02 and JFD 08 are no longer required and have been changed to "not used" (see attached markups).

NRC Question 3.1.1-1:

3.1.1-1 STS 3.1.1, SDM
ITS 3.1.1, SDM
ITS 5.6.4, COLR
CTS 15.03.10
DOC LA1

Point Beach is adopting a COLR, and it is stated that, "changes to the COLR limits will be controlled in accordance with 10 CFR 50.59."

Comment: Normally, COLRs are established in accordance with requirements set forth in an Administrative Controls section, and controlled in accordance with another Administrative Controls section. To be controlled in accordance with 10 CFR 50.59, the COLR would have to be incorporated into the FSAR, which is acceptable though not typical. Is the Point Beach COLR to be incorporated into the FSAR?

WE Response:

The PBNP COLR will be included in a newly developed "Technical Requirements Manual" (TRM). The TRM will be incorporated by reference into the FSAR; therefore, changes to the TRM will be done in accordance with 10 CFR 50.59 (also a requirement in newly created administrative procedure for the TRM). The COLR will be transmitted to the NRC in accordance with the requirements of proposed ITS 5.6.4.d, which states "the COLR, including any midcycle revisions or supplements, shall be provided upon issuance for each reload cycle to the NRC."

NRC Question 3.1.1-2:

3.1.1-2 ITS B3.1.1
STS B3.1.1
ITS SR 3.1.1.1, SDM Verification
JFD 10

Reference to a Reactivity Balance Calculation for verifying SDM is removed from the Bases because an alternative method of using a bounding boron concentration is used.

Comment: Include a brief discussion of the method(s) used for determining SDM in the Bases discussion of the SR.

WE Response:

WE has reviewed this question and the JFD and has determined that the NRC recommendation should not be implemented because a description of SDM verification methods in the Bases is not necessary. Shutdown margin verification is done more frequently in ITS (Modes 2, 3, 4, 5) than CTS (only during low power physics testing). Performing a reactivity balance calculation is a lengthy process and inappropriate when the unit is in the Mode of applicability for an extended period of time. Therefore, alternate methods will be used to perform SDM verification, including comparing current boron concentration to a previously determined boron concentration that bounds a conservative set of plant parameters.

It is recognized that JFD 10 of this section was not worded well. Therefore, JFD 10 of this section has been reworded to provide a clearer justification for deviation from this STS Bases attribute (see attached markups).

<u>NRC Question 3.1.4-1:</u>

3.1.4-1 ITS B3.1.3, MTC
STS B3.1.4, MTC
STS SR 3.1.4.2 and SR 3.1.4.3, lower MTC limit
JFD 2

JFD 2 states that, "Licensee control of the negative limit is adequate since the negative limit is verified within the reload safety analysis, and is indirectly monitored through periodic reactivity anomaly checks. The negative design limit will be controlled in accordance with the 10 CFR 50.59 process. Therefore, the level of safety will be maintained. As such the lower limit is considered non-bounding ...". Neither the CTS nor the ITS contain lower MTC limits.

Comment: While the lower MTC limit is verified within the reload analysis and monitored through anomaly checks, it is not clear why the lower limit is "non-bounding;" why isn't it in the ITS. Since the negative limit is controlled in accordance with the 10 CFR 50.59 process, is its presentation maintained current in the FSAR?

WE Response:
WE has reviewed this question and the JFD and has determined that justification for the proposed ITS change is adequate and that adoption of the STS lower MTC limit condition and surveillance requirement is not necessary.

The negative MTC limit is not maintained in the FSAR, it is verified within each core reload safety analysis, and is indirectly monitored through periodic reactivity anomaly checks. It is recognized that JFD 02 of this section was not worded well. Therefore, JFD 02 of this section

has been reworded to provide a clearer justification for deviation from this STS attribute (see attached markups).

NRC Question 3.1.4-2:

3.1.4-2 ITS B3.1.3, MTC
STS B3.1.4, MTC
JFD 7

JFD 7 addresses changes to MTC BOC Bases LCO and SR sections. Comment: The JFD 7 write-up is missing.

WE Response:

JFD 7 was inadvertently omitted from the original submittal. JFD 7 is in the ITS database but did not print out because of an error in the cross-reference table. This error has been corrected and JFD 07 is provided for NRC review (see attached markups).

NRC Question 3.1.5-1:

3.1.5-1 ITS 3.1.4 Rod Group Alignment Limits, Applicability
STS 3.1.5 Rod Group Alignment Limits, Applicability
CTS 15.3.10
JFD 9
DOC A6

ITS Applicability is Modes 1 and 2 with $K_{eff} \geq 1$, similar to CTS Applicability which requires control rod operability during power and low power operation. STS Applicability is Modes 1 and 2. The JFD 9 justification is that, "Rod group alignment limits are established to maintain acceptable power distribution (which)... is only of concern when the reactor is at power ($K_{eff} \geq 1$) which is consistent with the Mode of Applicability specified for the proposed Shutdown Bank Insertion Limits."

Comment: Rod alignment is relevant during start-up and not just at power after $K_{eff} \geq 1$. Insertion limits are applicable at power after $K_{eff} \geq 1$, and not before. In addition, it is appropriate that if rods are misaligned/inoperable they be fully inserted, and not just to a point at which $K_{eff} < 1$. Recommend that the STS Applicability be adopted.

WE Response:

WE has reviewed this question and the JFD and DOC and has determined that the NRC recommendation to adopt the STS applicability should be implemented.

The initial intent of the proposed alteration was to align the ITS with the CTS; however, it has been determined to be unnecessary since Operations already monitors these parameters during approach to criticality while starting up the reactor.

As a result of this change, JFD 09 of this section is no longer required and was changed to "not used," DOC A.06 of this section was revised, new DOC M.07 was created since this change is now more restrictive, the CTS markup, the NUREG markup and proposed ITS and associated Bases was revised to adopt the STS applicability (see attached markups).

Additional Corrections Required to ITS Section 3.1.4:

Additional corrections to the conversion package for ITS Section 3.1.4 have been identified as a result of ITS reviews and TSCR submittals and have been included in this RAI response for completeness.

TSCR 202 was approved by the NRC under License Amendment 195/200 on March 22, 2000. The Amendment changed the control rod exercise surveillance interval in CTS Table 15.4.1-2, item 10, from "every 2 weeks" to "quarterly." As a result, the CTS markup for LCO 3.01.04 was changed, the DOC L.04 for this item was changed to "not used," new DOC A.16 was created, and the NSHC for the old L.04 was changed to "not used" (see attached markups).

<u>NRC Question 3.1.6-1:</u>

3.1.6-1 ITS 3.1.5, Shutdown Bank Insertion Limits, Applicability
STS 3.1.6, Shutdown Bank Insertion Limits, Applicability
CTS 15.3.10.D.1
DOC A2
JFD 4

The ITS specifies an applicability of Modes 1 and 2 with $K_{eff} \geq 1$, similar to CTS Applicability which requires that shutdown banks must be fully withdrawn whenever the reactor is critical. STS Applicability is Modes 1 and 2 because the shutdown banks must be fully withdrawn prior to the control banks being withdrawn on an approach to criticality.

Comment: The STS applicability is the most appropriate and the easiest to comply with operationally. Recommend adopting the STS applicability.

WE Response:

WE has reviewed this question and the JFD and DOC and has determined that the NRC recommendation to adopt the STS applicability should be implemented.

The initial intent of the proposed alteration was to align the ITS with the CTS; however, it has been determined to be unnecessary since Operations already monitors these parameters during approach to criticality while starting up the reactor.

As a result of this change, JFD 04 of this section is no longer required and was changed to "not used," DOC A.02 of this section was changed to not used, new DOC M.01 was created since this change is now more restrictive, the CTS markup, the NUREG markup, and the proposed ITS and associated Bases was revised to adopt the STS applicability (see attached markups).

NRC Question 3.1.7-1:

3.1.7-1 ITS 3.1.6, Control Bank Insertion Limits
STS 3.1.7, Control Bank Insertion Limits
ITS SR 3.1.6.3
CTS 15.3.10.F/D
DOC LA2
JFD 8

The CTS defines fully withdrawn control rods as being ≥ 225 steps. The ITS SR 3.1.6.3 states, "...control banks not withdrawn from the core as specified in the COLR," deleting the STS word "fully" before "withdrawn."

Comment: By deleting the word "fully," the meaning of the SR is changed such that the verification need not be performed on some control rods withdrawn <225 steps that are assumed to satisfy correct sequence and overlap limits. Recommend retaining the word "fully."

WE Response:

WE has reviewed this question and the JFD and DOC and has determined that the NRC recommendation should be implemented.

Therefore, the word "fully" will be adopted and "as specified in the COLR" will not be added to the proposed ITS at the end of SR 3.1.6.3. Fully withdrawn is clearly defined in the COLR as a bank demand position ≥ 225 steps. As a result of this change, JFD 08 was slightly modified (still valid for Bases 3.1.6 addition under SR 3.1.6.3) and the NUREG markup and proposed ITS was changed as necessary (see attached markups).

NRC Question 3.1.8-1:

3.1.8-1 ITS 3.1.7, Rod Position Indication, Required Actions
STS 3.1.8, Rod Position Indication, Required Actions
CTS 15.3.10.B/C
DOC L2
JFD 7
JFD 11
JFD 13

The CTS requires control rods with inoperable individual position indicators or demand indicators have their positions checked, or the rod would be declared misaligned. If multiple indicators were inoperable or if the position checks could not be performed, the CTS would ultimately require the unit be shutdown. While the STS allows continued operation at ≤ 50% RTP with one RPI inoperable per group or one demand indicator inoperable per group, it would require unit shutdown with more than one RPI inoperable per group or more than one demand indicator inoperable per group.

Comment: The ITS allows continued operation at ≤ 50% RTP with more than one RPI inoperable per group or more than one demand indicator inoperable per group, that neither the CTS nor the STS permits. Recommend adopting the STS wording and structure with regard to inoperable RPIs per group and inoperable demand indicators per group, the required actions to reduce power to ≤ 50% RTP, and the default condition to be in Mode 3.

WE Response:

WE has reviewed this question and the JFD's and DOC and has determined that partial adoption of the NRC recommendations should be implemented. Specifically, PBNP will adopt the STS required actions of being in Mode 3 (D.1) if the associated required actions and completion times are not met.

Discussion:

There is currently no limit in the CTS on the number of inoperable RPIs and bank demand position indicators as long as the action statements of CTS LCO 15.3.10.C are met. If the actions are not met, then the actions for misaligned rod must be taken (CTS 15.3.10.B.1.b). If it is determined that more than one rod is misaligned and the required actions of CTS 15.3.10.B.1.b are not met, the reactor must be shutdown.

Therefore, the proposed ITS will be revised to be consistent with the CTS in requiring the unit to be shutdown if the required actions and associated completion times of LCO 3.1.7 are not met. As a result, the CTS markups were revised, DOC L.02 and NSHC L.02 are no longer required and were changed to "not used," new DOC A.06 was created, JFD 07 is no longer required and

was changed to "not used," the NUREG markup and the associated Bases were revised, and the ITS and associated Bases were revised (see attached markups).

NRC Question 3.1.8-2:

3.1.8-2 ITS 3.1.7, Rod Position Indication, Applicability
STS 3.1.8, Rod Position Indication, Applicability
CTS 15.3.10.B/C
DOC M2
DOC M3
JFD 10

The CTS Mode of Applicability is $\geq$ 10% RTP. The ITS Mode of Applicability is Modes 1 and 2 with $K_{eff} \geq 1$. The STS Mode of Applicability is Modes 1 and 2.

Comment: The STS applicability is the most appropriate (RPI Applicability is essential during an approach to criticality) and the STS applicability is easiest to comply with operationally. Recommend adopting the STS applicability.

WE Response:

WE has reviewed this question and the JFD and DOCs and has determined that the NRC recommendation to adopt the STS applicability should be implemented.

The initial intent of the proposed alteration was to align the ITS with the CTS; however, it has been determined to be unnecessary since Operations already monitors these parameters during approach to criticality while starting up the reactor.

As a result of this change, JFD 10 of this section is no longer required and was changed to "not used," DOC M.02 of this section was revised, the CTS markup, the NUREG markup, and the proposed ITS and associated Bases was revised to adopt the STS applicability (see attached markups).

In addition, during review of this section's Bases, it was determined that a slight addition to the Bases of ITS 3.1.7 was necessary for clarification of the effectiveness of incore moveable detector instrumentation in determining rod position. Therefore, Insert B 3.1.8-5 in the Bases of ITS 3.1.7, Action A.1 was slightly modified (see attached markups).

NRC Question 3.1.10-1:

3.1.10-1 Bases for ITS 3.1.8, PHYSICS TESTS Exceptions - MODE 2, Background
Bases for STS 3.1.10, PHYSICS TESTS Exceptions - MODE 2, Background
JFD 4

The STS PHYSICS TESTS Exceptions - MODE 2 Bases Background section state that "...test results are approved prior to continued power escalation...". The ITS PHYSICS TESTS Exceptions - MODE 2 Bases Background section is changed to state that "...test results are independently verified prior to continued power escalation...".

Comment: "Approval" connotes not only verification, but also some form of management acceptance. Do Point Beach procedures not require approval of test results?

WE Response:

WE has reviewed this question and the JFD and has determined that the NRC recommendation to adopt the STS wording should be implemented. As a result of this change, JFD 4 of this section was revised (still applicable for other deviations), the NUREG markup, and the proposed ITS Bases was revised to adopt the STS wording (see attached markups).

<u>NRC Question 3.2.4-1:</u>

3.2.4-1 ITS B3.2.4 Bases for SR 3.2.4.2
STS B3.2.4 Bases for SR 3.2.4.2
JFD 3

The ITS deletes STS information on incore detector monitoring using thimbles that is not applicable to Point Beach.

Comment: Why not include a brief discussion of the monitoring method(s) used at Point Beach?

WE Response:

WE has reviewed this question and the JFD and has determined that the NRC recommendation should be implemented. As a result, JFD 3 of this section and the Bases of the NUREG markup and proposed ITS was changed to reflect the attributes of the incore detector monitoring methods used at PBNP (see attached markups).

Additional Corrections Required to ITS Section 3.2.1:

Additional corrections to the conversion package for this section (i.e. 3.2) have been identified as a result of ITS reviews and TSCR submittals and have been included in this RAI response for completeness.

Changes have been indicated in the CTS markup for this section as a result of NRC approval of Amendment 193/198, dated 2/08/00 (related to incorporation of new fuel design). As a result of

incorporating this Amendment, DOC LA.01 of this section was also revised (see attached markups).

NRC Question 3.5.1-1:

3.5.1-1 DOC M.4
ITS SR 3.5.1.4
STS SR 3.5.1.4
CTS Table 4.1-3, Item 6 monthly sampling test
JFD 16

In the submittal, ITS SR Frequency following solution volume increase is 24 hours; the STS specifies 6 hours. The submittal in JFD 16 does not adequately justify the difference.

Comment: Adopt STS Frequency.

WE Response:

WE has reviewed this question and the JFD and DOC and has determined that justification for the proposed ITS is adequate and that adoption of the STS surveillance frequency is not feasible or necessary. It is recognized that JFD 16 of this section was not worded well. Therefore, JFD 16 of this section has been reworded to provide a clearer justification for deviation from this STS attribute (see below discussion and attached markups).

Discussion:

The STS frequency of 6 hours for obtaining a accumulator boron concentration sample was based on a Westinghouse 4 loop reactor design that has line taps outside containment to obtain accumulator boron samples. However, PBNP does not have this design attribute. A containment entry must be made at PBNP to obtain the necessary accumulator boron samples. Therefore, the additional time is necessary to allow coordination between Radiation Protection, Operations, and Chemistry for the containment entry evolution to obtain samples and perform boron titration's. If an unplanned volume addition was required during a back shift, additional personnel could conceivably be required to be called in to support this evolution, thereby necessitating additional time. This time frame is also acceptable based on rationale presented for restoring accumulators boron concentration to within limits (72 hours), as contain in Required Action A.1 of LCO 3.5.1 of NUREG 1431.

It should be noted that being in this surveillance requirement would be a rare occurrence at PBNP, since accumulator make up is almost always performed from the RWST (surveillance would not be applicable within these conditions).

Additional Corrections Required to ITS Section 3.5:

Additional corrections to the conversion package for ITS Section 3.5 have been identified as a result of ITS reviews by plant staff.

CTS 15.4.5.II.A.02 gives performance criteria for the SI , RHR, and CS pumps and states that "Acceptable levels of performance shall be that the pumps start, reach their required head at, and operate for at least 15 minutes on the full flow test lines." The DOC for this item was classified as an LA in section 3.5.2 (LA.3) and section 3.5.3 (LA.2) in the original submittal. This classification was incorrect and should have been classified as an L, because the performance criteria will not be retained in the proposed ITS. As a result of this change, the CTS markups for sections 3.5.2 and 3.5.3 have been changed, DOC LA.3 (in section 3.5.2) has been changed to "not used" and new DOC L.2 and new NSHC L.2 was created, and DOC LA.2 (in section 3.5.3) has been changed to "not used" and new DOC L.2 and new NSHC L.2 was created (see attached markups).

CTS 15.4.5.I.A.1.a specifies the conditions and manner in which ECCS testing must be conducted (RCS pressure less than or equal to 350 psig and RCS temperature less than or equal to 350 degrees, pump breakers may be placed into the "test" position), while CTS 15.4.5.I.A.2 specifies the manner in which equipment operation is to be verified (control board indication and visual indications). Similarly, the DOC for these items was classified as an LA in section 3.5.2 (LA.2) and section 3.5.3 (LA.3) in the original submittal. This classification was incorrect and should have been classified as an L, because the performance criteria will not be retained in the proposed ITS. As a result of this change, the CTS markups for sections 3.5.2 and 3.5.3 have been changed, DOC LA.2 (in section 3.5.2) has been changed to "not used" and new DOC L.3 and new NSHC L.3 was created, and DOC LA.3 (in section 3.5.3) has been changed to "not used" and new DOC L.3 and new NSHC L.3 was created (see attached markups).

Proposed ITS SR 3.5.1.2 states "Verify borated water volume in each accumulator is ≥15% and ≤40%." It has been determined that accumulator levels in the proposed ITS will be identified by volumes, consistent with CTS 15.3.3.A.b (i.e. $\geq$ 1100 ft^3 and $\leq$ 1136 ft^3). As a result of this correction JFD 08 was revised and the NUREG markup for section 3.5.1 and proposed ITS 3.5.1 was changed to reflect the values indicated in volumes (see attached markups).

ATTACHMENT 2
DISCARD AND INSERTION INSTRUCTIONS

VOLUME 1	
SECTION 2.0	
DISCARD	**INSERT**
DOC pages 1 of 5 through 5 of 5	DOC pages 1 of 5 through 5 of 5
CTS markup pages 1 of 7 through 7 of 7	CTS markup pages 1 of 8 through 8 of 8
JFD pages 1 of 2 through 2 of 2	JFD pages 1 of 2 through 2 of 2
ISTS markup page 2.0-1, 2.0-2	ISTS markup page 2.0-1, 2.0-2, Insert 2.0-1
ISTS Bases markup pages B2.0-2, B2.0-3, B 2.0-5, B2.0-6, Insert A, Insert B	ISTS Bases markup pages B2.0-2, B2.0-3, B 2.0-5, B2.0-6, Insert A
ITS page 2.0-1, and 2.0-2	ITS page 2.0-1
ITS Bases pages B2.0-1 through B2.0-10	ITS Bases pages B2.1.1-1 through B2.1.1-4, B2.1.2-1 through 2.1.2-3
VOLUME 2	
SECTION 3.1.1	
DISCARD	**INSERT**
JFD page 3 of 3	JFD page 3 of 3
SECTION 3.1.4	
DISCARD	**INSERT**
JFD page 2 of 4, 4 of 4	JFD page 2 of 4, 4 of 4
SECTION 3.1.5	
DISCARD	**INSERT**
DOC pages 1 of 13 through 13 of 13	DOC pages 1 of 13 through 13 of 13
CTS markup page 1 of 13, 2 of 13, 7 of 13	CTS markup page 1 of 13, 2 of 13, 7 of 13
JFD pages 1 of 5 through 5 of 5	JFD pages 1 of 4 through 4 of 4
ISTS markup page 3.1-8 through 3.1-10	ISTS markup page 3.1-8 through 3.1-10
ISTS Bases markup pages B3.1.5-5, B3.1.5-6, B3.1.5-8, B3.1.5-9	ISTS Bases markup pages B3.1.5-5, B3.1.5-6, B3.1.5-8, B3.1.5-9
NSHC pages 1 of 10 through 10 of 10	NSHC pages 1 of 9 through 9 of 9

ATTACHMENT 2
DISCARD AND INSERTION INSTRUCTIONS

SECTION 3.1.5 (continued)	
DISCARD	**INSERT**
ITS pages 3.1.4-1 through 3.1.4-4	ITS pages 3.1.4-1 through 3.1.4-4
ITS Bases pages B3.1.4-1 through B3.1.4-11	ITS Bases pages B3.1.4-1 through B3.1.4-8
SECTION 3.1.6	
DISCARD	**INSERT**
DOC pages 1 of 6 through 6 of 6	DOC pages 1 of 7 through 7 of 7
CTS markup pages 2 of 9, 6 of 9, 7 of 9	CTS markup pages 2 of 9, 6 of 9, 7 of 9
JFD pages 1 of 2, 2 of 2	JFD pages 1 of 2, 2 of 2
ISTS markup page 3.1-12	ISTS markup page 3.1-12
ISTS Bases markup pages B3.1.6-2, B3.1.6-3, B3.1.6-4, B3.1.6-5	ISTS Bases markup pages B3.1.6-2, B3.1.6-3, B3.1.6-4, B3.1.6-5
NSHC pages 1 of 5 through 5 of 5	NSHC pages 1 of 6 through 6 of 6
ITS pages 3.1.5-1, 3.1.5-2	ITS pages 3.1.5-1, 3.1.5-2
ITS Bases pages B3.1.5-1 through B3.1.5-5	ITS Bases pages B3.1.5-1 through B3.1.5-4
SECTION 3.1.7	
DISCARD	**INSERT**
JFD pages 1 of 3 through 3 of 3	JFD pages 1 of 3 through 3 of 3
ISTS markup page 3.1-16	ISTS markup page 3.1-16
ITS page 3.1.6-3	ITS page 3.1.6-3
SECTION 3.1.8	
DISCARD	**INSERT**
DOC pages 1 of 5 through 5 of 5	DOC pages 1 of 5 through 5 of 5
CTS markup pages 3 of 8 and 4 of 8	CTS markup pages 3 of 8 and 4 of 8
JFD pages 1 of 9 through 9 of 9	JFD pages 1 of 8 through 8 of 8
ISTS markup page 3.1-17, 3.1-18	ISTS markup page 3.1-17, 3.1-18

ATTACHMENT 2
DISCARD AND INSERTION INSTRUCTIONS

SECTION 3.1.8 (continued)	
DISCARD	**INSERT**
ISTS Bases markup pages B3.1.8-4 through B3.1.8-6, Insert B3.1.8-5	ISTS Bases markup pages B3.1.8-4 through B3.1.8-6, Insert B3.1.8-5
NSHC pages 1 of 5 through 5 of 5	NSHC pages 1 of 4 through 4 of 4
ITS page 3.1.7-1 through 3.1.7-3	ITS page 3.1.7-1 through 3.1.7-3
ITS Bases pages B3.1.7-1 through B3.1.7-6	ITS Bases pages B3.1.7-1 through B3.1.7-5
SECTION 3.1.10	
DISCARD	**INSERT**
JFD pages 1 of 2 through 2 of 2	JFD pages 1 of 2 through 2 of 2
ISTS Bases markup page B3.1.10-2	ISTS Bases markup page B3.1.10-2
ITS Bases pages B3.1.8-1 through B3.1.8-8	ITS Bases pages B3.1.8-1 through B3.1.8-8
SECTION 3.2.1	
DISCARD	**INSERT**
DOC page 7 of 9	DOC page 7 of 9
CTS markup pages 1 of 11 through 11 of 11	CTS markup pages 1 of 13 through 13 of 13
SECTION 3.2.4	
DISCARD	**INSERT**
JFD pages 1 of 2, 2 of 2	JFD pages 1 of 2, 2 of 2
ISTS Bases markup page B3.2.4-6, Insert B3.2.4-3	ISTS Bases markup page B3.2.4-6, Insert B3.2.4-3
ITS Bases pages B3.2.4-1 through B3.2.4-7	ITS Bases pages B3.2.4-1 through B3.2.4-6
SECTION 3.5.1	
DISCARD	**INSERT**
JFD page 3 of 7 and 7 of 7	JFD page 3 of 7 and 7 of 7
ISTS markup page 3.5-2	ISTS markup page 3.5-2
ITS pages 3.5.1-1 and 3.5.1-2	ITS pages 3.5.1-1 and 3.5.1-2

ATTACHMENT 2
DISCARD AND INSERTION INSTRUCTIONS

SECTION 3.5.2	
DISCARD	**INSERT**
DOC pages 1 of 12 through 12 of 12	DOC pages 1 of 13 through 13 of 13
CTS markup pages 18 of 22 and 20 of 22	CTS markup pages 18 of 22 and 20 of 22
NSHC pages 1 of 6 through 6 of 6	NSHC pages 1 of 7 through 7 of 7
SECTION 3.5.3	
DISCARD	**INSERT**
DOC pages 1 of 9 through 9 of 9	DOC pages 1 of 9 through 9 of 9
CTS markup pages 9 of 12 and 11 of 12	CTS markup pages 9 of 12 and 11 of 12
NSHC pages 1 of 4 through 4 of 4	NSHC pages 1 of 6 through 6 of 6

ENCLOSURE

Description of Changes - NUREG-1431 Section 2.0

24-May-00

DOC Number	DOC Text
A.01 Rev. A	In the conversion of Point Beach current Technical Specifications (CTS) to the proposed plant specific Improved Technical Specifications (ITS), certain wording preferences or conventions are adopted which do not result in technical changes (either actual or interpretational). Editorial changes, reformatting, and revised numbering are adopted to make the ITS consistent with the Standard Technical Specifications, Westinghouse Plants, NUREG-1431, Revision 1 (i.e., Improved Standard Technical Specifications (ISTS)).

CTS:	ITS:
15.02.01.01	2.01.01
15.02.02	2.01.02
15.02.02 APPL	2.01.02

DOC Number	DOC Text
A.02 Rev. B	Not Used

CTS:	ITS:
N/A	N/A

DOC Number	DOC Text
A.03 Rev. A	The Bases of the current Technical Specifications for this section have been completely replaced by revised Bases that reflect the format and applicable content of PBNP ITS, consistent with the Standard Technical Specifications for Westinghouse Plants, NUREG-1431. The revised Bases are as shown in the PBNP ITS Bases.

CTS:	ITS:
BASES	B 2.0

DOC Number	DOC Text
A.04 Rev. A	The CTS RCS Pressure Safety Limit states that it is applicable when there is fuel in the reactor vessel. The definition of Mode in the ITS requires there to be fuel in the reactor vessel to be in a specified Mode. Accordingly, the change from "with fuel assemblies installed in the reactor vessel" to "Modes 1, 2, 3, 4, 5, and 6" is an administrative change.

CTS:	ITS:
15.02.02	DELETED

DOC Number	DOC Text
A.05 Rev. A	The CTS provides an introductory statement (Objective) at the beginning of each Safety Limits section which provides a brief summary of the purpose for this Section. This information does not establish any regulatory requirements, since this information does not establish any regulatory requirement, this summary information is not required to provide adequate protection of public health and safety. Deletion of this information does not alter any requirement set forth in the Technical Specifications. The ITS equivalent of this information is contained in the Bases of each Specification, which provides a detailed explanation of the objective for each Specification.

CTS:	ITS:
15.02.01 OBJ	DELETED
15.02.02 OBJ	DELETED

Description of Changes - NUREG-1431 Section 2.0

24-May-00

DOC Number	DOC Text
L.01 Rev. A	The CTS Applicability of the Reactor Core Safety Limits is "during Operation" The proposed ITS Core Safety Limits for Point Beach have proposed an Applicability of Modes 1 and 2. The Core Safety Limits are imposed to restrict operation such that overheating and damage of the fuel will not occur. Damage from this mechanism is only probable during power operations (Modes 1 and 2).

CTS:	ITS:
15.02.01 APPL	2.01.01

LA.01 Rev. A	CTS 15.6.7, administrative actions to be taken if a Safety Limit is violated have been deleted from the Technical Specifications. Specification 15.6.7.B details company internal reporting requirements. These reporting requirements can be deleted from the Technical Specifications and controlled via licensee controlled mechanisms with no impact on safety. The Quality Assurance Manual provides details on internal reporting and review requirements. This document is maintained in accordance with 10 CFR 50.54. Changes to these requirements will be processed accordingly. Specification 15.6.7.C details reporting requirements which are duplicative of those required under 10 CFR50.72 and 50.73. Designation of the individual responsible for providing this information is adequately controlled by plant practices and processes. There is no safety reduction in removing detail from the Technical Specifications relative to the individual who must provide information to the NRC, as regulation 10CFR 50.72 and 50.73 will still require this information to be provided. Specification 15.6.7.D contains a requirement to have the Safety Limit violation report reviewed by the Chief Nuclear Officer and Chairman of the Off-Site Review Committee. This requirement will be deleted from the Technical Specifications. The ITS has added explicit Actions to the Technical Specifications requiring restoration of compliance with any Safety Limits violation, placing the plant in a safe and stable condition. The CTS requirement dictating personnel required to review the Safety Limits violation report has no direct bearing on plant safety and accordingly, can be deleted form the Technical Specifications and controlled by the licensee. The ITS has added explicit Actions to the Technical Specifications requiring restoration of compliance with any Safety Limits violation, placing the plant in a safe and stable condition. The Quality Assurance Manual provides details on internal reporting and review requirements. This document is maintained in accordance with 10 CFR 50.54. Changes to these requirements will be processed accordingly.

CTS:	ITS:
15.06.07.B	DELETED
15.06.07.C	DELETED
15.06.07.D	DELETED

Description of Changes - NUREG-1431 Section 2.0

24-May-00

DOC Number	DOC Text
LA.02 Rev. B	The Reactor Core Safety Limit Curves (CTS Figure 15.02.01-01 and 15.02.01-02) have been relocated to the Core Operating Limits Report (COLR) under licensee control. This is consistent with Approved TSTF-339, rev. 1, which relocated this figure out of the STS and into the COLR to be in accordance with the approved version of WCAP-14483-A "Generic Methodology for Expanded Core Operating Limits Report." The figures represent core limits on RCS temperature conditions as a function of pressurizer pressure and fractional rated power. These limits can be relocated with no impact on safety. The limits defined by the curve alert the licensee of a potential violation of a core safety limit. Additional evaluation will be required to determine if an actual safety limit (DNBR and fuel centerline melt design basis limits), which are included in the proposed ITS, has been violated. Therefore, there is no reduction in a level of safety by relocating the curves to the COLR as controls are still in place to define and ensure appropriate action is taken in the event of a violation of a safety limit. The Reactor Core Safety Limit curves are being replaced by the specific safety limits protected by the curves. These are the maximum fuel centerline temperature limit which prevents melting of the fuel and the DNBR correlation limits to protect against a departure of nucleate boiling. These are the actual safety limits and thus meet the intent of the CTS to protect against violation of the safety limits. As indicated above, the curves will be maintained under licensee control in the COLR and will continue to be used to alert the licensee to a potential violation of the defined safety limits. This change is less restrictive, since the curves are being relocated out of the Technical Specifications and into the COLR, which is under licensee control.

CTS:	ITS:
15.02.01.01 F 15.02.01-1	COLR
15.02.01.01 F 15.02.01-2	COLR
NEW	2.01.01.01

DOC Number	DOC Text
LB.01 Rev. A	Several administrative actions stated in CTS 15.6.7 for violation of a Safety Limit are duplicative of regulations, and are therefore unnecessary in the Technical Specifications. Specification 15.6.7.A requiring that the unit remain shutdown until NRC approval is received and Specification 15.6.7.D, detailing Safety Limit violation report content are duplicative of 10CFR 50.36 and 10CFR 50.73 respectively. Accordingly, these items are adequately controlled through regulations and do not have to be repeated in the Technical Specifications. Specification 15.6.7.C details reporting requirements which are duplicative of those required under 10 CFR50.72 and 50.73. These details are adequately controlled through regulations and do not have to be repeated in the Technical Specifications. There is no safety reduction in removing these details from the Technical Specifications, as regulation 10CFR 50.72 and 50.73 will still require this information to be provided. Specification 15.6.7.D contains a requirement to provide a Safety Limit violation report to the NRC within 10 days of the occurrence. This time frame is not sufficient to perform a detailed review of the event, the circumstances, effects of the occurrence on the plant, and corrective actions. Removing this requirement from the Technical Specifications will allow extension of the time frame for a written report (LER) to 30 days consistent with 10CFR 50.73. This change is acceptable based on the need to perform a detailed review and investigation as well as the fact that the ITS has added explicit Actions to the Technical Specifications requiring restoration of compliance with Safety Limits after any violation. Therefore, extending the time frame for a written report has no impact on safety, as the plant has been placed into a safe and stable condition. **CTS:** / **ITS:** 15.06.07.A — DELETED 15.06.07.C — DELETED 15.06.07.D — DELETED
M.01 Rev. A	The CTS defines exceeding the Core Safety Limit as when the combination of reactor coolant system average temperature and power level is above the appropriate pressure line. The proposed ITS for Point Beach defines violation of the Core Safety Limit based on "highest loop average" temperature versus the "average" temperature. This change is more restrictive than the existing Technical Specifications which would allow averaging of both RCS loops together. By defining the limit as being the highest loop average temperature, the most limiting loop is utilized in determining compliance with the Safety Limit, which represents the loop closest to approaching saturation temperature or the core exit quality limit assumed in the DNBR correlations. **CTS:** / **ITS:** 15.02.01.01 — 2.01.01

DOC Number	DOC Text
M.02 Rev. A	CTS requires the reactor to be shutdown (CTS 15.6.7.A), in the event of either Safety Limit (i.e. Core and RCS Pressure) being violated, but does not contain any explicit time limit for the shutdown. In addition, requiring a reactor shutdown for exceeding the RCS Pressure Safety Limit, is ambiguous when the reactor is not critical. The proposed ITS contains an explicit time limit of one hour to place the unit in Mode 3 (reactor shutdown) and to restore compliance in the event of a Safety Limit violation in Modes 1 and 2. One hour to restore compliance and to place the unit in Mode 3 is a reasonable amount of time to reduce RCS temperature, core power, or adjust RCS pressure (as applicable) to restore compliance with the Core or RCS Pressure Safety Limit, and to complete a plant shutdown, giving consideration to the severity of a Safety Limit violation. In Modes other than 1 and 2, the only Safety Limit which is applicable is the RCS pressure limit, which has been given an Action to restore compliance within 5 minutes. Restoration of RCS pressure within 5 minutes when the reactor is not critical is deemed to be adequate based on restoration actions necessary and the significance of a Safety Limit violation in this condition. The addition of this completion time and restoration actions are more restrictive requirements, imposed for consistency with NUREG 1431.

CTS:	ITS:
15.06.07.A	2.02.01 2.02.02
NEW	2.02.02.01 2.02.02.02

15.2.0 SAFETY LIMITS AND LIMITING SAFETY SYSTEM SETTINGS

15.2.1 SAFETY LIMIT, REACTOR CORE

L.01

Applicability:

Applies to the limiting combinations of thermal power, reactor coolant system pressure, and coolant temperature during operation.

A.05

~~Objective:~~

~~To maintain the integrity of the fuel cladding.~~

~~Specification:~~

LA.02

Insert 1

A.01

Reactor

pressurizer

1. The combination of thermal power level, ~~coolant~~ pressure, and coolant temperature shall not exceed the limits ~~shown in Figure 15.2.1-1 or Figure 15.2.1-2 as applicable for Units 1 and 2.~~ The safety limit is exceeded if the point defined by the combination of reactor coolant system average temperature and power level is at any time above the appropriate pressure line.

B COLR

Specification 2.1.1

A.01

highest loop

M.01

Basis:

The restrictions of this safety limit prevent overheating of the fuel and possible cladding perforation which would result in the release of fission products to the reactor coolant. Overheating of the fuel cladding is prevented by restricting fuel operation to within the nucleate boiling regime where the heat transfer coefficient is large and the cladding surface temperature is slightly above the coolant saturation temperature.

Operation above the upper boundary of the nucleate boiling regime could result in excess cladding temperature because of the onset of departure from nucleate boiling (DNB) and the resultant sharp reduction in heat transfer coefficient. DNB is not a directly measurable parameter during operation and therefore thermal power and Reactor Coolant temperature and pressure have been related to DNB.

A.03

This relation has been developed to predict the DNB flux and the location of DNB for axially uniform and non-uniform heat flux distributions. The local DNB heat flux ratio, DNBR, defined as the ratio of the heat flux that would cause DNB at a particular core location to the local heat flux, is indicative of the margin to DNB.

The DNB design basis is as follows: there must be at least a 95 percent probability at a 95 percent confidence level that DNB will not occur during steady state operation, normal operational transients, and anticipated transients and is an appropriate margin to DNB for all operating conditions.

The family of curves in Figure 15.2.1-1 is applicable to a core with any combination of 14 x 14 OFA and 14 x 14 upgraded OFA fuel assemblies. The family of curves in Figure 15.2.1-2 is applicable to any combination of 422V+ fuel assemblies, burned 14 x 14 OFA fuel assemblies, and burned 14 x 14 Upgraded OFA fuel assemblies, or a full core of 422V+ fuel assemblies. The use of these assemblies is justified by a cycle-specific reload analysis. The WRB-1 correlation is used to generate these curves. Uncertainties in plant parameters and DNB correlation predictions are statistically convoluted to obtain a DNBR uncertainty factor. This DNBR uncertainty factor establishes a value of design limit DNBR. This value of design limit DNBR is shown to be met in plant safety analyses, using values of input parameters considered at their nominal values.

B Amendme 193/198

These curves represent the loci of points of thermal power, Reactor Coolant System pressure and average temperature for which the calculated DNBR is no less than the design limit-DNBR or the average enthalpy at the vessel exit is less than the enthalpy of saturated liquid. Appropriate rod bow penalties have been included in the generation of these curves. The effects of fuel densification and possible clad flattening have also been taken into account.

An allowance is included in these curves for an increase in $F^N_{\Delta H}$ at reduced power based on the expression:

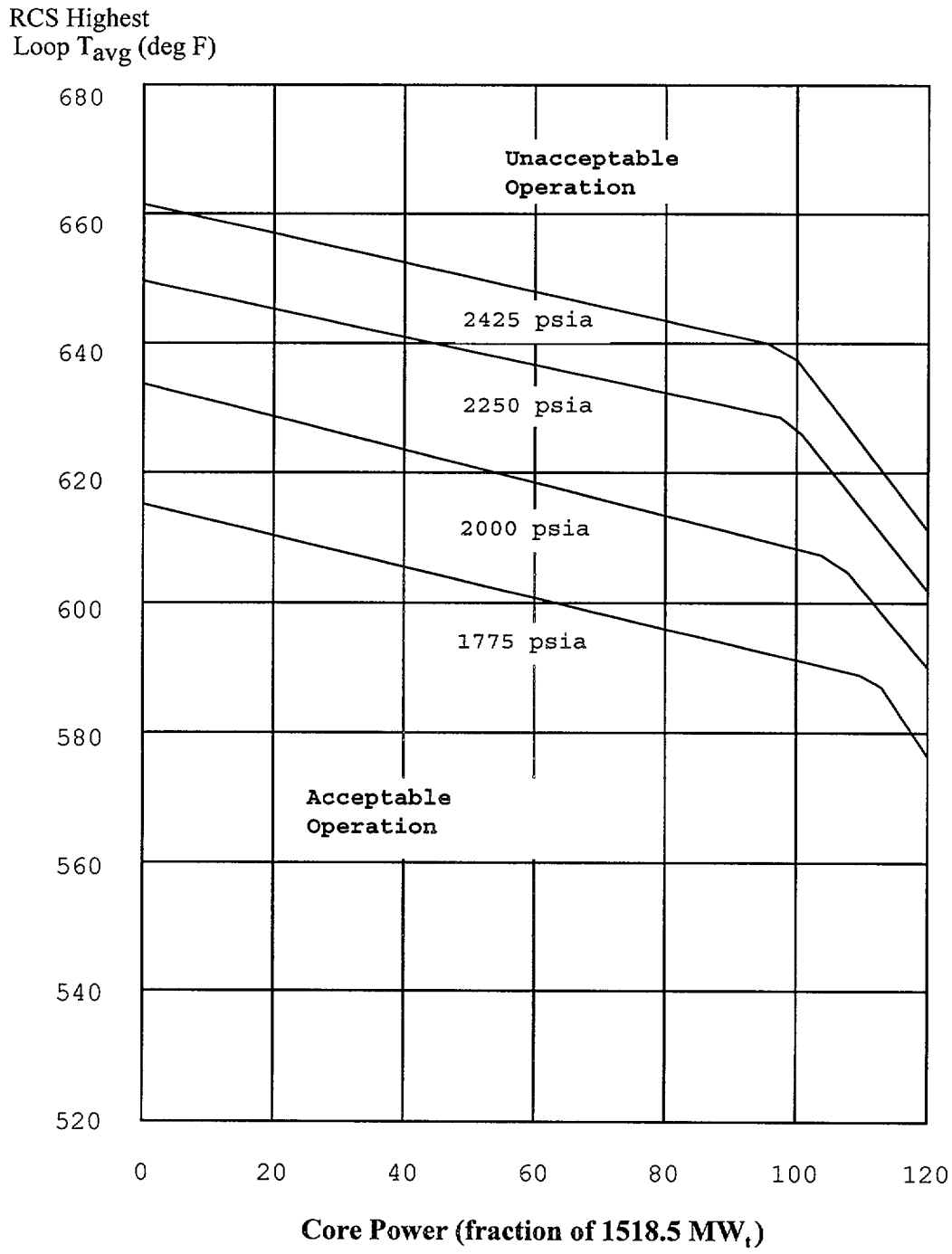
$$F^N_{\Delta H} = \text{Full Power } F^N_{\Delta H} [1 + 0.3 (1 - P)]$$

where P is a fraction of rated thermal power.

The hot channel factors are sufficiently large to account for the degree of malpositioning of full-length rods that is allowed before the reactor trip setpoints are reduced and rod withdrawal block and load runback may be required. Rod withdrawal block and load runback occur before reactor trip setpoints are reached. The Reactor Control and Protective System is designed to prevent any anticipated combination of transient conditions that would result in a DNB ratio of less than the design limit DNBR.

LA.02

Figure 15.2.1-1*
POINT BEACH NUCLEAR PLANT UNITS 1 AND 2
REACTOR CORE SAFETY LIMITS

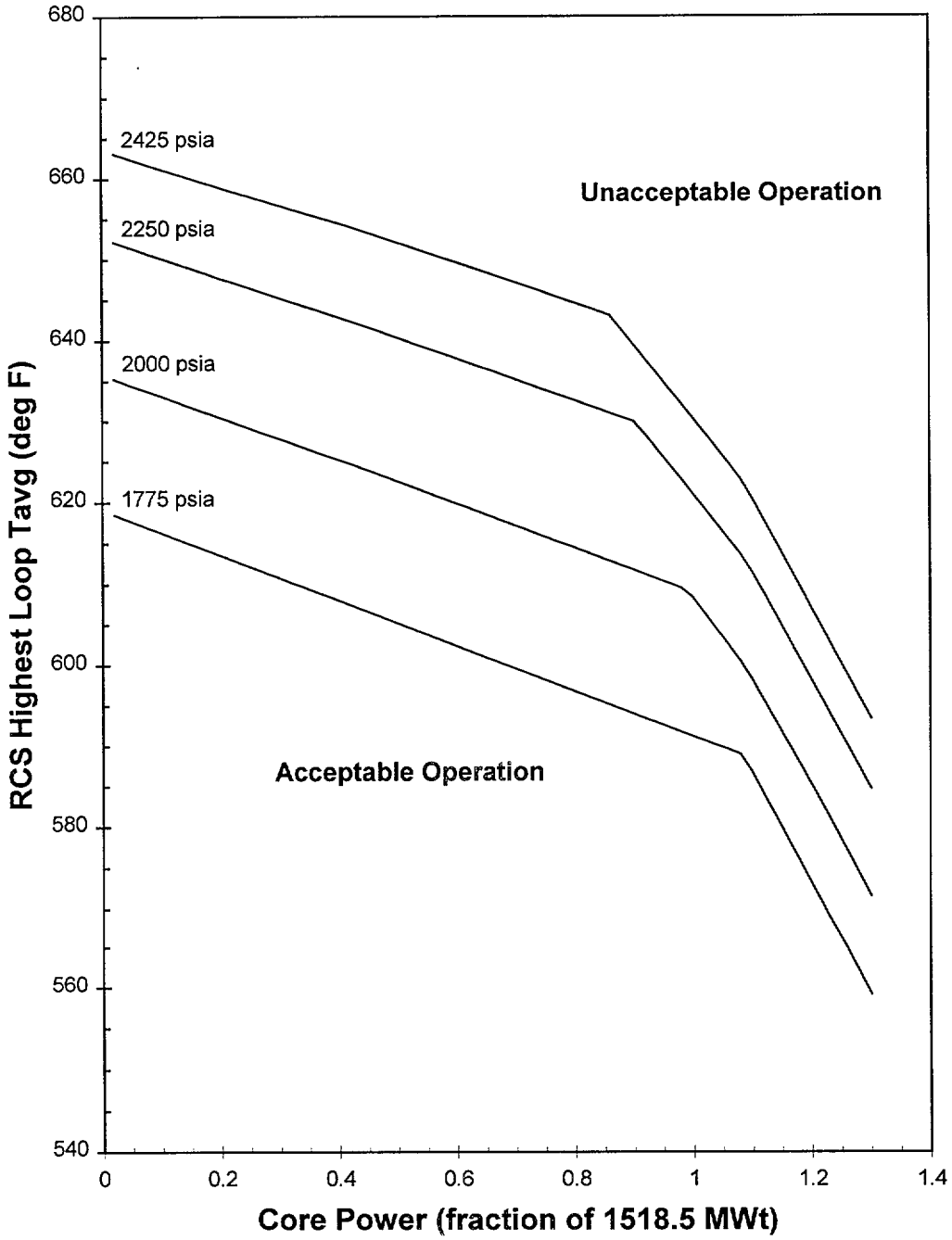


* This figure applies to core reloads with any combination of OFA and Upgraded OFA fuel assemblies.

LA.02

Figure 15.2.1-2*

POINT BEACH NUCLEAR PLANT UNITS 1 AND 2
REACTOR CORE SAFETY LIMITS



* This figure applies to core reloads with any combination of 422V+ fuel assemblies, burned OFA and burned Upgraded OFA fuel assemblies, or a full core of 422V+ fuel assemblies.

15.6.7 ACTION TO BE TAKEN IF A SAFETY LIMIT IS EXCEEDED

Specification

A. If a safety limit is exceeded, ~~the affected reactor shall be shut down and reactor operation shall not be resumed until approval is received from the NRC.~~

~~B. An immediate report shall be made to the Chief Nuclear Officer and the Chairman of the Off-Site Review Committee.~~

LA.01

~~C. The Chief Nuclear Officer shall report the circumstances to the NRC.~~

~~D. A Safety Limit Violation Report including a complete analysis of the circumstances leading to and resulting from the occurrence, effects upon facility components, systems or structures, together with recommendations to prevent a recurrence, shall be prepared. This report shall be submitted to the Chief Nuclear Officer and the Chairman of the Off-Site Review Committee. A Safety Limit Violation Report shall be submitted to the NRC by the Chief Nuclear Officer within 10 days of the occurrence.~~

LB.01

LB.01

LA.01

M.02

Core Safety Limit Action:	"restore compliance and be in MODE 3 within 1 hour."
RCS Pressure Safety Limit Action:	"In MODE 1 or 2, restore compliance and be in MODE 3 within 1 hour" AND "In Modes 3, 4, 5, and 6, restore compliance within 5 minutes."

A.01

15.2.2 SAFETY LIMIT, REACTOR COOLANT SYSTEM PRESSURE

Applicability

Applies to the maximum limit on Reactor Coolant System Pressure.

Objective

To maintain the integrity of the Reactor Coolant System.

A.05

Specification 2.1.2

Specification

The Reactor Coolant System pressure shall not exceed 2735 psig with fuel assemblies installed in the reactor vessel.

In Modes 1, 2, 3, 4, 5 and 6

A.04

Basis

The Reactor Coolant System[1] serves as a barrier preventing radionuclides contained in the reactor coolant from reaching the atmosphere. In the event of a fuel cladding failure the Reactor Coolant System is the primary barrier against the release of fission products. By establishing a system pressure limit, the continued integrity of the Reactor Coolant System is assured. The maximum transient pressure allowable in the Reactor Coolant System pressure vessel under the ASME Code, Section III is 110% of design pressure. The maximum transient pressure allowable in the Reactor Coolant System piping, valves and fittings under USAS Section B31.1 is 120% of design pressure. Thus, the safety limit of 2735 psig (110% of design pressure) has been established.[2]

The nominal settings of the power-operated relief valves (2335 psig), the reactor high-pressure trip and the safety valves (2485 psig) have been established to assure never reaching the Reactor Coolant System pressure safety limit except for the hypothetical locked rotor and rod ejection accidents which use the faulted condition stress limit acceptance criterion of 3105 psig (3120 psia). The initial hydrostatic test was conducted at 3110 psig to assure the integrity of the Reactor Coolant System.

A.03

~~Reference~~

~~(1) FSAR, Section 4~~

~~(2) FSAR, Section 4.3~~

A.01

Insert 1

specified in the COLR in order to preserve the following fuel design criteria:

2.1.1.1 The departure from nucleate boiling ratio (DNBR) shall be maintained:

- ≥ 1.22/1.21 (typical/thimble) for the WRB-1 correlation – cores not containing 422V+ fuel.
- ≥ 1.24/1.23 (typical/thimble) for the WRB-1 correlation – cores containing 422V+ fuel

OR

- ≥ 1.30 for the W-3 correlation when system pressure is > 1000 psia
- ≥ 1.45 for the W-3 correlation when system pressure is ≥ 500 psia and ≤ 1000 psia

2.1.1.2 The peak fuel centerline temperature shall be maintained < 5080°F, decreasing by 58°F per 10,000 MWD/MTU of burnup.

B

COLR & Amendment 193/198

Justification For Deviations - NUREG-1431 Section 2.0

24-May-00

JFD Number	JFD Text	
01 Rev. A	The Mode of Applicability for the RCS Pressure Safety Limit has been expanded to include Mode 6. This change is necessary, as Mode 6 is entered when the first vessel stud is less than fully tensioned. In this configuration, the RCS is still capable of being pressurized, accordingly the Safety Limit should still be applicable. The current Technical Specifications do not contain any Mode limitations on this Safety Limit when there is fuel in the vessel, therefore this change is consistent with the Point Beach Current Licensing Basis.	
	ITS:	**NUREG:**
	2.01.02	2.01.02
	2.02.02.02	2.02.02.02
	B 2.0	B 2.0
02 Rev. B	Not Used	
	ITS:	**NUREG:**
	N/A	N/A
03 Rev. A	Reference to the General Design Criteria (GDC) of 10 CFR 50 Appendix A has been deleted from the Bases of the Technical Specifications, substituting reference to the appropriate section of the FSAR which specifies the Point Beach design criteria. Point Beach was constructed and licensed prior to the GDC being issued. The Point Beach construction permit was issued prior to the GDCs being issued in 1971. Point Beach was designed and constructed utilizing the 1967 proposed GDCs. Accordingly, reference has been provided to the appropriate criteria and section of the Point Beach FSAR which provides explanation of Point Beach's design basis.	
	ITS:	**NUREG:**
	B 2.0	N/A
04 Rev. B	Not Used.	
	ITS:	**NUREG:**
	N/A	N/A
05 Rev. A	The brackets have been removed and the proper plant specific information has been provided.	
	ITS:	**NUREG:**
	2.01.02	2.01.02
	B 2.0	B 2.0

Justification For Deviations - NUREG-1431 Section 2.0

24-May-00

JFD Number	JFD Text	
06 Rev. A	A separate reference for the FSAR Section which describes the high reactor pressure reactor trip function as an overpressure protection function has been added to the Bases. This reference is necessary as the NUREG currently assumes this information is in the same Section as the Reactor Trip System setpoints.	
	ITS:	**NUREG:**
	B 2.0	B 2.0
07 Rev. A	Editorial changes to the NUREG 1431 bases to use plant specific terminology.	
	ITS:	**NUREG:**
	2.01.01	2.01.01
08 Rev. B	Not Used.	
	ITS:	**NUREG:**
	N/A	N/A

2.0 SAFETY LIMITS (SLs)

2.1 SLs

2.1.1 Reactor Core SLs

In MODES 1 and 2, the combination of THERMAL POWER, Reactor Coolant System (RCS) highest loop average temperature, and pressurizer pressure shall not exceed the SLs specified in Figure 2.1.1-1.

Insert 2.0-1

limits

01

2.1.2 RCS Pressure SL

and 6,

Approved TSTF 339

In MODES 1, 2, 3, 4, ~~and~~ 5, the RCS pressure shall be maintained ≤ [2735] psig.

05

B

COLR

2.2 SL Violations

2.2.1 If SL 2.1.1 is violated, restore compliance and be in MODE 3 within 1 hour.

2.2.2 If SL 2.1.2 is violated:

2.2.2.1 In MODE 1 or 2, restore compliance and be in MODE 3 within 1 hour.

or 6,

2.2.2.2 In MODE 3, 4, ~~or~~ 5, restore compliance within 5 minutes.

01

~~2.2.3 Within 1 hour, notify the NRC Operations Center, in accordance with 10 CFR 50.72.~~

~~2.2.4 Within 24 hours, notify the [Plant Superintendent and Vice President - Nuclear Operations].~~

~~2.2.5 Within 30 days a Licensee Event Report (LER) shall be prepared pursuant to 10 CFR 50.73. The LER shall be submitted to the NRC, the [offsite review function], and the [Plant Superintendent, and Vice President - Nuclear Operations].~~

~~2.2.6 Operation of the unit shall not be resumed until authorized by the NRC.~~

Approved TSTF-5, R.1

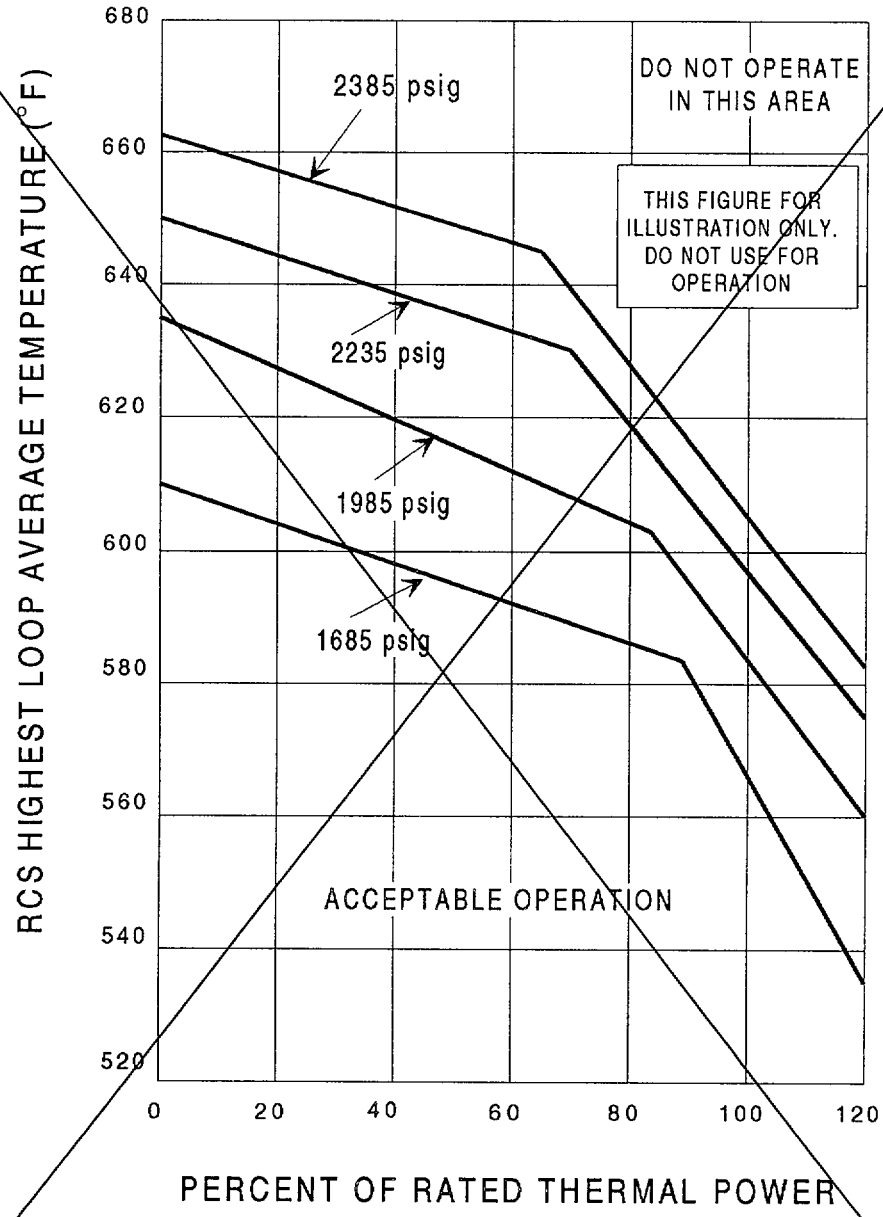


Figure 2.1.1-1 (page 1 of 1)
Reactor Core Safetv Limits

B
COLR

Insert 2.0-1

05

the COLR in order to preserve the following fuel design criteria:

2.1.1.1 The departure from nucleate boiling ratio (DNBR) shall be maintained ≥ [1.17 for the WRB-1/WRB-2 DNB correlations].

2.1.1.2 The peak fuel centerline temperature shall be maintained < [5080°F, decreasing by 58°F per 10,000 MWD/MTU of burnup].

≥ 1.22/1.21 (typical/thimble) for the WRB-1 correlation - cores not containing 422V+ fuel

≥ 1.24/1.23 (typical/thimble) for the WRB-1 correlation - cores containing 422V+ fuel

OR

≥ 1.30 for the W-3 correlation when system pressure is > 1000 psia

≥ 1.45 for the W-3 correlation when the system pressure is ≥ 500 psia and ≤ 1000 psia.

B

COLR

BASES (continued)

BACKGROUND (continued)

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

APPLICABLE SAFETY ANALYSES

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

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

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

B RAI 2.0-1

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

Approved TSTF 339

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

a. High pressurizer pressure trip;

b. Low pressurizer pressure trip;

c. Overtemperature ΔT trip;

d. Overpower ΔT trip;

e. Power Range Neutron Flux trip; and

f. Steam generator safety valves.

appropriate operation of the RPS and the steam generator safety valves.

The limitation that the average enthalpy in the hot leg be less than or equal to the enthalpy of saturated liquid also ensures that the ΔT measured by instrumentation, used in the RPS design as a measure of core power, is proportional to core power.

B COLR

BASES

APPLICABLE SAFETY ANALYSIS (continued)

The SLs represent a design requirement for establishing the RPS trip setpoints identified previously. LCO 3.4.1, "RCS Pressure, Temperature, and Flow Departure from Nucleate Boiling (DNB) Limits," or the assumed initial conditions of the safety analyses (as indicated in the FSAR, Ref. 2) provide more restrictive limits to ensure that the SLs are not exceeded.

B RAI 2.0-1

SAFETY LIMITS

Replace with Insert A

Approved TSTF 339

The curves provided in Figure B 2.1.1-1 show the loci of points of THERMAL POWER, RCS pressure, and average temperature for which the minimum DNBR is not less than the safety analyses limit, that fuel centerline temperature remains below melting, that the average enthalpy in the hot leg is less than or equal to the enthalpy of saturated liquid, or that the exit quality is within the limits defined by the DNBR correlation.

The curves are based on enthalpy hot channel factor limits provided in the COLR. The dashed line of Figure B 2.1.1-1 shows an example of a limit curve at 2235 psig. In addition, it illustrates the various RPS functions that are designed to prevent the unit from reaching the limit.

The SL is higher than the limit calculated when the AFD is within the limits of the $F_1(\Delta I)$ function of the overtemperature ΔT reactor trip. When the AFD is not within the tolerance, the AFD effect on the overtemperature ΔT reactor trips will reduce the setpoints to provide protection consistent with the reactor core SLs (Refs. 3 and 4).

B COLR

APPLICABILITY

SL 2.1.1 only applies in MODES 1 and 2 because these are the only MODES in which the reactor is critical. Automatic protection functions are required to be OPERABLE during MODES 1 and 2 to ensure operation within the reactor core SLs. The steam generator safety valves or automatic protection actions serve to prevent RCS heatup to the reactor core SL conditions or to initiate a reactor trip function, which forces the unit into MODE 3. Setpoints for the reactor trip functions are specified in LCO 3.3.1,

BASES

SAFETY LIMITS VIOLATIONS (continued)

Approved TSTF-5, R.1

~~2.2.6~~

~~If SL 2.1.1 is violated, restart of the unit shall not commence until authorized by the NRC. This requirement ensures the NRC that all necessary reviews, analyses, and actions are completed before the unit begins its restart to normal operation.~~

REFERENCES

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

FSAR, Section 3.1

B
RAI 2.0-1

2. FSAR, Section 7.2.

03

Approved TSTF 339

~~3. WCAP-8746-A, March 1977.~~

~~4. WCAP-9273-NP-A, July 1985.~~

B
COLR

Approved TSTF-5

~~5. 10 CFR 50.72.~~

~~6. 10 CFR 50.73.~~

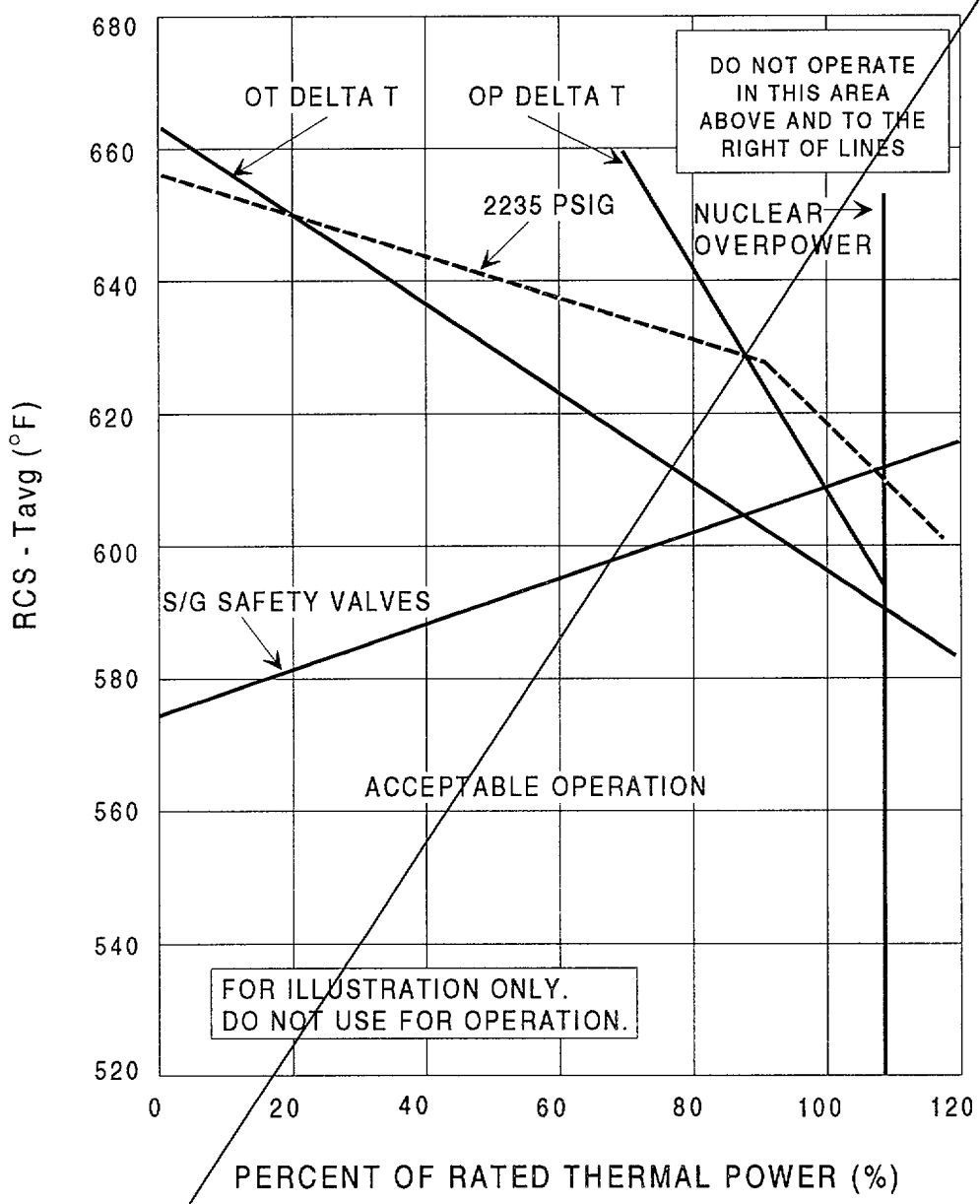


Figure B 2.1.1-1 (page 1 of 1)
Reactor Core Safety Limits vs. Boundary of Protection

B
COLR

Bases 2.1.1

Insert A

The reactor core SLs are established to preclude violation of the following fuel design criteria:

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

b. There must be at least a 95% probability at a 95% confidence level that the hot fuel pellet in the core does not experience centerline fuel melting.

The reactor SLs are use to define the various RPS functions such that the above criteria are satisfied during steady state operation, normal operational transients, and anticipated operational occurrences (AOOs). To ensure that the RPS precludes that violation of the above criteria, additional criteria are applied to the Overtemperature and Overpower ΔT reactor trip functions. That is, it must be demonstrated that the average enthalpy in the hot leg is less than or equal to the saturation enthalpy and that the core exit quality is within the limits defined by the DNBR correlation. Appropriate functioning of the RPS ensures that for variations in the THERMAL POWER, RCS Pressure, RCS average temperature, RCS flow rate, and ΔI that the reactor core SLs will be satisfied during steady state operation, normal operational transients and AOOs.

B

COLR

2.0 SAFETY LIMITS (SLs)

2.1 SLs

2.1.1 Reactor Core SLs

In MODES 1 and 2, the combination of THERMAL POWER, Reactor Coolant System (RCS) highest loop average temperature, and pressurizer pressure shall not exceed the limits specified in the COLR in order to preserve the following fuel design criteria:

2.1.1.1 The departure from nucleate boiling ratio (DNBR) shall be maintained:

≥ 1.22/1.21 (typical/thimble) for the WRB-1 correlation - cores not containing 422V+ fuel
≥ 1.24/1.23 (typical/thimble) for the WRB-1 correlation - cores containing 422V+ fuel

OR

≥ 1.30 for the W-3 correlation when system pressure is > 1000 psia
≥ 1.45 for the W-3 correlation when system pressure is ≥ 500 psia and ≤ 1000 psia

2.1.1.2 The peak fuel centerline temperature shall be maintained < 5080°F, decreasing by 58°F per 10,000 MWD/MTU of burnup.

2.1.2 RCS Pressure SL

In MODES 1, 2, 3, 4, 5, and 6 the RCS pressure shall be maintained ≤ 2735 psig. B COLR

2.2 SL Violations

2.2.1 If SL 2.1.1 is violated, restore compliance and be in MODE 3 within 1 hour.

2.2.2 If SL 2.1.2 is violated:

2.2.2.1 In MODE 1 or 2, restore compliance and be in MODE 3 within 1 hour.

2.2.2.2 In MODE 3, 4, 5, or 6 restore compliance within 5 minutes.

B 2.0 SAFETY LIMITS (SLs)

B 2.1.1 Reactor Core SLs

BASES

BACKGROUND

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

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

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

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

BASES

BACKGROUND (continued)

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

APPLICABLE SAFETY ANALYSES

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

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

b. The hot fuel pellet in the core must no t experience centerline fuel melting.

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

Automatic enforcement of these reactor core SLs is provided by the appropriate operation of the RPS and the steam generator safety valves.

B
RAI 2.0-1 & COLR

The SLs represent a design requirement for establishing the RPS trip setpoints identified previously. LCO 3.4.1, "RCS Pressure, Temperature, and Flow Departure from Nucleate Boiling (DNB) Limits," or the assumed initial conditions of the safety analyses (as indicated in the FSAR, Ref. 2) provide more restrictive limits to ensure that the SLs are not exceeded.

B
RAI 2.0-1

BASES

SAFETY LIMITS

The reactor core SLs are established to preclude violation of the following fuel design criteria:

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

b. There must be at least a 95% probability at a 95% confidence level that the hot fuel pellet in the core does not experience centerline fuel melting.

The reactor SLs are use to define the various RPS functions such that the above criteria are satisfied during steady state operation, normal operational transients, and anticipated operational occurrences (AOOs). To ensure that the RPS precludes that violation of the above criteria, additional criteria are applied to the Overtemperature and Overpower ΔT reactor trip functions. That is, it must be demonstrated that the average enthalpy in the hot leg is less than or equal to the saturation enthalpy and that the core exit quality is within the limits defined by the DNBR correlation. Appropriate functioning of the RPS ensures that for variations in the THERMAL POWER, RCS Pressure, RCS average temperature, RCS flow rate, and ΔI that the reactor core SLs will be satisfied during steady state operation, normal operational transients and AOOs. B COLR

APPLICABILITY

SL 2.1.1 only applies in MODES 1 and 2 because these are the only MODES in which the reactor is critical. Automatic protection functions are required to be OPERABLE during MODES 1 and 2 to ensure operation within the reactor core SLs. The steam generator safety valves or automatic protection actions serve to prevent RCS heatup to the reactor core SL conditions or to initiate a reactor trip function, which forces the unit into MODE 3. Setpoints for the reactor trip functions are specified in LCO 3.3.1, "Reactor Trip System (RTS) Instrumentation." In MODES 3, 4, 5, and 6, Applicability is not required since the reactor is not generating significant THERMAL POWER.

(continued)

SAFETY LIMIT VIOLATIONS

The following SL violation responses are applicable to the reactor core SLs. If SL 2.1.1 is violated, the requirement to go to MODE 3 places the unit in a MODE in which this SL is not applicable.

The allowed Completion Time o f 1 hour recognizes the importance of bringing the unit to a MODE of operation where this SL is not applicable, and reduces the probability of fuel damage.

REFERENCES

1. FSAR, Section 3.1.
2. FSAR, Section 7.2.

B
RAI 2.0-1

B 2.0 SAFETY LIMITS (SLs)

B 2.1.2 Reactor Coolant System (RCS) Pressure SL

BASES

BACKGROUND

The SL on RCS pressure protects the integrity of the RCS against overpressurization. In the event of fuel cladding failure, fission products are released into the reactor coolant. The RCS then serves as the primary barrier in preventing the release of fission products into the atmosphere. By establishing an upper limit on RCS pressure, the continued integrity of the RCS is ensured. According to the Point Beach design criteria (Ref. 1), the reactor pressure coolant boundary (RCPB) design conditions are not to be exceeded during normal operation and anticipated operational occurrences (AOOs). Also, in accordance with the Point Beach design criteria (Ref. 1), reactivity accidents, including rod ejection, do not result in damage to the RCPB greater than limited local yielding.

The design pressure of the RCS is 2500 psia. During normal operation and AOOs, RCS pressure is limited from exceeding the design pressure by more than 10%, in accordance with Section III of the ASME Code (Ref. 2). To ensure system integrity, all RCS components are hydrostatically tested at 125% of design pressure, according to the ASME Code requirements prior to initial operation when there is no fuel in the core. Following inception of unit operation, RCS components shall be pressure tested, in accordance with the requirements of ASME Code, Section XI (Ref. 3).

Overpressurization of the RCS could result in a breach of the RCPB. If such a breach occurs in conjunction with a fuel cladding failure, fission products could enter the containment atmosphere, raising concerns relative to limits on radioactive releases specified in 10 CFR 100, "Reactor Site Criteria" (Ref. 4).

APPLICABLE SAFETY ANALYSES

The RCS pressurizer safety valves, the main steam safety valves (MSSVs), and the reactor high pressure trip have settings established to ensure that the RCS pressure SL will not be exceeded.

The RCS pressurizer safety valves are sized to prevent system pressure from exceeding the design pressure more than 10%, as specified in Section III of the ASME Code for Nuclear Power Plant Components (Ref. 2). The transient that establishes the required relief capacity, and hence valve size requirements and lift settings, is a complete loss of external load without a direct reactor trip. During the transient, no control actions are assumed, except that the safety valves on the secondary plant are assumed to open when the steam pressure reaches the secondary plant safety valve settings, and nominal feedwater supply is maintained.

BASES

APPLICABLE SAFETY ANALYSES (continued)

The Reactor Trip System setpoints (Ref. 5), together with the settings of the MSSVs, provide pressure protection for normal operation and AOOs. The reactor high pressure trip setpoint is specifically set to provide protection against overpressurization (Ref. 7). The safety analyses for both the high pressure trip and the RCS pressurizer safety valves are performed using conservative assumptions relative to pressure control devices.

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

a. Pressurizer power operated relief valves (PORVs);

b. Atmospheric Steam Dumps;

c. Condenser Steam Dumps;

d. Reactor Control System;

e. Pressurizer Level Control System; or

f. Pressurizer spray valves.

SAFETY LIMITS

The maximum transient pressure allowed in the RCS pressure vessel under the ASME Code, Section III, is 110% of design pressure. The maximum transient pressure allowed in the RCS piping, valves, and fittings under USAS, Section B31.1 (Ref. 6) is 120% of design pressure. The most limiting of these two allowances is the 110% of design pressure; therefore, the SL on maximum allowable RCS pressure is 2735 psig.

APPLICABILITY

SL 2.1.2 applies in MODES 1, 2, 3, 4, 5, and 6 because this SL could be approached or exceeded in these MODES due to overpressurization events.

BASES

SAFETY LIMIT VIOLATIONS

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

Exceeding the RCS pressure SL may cause immediate RCS failure and create a potential for radioactive releases in excess of 10 CFR 100, "Reactor Site Criteria," limits (Ref. 4).

The allowable Completion Time of 1 hour recognizes the importance of reducing power level to a MODE of operation where the potential for challenges to safety systems is minimized.

If the RCS pressure SL is exceeded in MODE 3, 4, 5, or 6 RCS pressure must be restored to within the SL value within 5 minutes. Exceeding the RCS pressure SL in MODE 3, 4, 5 or 6 is more severe than exceeding this SL in MODE 1 or 2, since the reactor vessel temperature may be lower and the vessel material, consequently, less ductile. As such, pressure must be reduced to less than the SL within 5 minutes. The action does not require reducing MODES, since this would require reducing temperature, which would compound the problem by adding thermal gradient stresses to the existing pressure stress.

REFERENCES

1. FSAR, Section 4.1.
2. ASME, Boiler and Pressure Vessel Code, Section III, Article NB-7000.
3. ASME, Boiler and Pressure Vessel Code, Section XI, Article IWX-5000.
4. 10 CFR 100.
5. FSAR, Section 7.2.
6. USAS B31.1, Standard Code for Pressure Piping, American Society of Mechanical Engineers, 1967.
7. FSAR, Section 4.2.

Justification For Deviations - NUREG-1431 Section 3.01.01

31-May-00

JFD Number	JFD Text
10 Rev. B	The Bases for NUREG 1431 SR 3.1.1.1 states that shutdown margin shall be verified by performing a reactivity balance calculation. Calculation of shutdown margin is one method of determining compliance with the LCO, however, shutdown margin can also be verified by comparing current boron concentration to a concentration previously determined to bound a conservative set of plant conditions. Shutdown margin verification is done more frequently in ITS (Modes 2, 3, 4, 5) than CTS (only during low power physics testing). Performing a reactivity balance calculation is a lengthy process and unnecessary when the unit is in the Mode of applicability for an extended time period. In this case the shutdown margin (SDM) calculations are performed in advance by reactor engineering, considering all the inputs described in the SR 3.1.1.1 Bases discussion. Conservative input values are used for different plant operating modes to perform the SDM calculations. The results of the SDM calculations are tabulated and provided to the PBNP operators. The operators then compare actual boron concentration to the tabulated results for a given plant condition. If actual boron concentration is greater than the tabulated value, the Technical Specification required SDM is met. Elimination of the words "performing a reactivity balance calculation" from the proposed Bases was done to avoid potential confusion with respect to whether a complete calculation needed to be performed. Adding the above described SDM verification method was considered to be an insertion of unnecessary detail in the Bases. Therefore, the Bases has been modified to remove reference to performing calculations to verify shutdown margin. **ITS:** B 3.01.01 — **NUREG:** B 3.01.01
11 Rev. A	The Bases for SR 3.1.1.1. lists items that are required to be considered in the determination of shutdown margin to include control bank position. Shutdown margin must also consider the effects of the shutdown bank. As such, reference to shutdown banks has been added to the proposed ITS Bases. **ITS:** B 3.01.01 — **NUREG:** B 3.01.01

JFD Number	JFD Text
02 Rev. B	The negative MTC limit and associated Surveillance Requirements have not been adopted in the proposed ITS for Point Beach. NUREG 1431 contains an LCO requirement and Surveillance Requirement SR 3.1.4.2 for a lower MTC limit. This SR is not required to be performed until within 7 EFPD of reaching a RCS boron concentration equivalent to a Rated Thermal Power (RTP) All Rods Out (ARO) configuration of 300 ppm. Accordingly, this SR is not performed until late in core life, relying upon design predictions and monitoring for a reactivity anomaly surveillance to assure operation within limits for the majority of the fuel cycle. The CTS does not contain an LCO requirement nor surveillance's for a negative MTC limit. The negative MTC is one of many design parameters which are verified as part of the reload analysis review (during both the core design and safety analysis process). Core design parameters and performance assumptions are validated during physics testing at the beginning of each cycle. If those tests meet all of their applicable acceptance criteria, then the cycle design is considered acceptable, including the design parameters of positive and negative MTC. Proposed ITS LCO 3.1.2 "Core Reactivity" contains an SR (SR 3.1.2.1), which requires periodic performance of a reactivity balance. SR 3.1.2.1 provides continuing assurance that the core is performing as expected, reactivity is within design expectations, and that the calculational models used to design the core and generate the safety analysis are valid. Failure of this SR 3.1.2.1 results in entry into Required Actions A.1 and A.2 of LCO 3.1.2. The Bases for these Actions has been modified to specify examples of design calculations inputs which need to be verified, of which MTC is one. Calculation methodology/core design provides sufficient margin for negative MTC. Verification of design parameters and monitoring for reactivity anomalies provide assurance that the design assumptions of MTC will be met throughout core life. As such, a specific LCO limitation is not considered necessary and has not been adopted.

ITS:	NUREG:
B 3.01.03	B 3.01.04
N/A	LCO 3.01.04 COND C LCO 3.01.04 COND C RA C.1 SR 3.01.04.02 SR 3.01.04.03
SR 3.01.03.01	SR 3.01.04.01

Justification For Deviations - NUREG-1431 Section 3.01.04

31-May-00

JFD Number	JFD Text
07 Rev. B	The Bases for NUREG 1431 LCO 3.1.4 states that MTC is most positive at beginning of core life. Cores containing fuel assemblies with integral burnable poisons (which can be included in the PBNP fuel assemblies) are not necessarily in their most negative value at the beginning of core life. These cores typically peak within the first few effective full power months of operation. The proposed Bases for the ITS has been changed accordingly to delete this attribute. **ITS:** B 3.01.03 **NUREG:** B 3.01.04
08 Rev. A	As indicated in the Bases for Action A.1 of NUREG LCO 3.1.4, RCS boron concentration is the principle determining factor for the moderator temperature coefficient. Therefore, Action A.1 is modified to require administrative limits on RCS boron concentration instead of control bank withdrawal limits. These administrative controls ensure MTC is maintained within limits and preserve the intent of Action A.1. In addition, as discussed in JFD 05, the upper MTC limit is determined for an all rods out configuration, therefore, it is appropriate to place the control limits on RCS boron concentration. **ITS:** LCO 3.01.03 COND A RA A.1 **NUREG:** LCO 3.01.04 COND A RA A.1

Description of Changes - NUREG-1431 Section 3.01.05

01-Jun-00

DOC Number	DOC Text

A.01
Rev. A

In the conversion of Point Beach current Technical Specifications (CTS) to the proposed plant specific Improved Technical Specifications (ITS), certain wording preferences or conventions are adopted which do not result in technical changes (either actual or interpretational). Editorial changes, reformatting, and revised numbering are adopted to make the ITS consistent with the Standard Technical Specifications, Westinghouse Plants, NUREG-1431, Revision 1 (i.e., Improved Standard Technical Specifications (ISTS)).

CTS:	ITS:
15.03.10	LCO 3.01.04
15.03.10.B.01	LCO 3.01.04 SR 3.01.04.01
15.03.10.B.01.A.03	LCO 3.01.04 COND A
15.03.10.B.01.A.03.A	LCO 3.01.04 COND A RA A.1.1 LCO 3.01.04 COND A RA A.1.2
15.03.10.B.01.A.03.B	LCO 3.01.04 COND A RA A.2
15.03.10.B.01.B.01	LCO 3.01.04 COND B RA B.1
15.03.10.B.01.B.01.A	LCO 3.01.04 COND B RA B.2.1.1
15.03.10.B.01.B.01.C	LCO 3.01.04 COND B RA B.2.3
15.03.10.B.01.B.01.F	LCO 3.01.04 COND C LCO 3.01.04 COND C RA C.1
15.03.10.B.01.B.02	LCO 3.01.04 COND D
15.03.10.B.01.B.02.A	LCO 3.01.04 COND D RA D.1.1 LCO 3.01.04 COND D RA D.1.2
15.03.10.H	SR 3.01.04.03
15.03.10.H.01	SR 3.01.04.03
15.04.01 T 15.04.01-02 09 (A)	SR 3.01.04.03 SR 3.01.04.03
15.04.01 T 15.04.01-02 10	SR 3.01.04.02

A.02
Rev. A

The CTS provides an introductory statement (Applicability) which simply states which systems/components are addressed within a given section. This same information, while worded differently, is contained within the title of each ITS LCO. Accordingly, this change is a change in format with no change in technical requirement.

CTS:	ITS:
15.03.10 APPL	LCO 3.01.04

Description of Changes - NUREG-1431 Section 3.01.05

01-Jun-00

DOC Number	DOC Text
A.03 Rev. A	The CTS provides an introductory statement (Objective) at the beginning of this Section of the Technical Specifications which provides a brief summary of the purpose for this Section. This information is contained in the Bases Section of the ITS. This information does not establish any regulatory requirements for the systems and components addressed within this Section. Accordingly, deletion of this information does not alter any requirement set forth in the Technical Specifications. This change is administrative and consistent with the format and presentation for the ITS as provided in NUREG 1431.

CTS:	ITS:
15.03.10 OBJ	B 3.01.04

A.04 Rev. A	The CTS requires all shutdown and control rods to be operable during power and low power operation. All indicated rod positions are required to be within an alignment limit based upon demanded rod position. ITS LCO 3.1.4 will require all shutdown and control rods to be operable and within their alignment limits. The rod alignment limits themselves have been moved to ITS SR 3.1.4.1. ITS SR 3.0.1 establishes the requirement that surveillances must be met when the LCO is applicable. Moving this limit to a surveillance makes the presentation of this LCO more concise, while retaining the same regulatory requirement through application of SR 3.0.1. Accordingly, these changes do not represent a change in intent or usage and are therefore administrative.

CTS:	ITS:
15.03.10.B.01	LCO 3.01.04

A.05 Rev. A	The CTS Actions for untrippable and misaligned control rods contain an Action which requires verification that Shut Down Margin (SDM) exceeds its required value which is specified in CTS Table 15.3.10-2. CTS Table 15.3.10-2 has been proposed for relocation to the Core Operating Limits Report (COLR) as discussed in Description of Change LA.1 of LCO 3.1.1. Therefore, the Actions for untrippable and misaligned control rods have been changed to reference the SDM limits provided in the COLR. This change has been classified as an administrative change relative to this LCO, as the basis for relocation of the SDM limit itself has been addressed in LCOs 3.1.1.

CTS:	ITS:
15.03.10.B.01.A.01.A	COLR
15.03.10.B.01.A.03.A	COLR
15.03.10.B.01.B.01.A	COLR
15.03.10.B.01.B.01.C	COLR
15.03.10.B.01.B.02.A	COLR

DOC Number	DOC Text
A.06 Rev. B	CTS 15.3.10.B.1.A.1.C, 15.3.10.B.01.A.03.B, and 15.3.10.B.01.B.02.B requires the unit to be placed into Hot Shutdown if the LCO Actions are not met. The CTS definition of Hot Shutdown requires the reactor to be greater than or equal to 540 degrees and subcritical by greater than or equal to the required Shutdown Margin (changes to this definition are addressed in Description of Change M.2 of Section 1.0 of this conversion package). This condition is equivalent to ITS Mode 3. Therefore, these changes are administrative.

CTS:	ITS:
15.03.10.B.01.A.01.C	LCO 3.01.04 COND A RA A.2
15.03.10.B.01.A.03.B	LCO 3.01.04 COND A RA A.2
15.03.10.B.01.B.02.B	LCO 3.01.04 COND D RA D.2

DOC Number	DOC Text
A.07 Rev. A	The CTS specifies control rod alignment limits are to be fulfilled using the demand and individual rod position indicators. CTS 15.3.10.B.1.b.1 and 2 provides an exception to the use of the demand and individual rod indicators for determining alignment when the reason for the misalignment is caused by a malfunctioning position indicator. The proposed ITS LCO 3.1.4 will continue to require control rod alignment, while ITS LCO 3.1.7 establishes the preferential means of determining rod position. Based on the restructuring of the ITS with its associated usage rules, it is no longer necessary to specifically state "except for misalignments caused by malfunctioning rod position indicators". This change is administrative.

CTS:	ITS:
15.03.10.B.01.B.01	DELETED LCO 3.01.04 LCO 3.01.04 LCO 3.01.04 COND B
15.03.10.B.01.B.02	DELETED LCO 3.01.04

DOC Number	DOC Text
A.08 Rev. A	The CTS specifies that FQ(Z) and FN Delta H are to be verified to be within limits within 72 hours of determining that a control rod is misaligned. The ITS has substituted reference to the Surveillance Requirements (SR 3.2.1.1, SR 3.2.1.2, and SR 3.2.2.1) which are used to verify that these limits are met in place of reference to the limitation itself. The specific surveillance referenced to verify that FQ(Z) and FN Delta H are met have been previously addressed in DOC L.04 of LCO 3.2.1. Reference to the specific Surveillances which verify that the thermal limits are met is consistent with the format and presentation of NUREG 1431 and is administrative.

CTS:	ITS:
15.03.10.B.01.B.01.D	LCO 3.01.04 COND B RA B.2.4
15.03.10.B.01.B.01.E	LCO 3.01.04 COND B RA B.2.5

DOC Number	DOC Text
A.09 Rev. A	The CTS specifies Actions for control rods which do not meet their rod drop time which state that; if the reactor is critical the control rod must be declared untrippable, and if the reactor is subcritical the reactor must be maintained in a subcritical condition. The proposed ITS, while not presented in the same fashion, establishes the same Actions. If a control rod is determined to be outside of its required drop time when the reactor is subcritical, LCO 3.0.4 prohibits entry into Mode 1 or 2 (reactor critical) because the Actions for an inoperable control rod do not allow indefinite operation in Modes 1 or 2. If a control rod is determined to be outside of its required rod drop time with the reactor critical (ITS Modes 1 and 2), The ITS requires that the control rod be declared inoperable, which ultimately requires the unit to be placed into Mode 2 with Keff less than 1.0 within six hours, which as discussed in Description of Change A.06 of this LCO is equivalent to the CTS Action. Accordingly, this change is administrative.

CTS:	ITS:
15.03.10.H.01	DELETED
15.03.10.H.01.A	DELETED
15.03.10.H.01.B	DELETED

DOC Number	DOC Text
A.10 Rev. A	The CTS states that rod drop timing will be performed for all full length control rods while the ITS specifies that rod drop timing will be performed for all control rods. The Point Beach design no longer incorporates partial length control rods; therefore, deletion of this nomenclature does not alter the testing performed. All control rods will continue to be drop timed.

CTS:	ITS:
15.04.01 T 15.04.01-02 09 (A)	DELETED

DOC Number	DOC Text
A.11 Rev. A	The CTS requires control rod drop timing to be performed at rated reactor coolant flow. The ITS will continue this practice, but has changed the phrasing of this prerequisite condition to "all reactor coolant pumps running". The reactor coolant pumps are verified to provide a minimum of 100% of the required forced circulation through the reactor core by proposed SR 3.4.1.3. Accordingly, running all reactor coolant pumps is equivalent to the CTS requirement to establish rated reactor coolant flow.

CTS:	ITS:
15.04.01 T 15.04.01-02 09 (A)(3)	SR 3.01.04.03

DOC Number	DOC Text
A.12 Rev. A	CTS 15.3.10.B.1 states that control rod must be operable during power and low power operation which has previously been established to be equivalent to ITS Modes 1 and 2 with Keff greater than or equal to 1.0, as addressed in Description of Change A.6 of this Section. Line item 10 of CTS Table 15.4.1-2, requires performance of partial control rod movement tests with Note 18 stating that the partial movement testing is not required to be performed if the reactor is subcritical (ITS Mode 3). ITS SR 3.0.1 establishes the requirement that surveillances must be met when the LCO is applicable (ITS Modes 1 and 2 with Keff greater than or equal to 1.0) which is equivalent to power and low power operation, making Note 18 unnecessary in the ITS. Accordingly, the deletion of Note 18 as applied to line item 10 of Table 15.4.1-2 is administrative.

CTS:	ITS:
15.04.01 T 15.04.01-02 10 (18)	DELETED

DOC Number	DOC Text
A.13 Rev. A	Note 22 to line item 19 of CTS Table 15.4.1-2, states that shiftly control rod alignment channel checks are not required during periods of cold shutdown and refueling, but must be performed prior to reactor criticality if it had not been performed within its previous surveillance interval. This frequency notation is ambiguous in that it does not provide any specific guidance between cold shutdown and reactor critical operations. The CTS Mode of Applicability for control rod operability and alignment has been determined to be equivalent to ITS Mode 1 and 2 with Keff greater than or equal to 1.0 as stated in Description of Change A.6 of this Section. CTS 15.4.0.1 states that surveillance requirements shall be met when the system or component is required to be operable. By applying Specification 15.4.0.1 to the "Plant Conditions When Required" as modified by Note 22, the CTS required mode of performance for this surveillance has been determined to be equivalent to ITS Modes 1 and 2 with Keff greater than or equal to 1.0. ITS SR 3.0.1 establishes the requirement that surveillances must be met when the LCO is applicable. As such, the ITS mode of performance for this surveillance is equivalent to the CTS, making this change administrative.

CTS:	ITS:
15.04.01 T 15.04.01-01 19	LCO 3.01.04
15.04.01 T 15.04.01-01 19 (22)	DELETED
15.04.01 T 15.04.01-01 19.A	LCO 3.01.04
15.04.01 T 15.04.01-01 19.A (22)	DELETED
15.04.01 T 15.04.01-01 ALL	LCO 3.01.04

DOC Number	DOC Text
A.14 Rev. A	CTS Table 15.4.1-1 line item 19 requires the performance of a channel check for control rods on a shiftly basis, which has been concluded to be equivalent to the ITS Surveillance Requirements which verify that the control rods are within their alignment limits. The control rod analog and demand position indicators do not provide any protective functions. These channels are used solely for the purpose of verifying that the control rod alignment limits are maintained. A channel check as discussed in CTS Section 15.4.1 is intended to be a simple observation of instrument function, which is fulfilled through verification of control rod alignment limits. Performance of the proposed ITS surveillances while stated to verify operational limits still encompasses an observation of required channel function while clarifying the intended control rod alignment check. This change is administrative.

CTS:	ITS:
15.04.01 T 15.04.01-01 19	SR 3.01.04.01
15.04.01 T 15.04.01-01 19.A	SR 3.01.04.01

A.15 Rev. A	The Bases of the current Technical Specifications for this section have been completely replaced by revised Bases that reflect the format and applicable content of PBNP ITS, consistent with the Standard Technical Specifications for Westinghouse Plants, NUREG-1431. The revised Bases are as shown in the PBNP ITS Bases.

CTS:	ITS:
BASES	B 3.01.04

A.16 Rev. B	The CTS requires the control rods to be partially moved quarterly to confirm that the control rods are not impaired in any fashion that could impact the control rods capability to trip upon demand. The frequency for performing this test in the ITS is 92 days, which is equivalent to the CTS frequency. Accordingly, this change is administrative.

CTS:	ITS:
15.04.01 T 15.04.01-02 10	SR 3.01.04.02

DOC Number	DOC Text
L.01 Rev. A	CTS Action 15.3.10.B.1.a.1.a and b require Shutdown Margin (SDM) to either be; verified to exceed the applicable value, or to be restored by boration within one hour when a control rod is found to be untrippable. Similarly, CTS Action 15.3.10.B.1.b.1.a requires SDM to either be; verified to exceed the applicable value, or to be restored by boration within one hour when a control rod is found to be misaligned. In the unlikely event of an untrippable or misaligned control rod with SDM not within limits, the CTS Action to restore SDM via boration within one hour is not considered to be a viable action. Restoration of SDM would require determination of the SDM deficit, quantification of the amount of boration required, initiation and completion of the boration, and a confirmatory sample to conclude that the required RCS boron concentration was achieved. The proposed ITS will require initiation of boration to restore SDM within one hour. Relaxing the Required Actions from restoring SDM by boration to the initiation of boration will require prompt action to be initiated to restore SDM (boration) without requiring entry into the default Condition and Required Action if restoration of SDM takes in excess of one hour. Entry into the default Conditions and Required Actions will require the unit to be placed into Mode 3 within six hours. The initiation of a unit shutdown will not restore SDM to within limits; continued boration is the only appropriate Action for restoration of SDM.

CTS:	ITS:
15.03.10.B.01.A.01.B	LCO 3.01.04 COND A RA A.1.2
15.03.10.B.01.A.03.A	LCO 3.01.04 COND A RA A.1.2
15.03.10.B.01.B.01.A	LCO 3.01.04 COND B RA B.2.1.2
15.03.10.B.01.B.02.A	LCO 3.01.04 COND D RA D.1.2

L.02 Rev. A	The CTS requires rod drop testing to be performed in the hot condition for rods which have had maintenance performed. The CTS defines the hot shutdown condition as being subcritical by at least the required Shutdown Margin, with Tavg being greater than 540 degrees. The ITS will require rod drop testing to be performed prior to reactor criticality with RCS temperature greater than or equal to 500 degrees. The 40 degree decrease in testing condition will not significantly alter control rod drop time, in fact rod drop times at reduced temperatures have been shown to be slightly longer due to increased RCS density. Allowing testing at this slightly reduced temperature is still representative of operating conditions, while allowing added scheduling flexibility which was the intent of the CTS.

CTS:	ITS:
15.04.01 T 15.04.01-02 09 (A)(4)	SR 3.01.04.03

DOC Number	DOC Text	
L.03 Rev. A	The CTS requires rod drop testing to be performed under both hot and cold conditions, with only the hot drop tests requiring to be timed. Cold drop testing is considered to be a good practice for verification of control rod trippability prior to plant heat up where rod drop timing is performed; but is not required by the NUREG. Performance of rod drops in a cold condition could prevent having to return the plant to a cold condition to enact repairs if a problem were disclosed at a higher temperature. There are no credible failure mechanisms that would exist solely under cold conditions nor has there been an occurrence of a control rod failing to trip under cold condition alone. Satisfactory demonstration of control rod trippability in a hot condition, as proposed, is sufficient to provide adequate assurance of function prior to entry into the Mode of Applicability for control rods.	
	CTS:	**ITS:**
	15.04.01 T 15.04.01-02 09 (A)(3)	DELETED
L.04 Rev. B	Not Used.	
	CTS:	**ITS:**
	N/A	N/A

DOC Number	DOC Text
L.05 Rev. A	CTS Table 15.4.1-1 footnote 18 requires rod positions to be logged once per hour, after load changes in excess of 10% power, and after rod motion in excess of 30 steps when the on-line computer is inoperable. This verification is required in addition to routine verification of analog rod position and demand position indication which is required once per shift. Actual and demanded control rod positions are monitored by the on-line computer, which will initiate an alarm if rod alignment exceeds predetermined limits. However, the on-line computer alarm function does not provide any safety function nor does it input to any protection circuits. This alarm merely provides a non-safety means of alerting personnel to a condition which does not comply with an LCO requirement. Inoperability of the alarm in and of itself does not lead to a control rod misalignment. Alarm inoperability represents a reduction in monitoring capability for a condition (control rod misalignment) which rarely occurs. Control rod positions are required to be routinely verified once every 12 hours by the proposed ITS. Deletion of the increased surveillance and conditional frequencies (logged once per hour, after load changes in excess of 10% power, and after rod motion in excess of 30 steps with an inoperable alarm) does not alleviate the responsibility of the licensee to be vigilant of plant conditions and LCO compliance. Typically, the unit is operated with the control rods well within their associated alignment limits with significant rod motion made only due to planned evolutions. However, significant rod motion could be the result of an unplanned evolution such as a large generator load rejection. Unplanned evolutions of this nature are readily apparent and result in increased monitoring of affected parameters and significant plant conditions. Rod position and demand position indicators in combination with routine surveillance verification (every 12 hours) provides adequate assurance that non-compliance is readily detectable without the need for increased monitoring. Accordingly, this requirement may be deleted form the Technical Specifications as it is not required to provide adequate protection of public health and safety.

CTS:	ITS:
15.04.01 T 15.04.01-01 19.B	DELETED
15.04.01 T 15.04.01-01 NOTE (18)	DELETED

DOC Number	DOC Text
L.06 Rev. A	The CTS states that control rod drop timing must be performed on rods which had maintenance performed on them. This provision is no longer necessary in the ITS. ITS SR 3.1.4.3 requires rod drop testing to be performed to verify control rod operability. Post maintenance testing is captured through application of SR 3.0.1 and SR 3.0.2. SR 3.0.1 establishes the requirement that surveillances must be met when the LCO is applicable. Implicit in the application of SR 3.0.1, is the need to ensure that all Surveillance Requirements remain valid upon completion of maintenance. Following any maintenance, a review of applicable Surveillance Requirements must be conducted to determine the appropriate post maintenance testing that must be completed in order to declare the affected equipment operable. This includes ensuring applicable surveillances are not invalidated by the maintenance performed and their most recent performance is within its required frequency of performance in accordance with SR 3.0.2. If the review determines that the maintenance performed could not invalidate the surveillances and the last performed surveillance was within the required periodicity, then the surveillance would not be required post maintenance. This is less restrictive than the CTS requirement to perform rod drop testing following maintenance with no exceptions.

CTS:	ITS:
15.04.01 T 15.04.01-02 09 (A)(4)	DELETED

DOC Number	DOC Text
L.07 Rev. A	The CTS required frequency for performance of rod alignment verifications is "once per shift", while the proposed frequency of performance for the ITS is every 12 hours. The nominal Point Beach shift duration is 8 hours. Therefore this change extends the nominal time between performances of this surveillance by 4 hours, resulting in a relaxation of the current requirement. This relaxation is acceptable, because the 12 hour Frequency takes into account other rod position information that is continuously available to the operator in the control room, so that during actual rod motion, deviations can immediately be detected.

CTS:	ITS:
15.04.01 T 15.04.01-01 19	SR 3.01.04.01
15.04.01 T 15.04.01-01 19.A	SR 3.01.04.01
15.04.01 T 15.04.01-01 S - EACH SHIFT	SR 3.01.04.01

DOC Number	DOC Text
LA.01 Rev. A	The CTS requires performance of rod worth measurements following each refueling shutdown prior to power operation. This surveillance has been moved to licensee control. The ITS will continue to require performance of a reactivity balance prior to entry into Mode 1 (greater than or equal to 5% power). The reactivity balance is a measure of the predicted versus measured core reactivity which will provide an indirect qualitative verification of control rod worth. The positive reactivity inherent in the core design must be balanced by the negative reactivity of the control components, thermal feedback, neutron leakage, and neutron absorbers. Meeting the acceptance limits of the reactivity balance provides assurance that Design Basis Accident and transient analyses remain valid. Large reactivity differences would be indicative of unanticipated changes in fuel and neutron absorbers. Changes in excess of 1% delta K/k must be evaluated in accordance with the reactivity balance LCO (ITS LCO 3.1.2) to determine that the core is acceptable for continued operation. The Bases of ITS LCO 3.1.8 list control rod worth as a core physics test for which the Mode 2 physics testing exception LCO is intended to be used. In addition, control rod worth is a physics test which is specified in ANSI/ANS 19.6.1-1985, "Reload Startup Physics Tests for Pressurized Water Reactors," which is used as a basis document for physics testing at Point Beach. The CTS does not specify a specific acceptance criteria nor does it specify the number of control rods which must be verified during the performance of the control rod worth test, leaving these variables currently to licensee control. A ten percent margin has been assumed between the calculated control rod worths and the worth assumed in the safety analysis. Previous performances of this test have found that the analysis assumptions relative to predicted ejected rod worth and power peaking factors are consistently overpredicted. Based on the above, it has been concluded that performance of a reactivity balance provides sufficient confidence that the assumptions of the safety analysis are maintained and relocation of the control rod worth tests to licensee control is acceptable based on the absence of acceptance criteria and the margins that exist. Control of the rod worth measurement test, will like other core physics tests continue to be controlled in accordance with 10CFR 50.59. **CTS:** 15.04.01 T 15.04.01-02 09 (B) **ITS:** TRM 3.01.01
M.01 Rev. A	The CTS defines control rod operability as trippability. When a control rod fails to step on demand, the CTS allows up to six hours to determine whether the problem is due to an electrical problem in the rod control system (control rod still operable-trippable), or a problem exists which could inhibit the control rods ability to trip in upon demand. The ITS will continue to define control rod operability based upon its ability to trip upon demand, but delete the CTS provision allowing up to six hours to determine operability. **CTS:** 15.03.10.B.01 **ITS:** DELETED

DOC Number	DOC Text
M.02 Rev. A	CTS 15.3.10.B.1.a.1 and 15.3.10.B.1.a.2 will allow continuous operations with a single untrippable control rod provided shutdown margin is maintained and the rod insertion limits are adjusted to account for the reactivity worth of the stuck rod. The proposed ITS will require the unit to be shutdown whenever one or more control rods are determined to be untrippable. Deletion of the provision to allow continued operation with a single control rod stuck is more restrictive than the CTS, consistent with the provisions of NUREG 1431. The proposed definition of shutdown margin will continue to require the worth of any stuck rod to be considered. In addition, CTS Actions 15.3.10.B.1.a and 15.3.10.B.1.c have been combined into one Condition (ITS Condition A) based on the Actions for one or more inoperable control rods being the same.

CTS:	ITS:
15.03.10.B.01.A.01.A	LCO 3.01.04 COND A
15.03.10.B.01.A.01.C	LCO 3.01.04 COND A RA A.2
15.03.10.B.01.A.02	DELETED
15.03.10.B.01.A.02.A	DELETED
15.03.10.B.01.A.02.B	DELETED
15.03.10.B.01.A.02.C	DELETED

M.03 Rev. A	The CTS allows continuous operation with a misaligned control rod at power levels not to exceed 75%, with analysis of hot channel factors and allowable power level required only if operation above 75% power with a misaligned control rod is desired. The proposed ITS will restrict operation with a misaligned control rod to less than or equal to 75% power with a misaligned control rod indefinitely. For continued operation at any power level less than or equal to 75% power, ITS will require a reevaluation and confirmation of safety analysis results within 5 days of a control rod becoming misaligned. This change is an added restriction on plant operation being proposed consistent with NUREG 1431.

CTS:	ITS:
15.03.10.B.01.B.01.G	LCO 3.01.04 COND B RA B.2.6

DOC Number	DOC Text
M.04 Rev. A	The CTS 15.3.10.H requires control rod drop time to be verified with RCS temperature greater than the minimum temperature for criticality. CTS 15.3.1.F establishes the minimum temperature for criticality as being to the left of the criticality curve presented on the plant heatup limitations curve (Figure 15.3.1-1). The plant heat up curve criticality limit is based on achieving a minimum vessel temperature of no lower than 40 degrees above the minimum permissible temperatures calculated in Appendix G of the ASME Code (360 to approximately 445 degrees dependent upon RCS pressure). The proposed ITS will require control rod drop time to be verified with RCS Tavg greater than or equal to 500 degrees. The testing condition proposed is to simulate a reactor trip under actual conditions from an operating condition. This change is a more restrictive change, because the CTS would allow testing to be performed as low as 360 degrees in fulfilling this CTS requirement. **CTS:** 15.03.10.H.01 **ITS:** SR 3.01.04.03
M.05 Rev. A	The CTS requires power to be reduced to less than or equal to 75% power within eight hours if a control rod is not within its alignment limit. The ITS reduces this time frame to two hours. Two hours is a reasonable time to either restore a control rod to within its alignment limit or to reduce reactor power to less than or equal to 75% power. This time frame is consistent with that contained in NUREG 1431 and is a more restrictive requirement. **CTS:** 15.03.10.B.01.B.01.B **ITS:** LCO 3.01.04 COND B RA B.2.2
M.06 Rev. A	The CTS requires control rods to be periodically tested by "partial movement of all rods". This test is intended to confirm that the control rods are capable of tripping upon demand. The proposed ITS will verify freedom of movement by requiring the control rods to be moved a minimum of at least ten steps in either direction. This imposes an additional acceptance criteria for control rod freedom of movement which does not exist in the CTS. This change is a more restrictive change being made consistent with NUREG 1431. **CTS:** 15.04.01 T 15.04.01-02 10 **ITS:** SR 3.01.04.02
M.07 Rev. B	CTS 15.3.10.B.1 requires all shutdown and control rods to be operable (trippable and aligned) during power and low power operation. The proposed ITS 3.1.4 (NUREG-1431 LCO 3.1.5) applicability is Modes 1 and 2, which is more restrictive than the CTS (i.e. reactor can be subcritical in Mode 2). Therefore, adopting the NUREG-1431 Mode of applicability is more restrictive. **CTS:** 15.03.10.B.01 **ITS:** LCO 3.01.04

A.1

15.3.10 CONTROL ROD AND POWER DISTRIBUTION LIMITS

Applicability

Applies to the operation of the control rods and to core power distribution limits. A.02

Objective A.03

To insure (1) core subcriticality after a ~~reactor trip~~, (2) a limit on potential reactivity ~~insertions~~ from a hypothetical rod cluster control ~~assembly~~ (RCCA) ejection, and (3) ~~an acceptable~~ core power distribution during ~~power~~ operation.

Specification

A. SHUTDOWN MARGIN

1. The shutdown margin shall exceed the applicable value as shown in Figure 15.3.10-2 under all steady-state operating conditions from 350°F to full power. If the shutdown margin is less than the applicable value of Figure 15.3.10-2, within 15 minutes initiate boration to restore the shutdown margin. < See LCO 3.1.1 >

2. A shutdown margin of at least 1% Δk/k shall be maintained when the reactor coolant temperature is less than 350°F. If the shutdown margin is less than this limit, within 15 minutes initiate boration to restore the shutdown margin.

B. ROD OPERABILITY AND BANK ALIGNMENT LIMITS LCO 3.1.4

Mode 1 and 2 M.7

1. During power and low power operation, all shutdown and control rods shall be operable, with all individual indicated rod positions within twelve steps of their bank demand position, except when the bank demand position is ≤30 steps or ≥215 steps. In this case, all individual indicated rod positions shall be within 24 steps of their bank demand position. limits A.4 SR 3.1.4.1

B RAI 3.1.

M.1 ~~If an RCCA does not step in upon demand, up to six hours is allowed to determine whether the problem with stepping is an electrical problem. If the problem cannot be resolved within six hours, the RCCA shall be declared inoperable until it has been verified that it will step in or would drop upon demand~~.

M.2

a. Rod Operability Requirements

LCO 3.1.4 Cond A

(1) If one ~~rod is~~ or more rods are determined to be untrippable, perform the following actions:

A.1

LCO 3.1.4 RA A.1.1 and A.1.2

M.2

AND

A.5

L.1

initiate action to

(a) Within one hour verify that the shutdown margin exceeds the applicable value ~~as shown in Figure 15.3.10-2~~; provided in the COLR,
OR
(b) Within one hour restore the shutdown margin by boration;
~~OR~~
(c) Within six hours be in hot shutdown.

LCO 3.1.4 RA A.2

B

RAI 3.1.5-1

A.6

Mode 3

~~(2) If sustained power operation with an untrippable rod is desired, perform the following actions:~~

~~(a) Within one hour verify that the shutdown margin exceeds the applicable value as shown in Figure 15.3.10-2; OR within one hour restore the shutdown margin by boration;~~
~~AND~~
~~(b) Within six hours, adjust the insertion limits to reflect the worth of the untrippable rod.~~

~~(c) If the above actions and associated completion times are not met, be in hot shutdown within six hours.~~

A.1

Consolidated with Cond A, RA A.1.1, A.1.2, and A.2

(3) If more than one rod is determined to be untrippable, perform the following actions:

(a) Within one hour verify that the shutdown margin exceeds the applicable value ~~as shown in Figure 15.3.10-2~~; OR within one hour restore the shutdown margin by boration;
AND
(b) Within six hours be in hot shutdown.

initiate action to

L.1

A.6

Mode 3

provided in the COLR,

A.5

A.7

LCO Statement/ Cond B

b. Rod Bank Alignment Limits

(1) If it has been determined that one rod is not within alignment limits, ~~and the indicated misalignment is not being caused by malfunctioning rod position indication,~~ within one hour restore the rod to within alignment limits; OR perform the following actions:

RA B.1

L.1

initiate action to

(a) Within one hour verify that the shutdown margin exceeds the applicable value ~~as shown in Figure 15.3.10-2~~; OR within one hour restore the shutdown margin by boration;
AND

provided in the COLR,

A.5

RA B 2.2

(b) Within ~~eight~~ two hours reduce thermal power to ≤75 percent of rated thermal power;
AND

M.5

RA B.2.1.1/ B.2.1.2

A.1

TABLE 15.4.1-2 (Continued)

		Test	Frequency
< See LCOs 3.7.15 and 3.7.16 >			
7.	Spent Fuel Pit	a) Boron Concentration	Monthly
		b) Water Level Verification	Weekly
8.	Secondary Coolant	Gross Beta-gamma Activity or gamma isotopic analysis	Weekly[6]
< See LCO 3.7.18 >		Iodine concentration	Weekly when gross Beta-gamma activity equals or exceeds 1.0 μCi/g [6]
SR 3.1.4.3			
9.	Control Rods	a) Rod drop times of all ~~full length~~ rods [2]	Each refueling or after maintenance that could affect proper functioning [2]
Verify rod freedom of movement (trippability) by moving each control rod not fully inserted in the core ≥ 10 steps in either direction		b) Rodworth measurement	~~Following each refueling shutdown prior to commencing power operation~~
10.	Control Rod	~~Partial movement of all rods~~	~~Quarterly~~ [16]
SR 3.1.4.2			92 days
11.	Pressurizer Safety Valves	Set point < See LCO 3.4.10 >	Every five years [11]
12.	Main Steam Safety Valves	Set Point < See LCO 3.7.1 >	Every five years [11]
13.	Containment Isolation Trip	Functioning < See LCO 3.6.3 and	Each refueling shutdown
14.	Refueling System Interlocks	Functioning < See LCO 3.9.1 >	Each refueling shutdown
15.	Service Water System	Functioni < See LCO 3.7.8 >	Each refueling shutdown
16.	Primary System Leakage	Evaluate < See LCO 3.4.13 >	Monthly [6]
17.	Diesel Fuel Supply	Fuel invento < See LCO 3.8.3 >	ily
18.	Deleted		
19.	Deleted		
20.	Boric Acid System	Storage Tank and piping temperatures ≥ temperature required by Table 15.3.2-1	Daily[19]

A.1

Prior to reactor criticality after each removal of

A.11/L.3

A.10

LA.1

L.6/ L.2

M.6

A.12

A.16

< See LCO 3.5.2 >

B

Amendmen 195/200

Justification For Deviations - NUREG-1431 Section 3.01.05

24-May-00

JFD Number	JFD Text
01 Rev. A	The CTS requires all shutdown and control rods to be within an alignment limit which is based on demanded control rod position. Between 30 and 215 step of demand, the limit is 12 steps with the limit becoming 24 steps at less than or equal to 30 and greater than or equal to 215 steps of demand. NUREG 1431 presents the control rod alignment limit as a fixed value of 12 steps for the entire range of demand position. The proposed ITS LCO has been rephrased to require the control rod alignment to be maintained within limits, with the variable limit specified in SR 3.1.5.1. Complementary changes have been proposed to the Bases and examples of rod misalignment within consistent with the requirements. This change is necessary to retain the variable alignment limit contained in the CTS while maintaining a concise LCO statement. Variable rod alignment limits are required based on non-linearities that exist in the analog rod position indication system at Point Beach. The expanded limits for rod alignment at the extreme ends of control rod demanded position are acceptable based on the relatively low rod worth and peaking factors in this range.

ITS:	NUREG:
B 3.01.04	B 3.01.05
LCO 3.01.04	LCO 3.01.05
SR 3.01.04.01	SR 3.01.05.01

Justification For Deviations - NUREG-1431 Section 3.01.05

24-May-00

JFD Number	JFD Text
02 Rev. A	The NUREG LCO 3.1.5 specifies the instrumentation necessary for LCO compliance within the LCO statement, requiring "indicated rod positions". By stating that the LCO requires "indicated rod position", an inoperable rod position indicator would result in non-compliance with this LCO statement, even though the Actions contined in ITS LCO 3.1.7 provide appropriate compensitory Action for the loss of an indicator. This change simplifies the LCO presentation while retaining the CLB variable rod alignment limits as addressed in Justification for Deviation 1 of this Section. LCO 3.1.5 ensures control rod alignment is maintained within the limit. The prefered means of determining indicated rod position is through the use of the rod position indicator system, however alternate means are addressed in ITS LCO 3.1.7 if a position indicator is inoperable. The CTS contains this concept in CTS 15.3.10.B.1.b.1 which establishes the Action for a control rod which has been determined not to be within its alignment limits based on the inoperability of the position indicator itself. Although the specific instrumentation required for LCO compliance (i.e. the individual indicated rod position) is deleted from the LCO statement, the ITS will continue to require that control rods be maintained within their alignment limits in ITS LCO 3.1.4 while establishing the means of determining rod position in ITS LCO 3.1.7. This change is consistent with the CTS. Changes to the LCO made by TSTF 107, Revision 4, are not applicable with these proposed changes. **ITS:** B 3.01.04 — **NUREG:** B 3.01.05 LCO 3.01.04 — LCO 3.01.05
03 Rev. A	NUREG 1431 requires verification of steady state FQ (SR 3.2.1.1) within 72 hours of one control rod being found to be outside its alignment limits. The ITS will require verification of the steady state FQ limit (FQC(Z)), and the transient FQ limit (FQW(Z)) within 72 hours of one control rod being found to be outside its alignment limits. As discussed in Description of Change L.04 of LCO 3.2.1, the surveillance methodology at Point Beach, by which FQ is determined uses Relaxed Axial Offset Control. Under this methodology, FQ is approximated using two independent limits FQC(Z) and FQW(Z), which are verified in Surveillance Requirements SR 3.2.1.1 and 3.2.1.2 in the proposed ITS. FQC(Z) is the actual measured heat flux at equilibrium conditions, corrected for measurement and manufacturing tolerances, and FQW(Z) is FQC(Z) corrected for projected worst case heat flux redistributions. Verification of the steady state limit alone does not ensure that the transient limit is met. Therefore, adding a requirement to verify that the transient limit (FQW(Z) - SR 3.2.1.2) is met as well as the steady state limit (FQC(Z) - SR 3.2.1.1) is consistent with verification that the overall FQ limit is maintained as required by the CTS. This implements approved TSTF 314, Revision 0. **ITS:** B 3.01.04 — **NUREG:** B 3.01.05 LCO 3.01.04 COND B RA B.2.4 — LCO 3.01.05 COND B RA B.2.4

Justification For Deviations - NUREG-1431 Section 3.01.05

24-May-00

JFD Number	JFD Text
04 Rev. A	Brackets have been removed and the appropriate plant specific information has be input. Appropriate changes made within the Bases text as appropriate. **ITS:** B 3.01.04, SR 3.01.04.03 **NUREG:** B 3.01.05, SR 3.01.05.03
05 Rev. A	Reference to the General Design Criteria (GDC) of 10 CFR 50 Appendix A has been deleted from the Bases of the Technical Specifications, substituting reference to the appropriate section of the FSAR which specifies the Point Beach design criteria. Point Beach was constructed and licensed prior to the GDC being issued. The Point Beach construction permit was issued prior to the GDCs being issued in 1971. Point Beach was designed and constructed utilizing the 1967 proposed GDCs. Accordingly, reference has been provided to the appropriate criteria and section of the Point Beach FSAR which provides explanation of Point Beach's design basis. In addition, References 5 through 7 of the NUREG Bases References Section are not necessary, as the information which is tied to these references is contained in the same Section of the FSAR previously stated as Reference 4. **ITS:** B 3.01.04 **NUREG:** B 3.01.05
06 Rev. A	The Bases for NUREG 1431 contains a generic description of the number of control rod groups per bank. This description has been replaced with information reflective of the Point Beach design. **ITS:** B 3.01.04 **NUREG:** B 3.01.05
07 Rev. A	NUREG 1431 refers to the equipment used to provide individual rod position indication as being digital, while the equipment installed and used at Point Beach is analog. Accordingly, the description of the equipment and terminology used in the proposed ITS has been alter to reflect Point Beach's design. **ITS:** B 3.01.04 **NUREG:** B 3.01.05

Justification For Deviations - NUREG-1431 Section 3.01.05

24-May-00

JFD Number	JFD Text	
08 Rev. A	NUREG LCO 3.1.5 requires all control rods to be operable and within their alignment limits. Control rod alignment is verified by SR 3.1.5.1, while operability is defined by SR 3.1.5.2 and SR 3.1.5.3 which verify control rod freedom of movement (trippability) and control rod drop time. NUREG 1431 contains Conditions and Required Actions to address control rod untrippability and misalignment, but does not contain a Condition to address rod drop times that are out of limits. While rod drop timing is required to be performed prior to the reactor being made critical, it is not inconceivable that a control rod could be found to be outside of its rod drop time with the reactor critical, for which there is no Condition specified. Condition A has been rewritten to be applied to inoperable control rods so that it will encompass both untrippable control rods and control rods with excessive drop times. This change has been made to assure that shutdown margin is verified in addition to requiring a timely plant shutdown, as application of ITS LCO 3.0.3 alone would not require verification of shutdown margin and correction of shutdown margin if required. Complementary Bases changes have been made to reflect this change. These changes are consistent with approved TSTF 107, Revision 4.	
	ITS:	**NUREG:**
	B 3.01.04	B 3.01.05
09 Rev. B	Not Used.	
	ITS:	**NUREG:**
	N/A	N/A
10 Rev. A	Point Beach uses the related axial offset methodology for determining compliance with the FQ heat flux hot channel factor. As such, reference to FQ has been revised to reflect FQW(Z) and FQC(Z) as stipulated in ITS LCO 3.2.1. This change is consistent with approved TSTF 314, Revision 0.	
	ITS:	**NUREG:**
	B 3.01.04	B 3.01.05

Approved TSTF 136

3.1 REACTIVITY CONTROL SYSTEMS

3.1.~~5~~ 4 Rod Group Alignment Limits

Approved TSTF 136 2 Approved TSTF 11

LCO 3.1.~~5~~ All shutdown and control rods shall be OPERABLE, with ~~all~~ individual ~~indicated~~ rod positions within ~~12 steps of their group step counter demand position~~ limits.

1

APPLICABILITY: MODES 1 and 2.

B RAI 3.1.5-1

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more rod(s) ~~untrippable~~ inoperable. 8	A.1.1 Verify SDM ~~is ≥ [1.6]% Δk/k~~.	1 hour
	OR	
	A.1.2 Initiate boration to restore SDM to within limit.	1 hour
	AND	
	A.2 Be in MODE 3.	6 hours
B. One rod not within alignment limits.	B.1 Restore rod to within alignment limits.	1 hour
	OR	
	B.2.1.1 Verify SDM ~~is ≥ [1.6]% Δk/k~~.	1 hour
	OR	

B RAI 3.1.5-1

(continued)

to be within the limits provided in the COLR

Approved TSTF 9

Approved TSTF 136

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
B. (continued)	B.2.1.2 Initiate boration to restore SDM to within limit.	1 hour
	AND	
	B.2.2 Reduce THERMAL POWER to ≤ 75% RTP.	2 hours
	AND	
	B.2.3 Verify SDM ~~is ≥ [1.6]% Δk/k~~ to be within the limits provided in the COLR.	Once per 12 hours
	AND	
	B.2.4 Perform SR 3.2.1.1. and SR 3.2.1.2	72 hours
	AND	
	B.2.5 Perform SR 3.2.2.1.	72 hours
	AND	
	B.2.6 Re-evaluate safety analyses and confirm results remain valid for duration of operation under these conditions.	5 days
C. Required Action and associated Completion Time of Condition B not met.	C.1 Be in MODE 3.	6 hours

Approved TSTF 9

3

B

RAI 3.1.5-1

(continued)

Approved TSTF 136

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
D. More than one rod not within alignment limit.	D.1.1 Verify SDM ~~is ≥ [1.6]% Δk/k~~. to be within the limits provided in the COLR (Approved TSTF 9)	1 hour
	OR	
	D.1.2 Initiate boration to restore required SDM to within limit.	1 hour
	AND	
	D.2 Be in MODE 3.	6 hours

B RAI 3.1.5-1

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
SR 3.1.~~5~~ 4.1 ~~Verify individual rod positions within alignment limit.~~ (Approved TSTF 136; Replace with Insert 3.1.5-01 — 1)	12 hours ~~AND Once within 4 hours and every 4 hours thereafter when the rod position deviation monitor is inoperable~~

(continued)

Approved TSTF 110

BASES

Approved TSTF 136 4

LCO (continued)

RCCAs and banks maintain the correct power distribution and rod alignment.

The requirement to maintain t he rod alignment to within plus or minus 12 steps is conservative. The minimum misalignment assumed in safety analysis is 24 steps (15 inches), and in some cases a total misalignment from fully withdrawn to fully inserted is assumed. Failure to meet the requirements of this LCO may produce unacceptable power peaking factors and LHRs, or unacceptable SDMs, all of which may constitute initial conditions inconsistent with the safety analysis.

APPLICABILITY

The requirements on RCCA OPERABILITY and alignment are applicable in MODES 1 and 2 because these are the only MODES in which neutron (or fission) power is generated, and the OPERABILITY (i.e., trippability) and alignment of rods have the potential to affect the safety of the plant. In MODES 3, 4, 5, and 6, the alignment limits do not apply because the control rods are bottomed and the reactor is shut down and not producing fission power. In the shutdown MODES, the OPERABILITY of the shutdown and control rods has the potential to affect the required SDM, but this effect can be compensated for by an increase in the boron concentration of the RCS. See LCO 3.1.1, "SHUTDOWN MARGIN (SDM) ~~$-T_{avg} > 200°F$,~~" for SDM in MODES 3, 4, and 5 and LCO 3.9.1, "Boron Concentration," for boron concentration requirements during refueling.

B RAI 3.1.5-1

MODE 2 with Keff < 1.0, and

Approved TSTF 136

ACTIONS

A.1.1 and A.1.2

inoperable 8

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

In this situation, SDM verification must include the worth

Approved TSTF 136 4

ACTIONS (continued)

of the untrippable rod, as well as a rod of maximum worth.

A.2 inoperable 8

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

The allowed Completion Time is reasonable, based on operating experience, for reaching MODE 3 from full power conditions in an orderly manner and without challenging plant systems. B RAI 3.1.5-1

B.1

When a rod becomes misaligned, it can usually be moved and is still trippable. If the rod can be realigned within the Completion Time of 1 hour, local xenon redistribution during this short interval will not be significant, and operation may proceed without further restriction.

An alternative to realigning a single misaligned RCCA to the group average position is to align the remainder of the group to the position of the misaligned RCCA. However, this must be done without violating the bank sequence, overlap, and insertion limits specified in LCO 3.1.~~6~~ 5 "Shutdown Bank Insertion Limits," and LCO 3.1.~~7~~ 6 "Control Bank Insertion Limits." The Completion Time of 1 hour gives the operator sufficient time to adjust the rod positions in an orderly manner.

Approved TSTF 136

B.2.1.1 and B.2.1.2

With a misaligned rod, SDM must be verified to be within limit or boration must be initiated to restore SDM to within limit.

In many cases, realigning the remainder of the group to the misaligned rod may not be desirable. For example, realigning control bank B to a rod that is misaligned

Approved TSTF 136

ACTIONS (continued)

Approved TSTF 331

Insert 3.1.5-3

is available to perform evaluations of accident analysis to determine that core limits will not be exceed ed during a Design Basis Event for the duration of operation under these conditions. A Completion Time of 5 days is sufficient time to obtain the required input data and to perform the analysis.

D. Approved TSTF 15

~~C.~~1.1 and ~~C.~~1.2

More than one control rod becoming misaligned from its group average position is not expected, and has the potential to reduce SDM. Therefore, SDM must be evaluated. One hour allows the operator adequate time to determine SDM. Restoration of the required SDM, if necessary, requires increasing the RCS boron concentration to provide negative reactivity, as described in the Bases of LCO 3.1.1. The required Completion Time of 1 hour for initiating boration is reasonable, based on the time required for potential xenon redistribution, the low probability of an accident occurring, and the steps required to complete the action. This allows the operator sufficient time to align the required valves and start the boric acid pumps. Boration will continue until the required SDM is restored.

~~C.~~2 D. Approved TSTF 15

If more than one rod is found to be misaligned or becom es misaligned because of bank movement, the unit conditions fall outside of the accident analysis assumptions. Since automatic bank sequencing would continue to cause misalignment, the unit must be brought to a MODE or Condition in which the LCO requirements are not applicable. To achieve this status, the unit must be brought to at least MODE 3 within 6 hours.

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

B RAI 3.1.5-1

Approved TSTF 15

C.1 ~~D.1~~

When Required Actions cannot be completed within their Completion Time, the unit must be brought to a MODE or

Approved TSTF 136

BASES

Approved TSTF 136 → 4

ACTIONS (continued)

Condition in which the LCO requirements are not applicable. To achieve this status, the unit must be brought to a t least MODE 3 within 6 hours, which obviates concerns about the development of undesirable xenon or power distributions. The allowed Completion Time of 6 hours is reasonable, based on operating experience, for reaching MODE 3 from full power conditions in an orderly manner and without challenging the plant systems.

B RAI 3.1.5-1

Approved TSTF 136 → 4

SURVEILLANCE REQUIREMENTS

SR 3.1.5.1

Verification that individual rod positions are within alignment limits at a Frequency of 12 hours provides a history that allows the operator to detect a rod that is beginning to deviate from its expected position. If the rod position deviation monitor is inoperable, a Frequency of 4 hours accomplishes the same goal. The specified Frequency takes into account other rod position information that is continuously available to the operator in the control room, so that during actual rod motion, deviations can immediately be detected.

Approved TSTF 110

Approved TSTF 136

SR 3.1.5.2

4

Verifying each control rod is OPERABLE would require that each rod be tripped. However, in MODES 1 and 2, tripping each control rod would result in radial or axial power tilts, or oscillations. Exercising each individual control rod every 92 days provides increased confidence that all rods continue to be OPERABLE without exceeding the alignment limit, even if they are not regularly tripped. Moving each control rod by 10 steps will not cause radial or axial power tilts, or oscillations, to occur. The 92 day Frequency takes into consideration other information available to the operator in the control room and SR 3.1.5.1, which is performed more frequently and adds to the determination of OPERABILITY of the rods. Between required performances of SR 3.1.5.2 (determination of control rod OPERABILITY by movement), if a control rod(s) is discovered to be immovable, but remains trippable and aligned, the control rod(s) is considered to be OPERABLE. At any time, if a

8

B RAI 3.1.5-1

Approved TSTF 136 → 4

No Significant Hazards Considerations - NUREG-1431 Section 3.01.05

24-May-00

NSHC Number	NSHC Text
A Rev. A	In accordance with the criteria set forth in 10 CFR 50.92, PBNP has evaluated this proposed Technical Specifications change and determined it does not represent a significant hazards consideration. The following is provided in support of this conclusion. 1. Does the change involve a significant increase in the probability or consequences of an accident previously evaluated? The proposed change involves reformatting and rewording of the current Technical Specifications. The reformatting and rewording process involves no technical changes to existing requirements. As such, this change is administrative in nature and does not impact initiators of analyzed events or assumed mitigation of accident or transient events. Therefore, this change does not increase 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 require a physical alteration of the plant (no new or different type of equipment will be installed) or changes in parameters governing normal plant operation. The proposed change will not impose any new or eliminate any old requirements. Thus, this change does not create the possibility of a new or different kind of accident from any accident previously evaluated. 3. Does this change involve a significant reduction in a margin of safety? The proposed change will not significantly reduce the margin of safety because it has no impact on any safety analysis assumptions. This change is administrative. As such, there is no technical change to the requirements and, therefore, there is no reduction in the margin of safety.

NSHC Number	NSHC Text
L.01 Rev. A	In accordance with the criteria set forth in 10 CFR 50.92, PBNP has evaluated this proposed Technical Specifications change and determined it does not represent a significant hazards consideration. The following is provided in support of this conclusion. 1. Does the change involve a significant increase in the probability or consequences of an accident previously evaluated? The proposed change will relax the Technical Specification Required Actions requiring initiation of boration versus restoration of SDM. While SDM is assumed in various analyses, failing to meet the SDM limit is not an accident precursor. The proposed change does not involve any physical alteration of plant systems, structures or components, changes in parameters governing normal plant operation, or methods of operation. Accordingly, there will be no significant change in the probability of accidents previously evaluated. The additional time allowed for this change to enact restoration actions does not represent an increase in the consequences of accidents previously evaluated, as the plant condition during this extended period is the same as those which are currently allowed for a finite period before requiring a unit shutdown, and a unit shutdown in itself will not directly restore SDM. Accordingly, the consequences are not increased as a result of this change. 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 any physical alteration of plant systems, structures or components, nor does it alter parameters governing normal plant operation. The proposed change does not introduce a new mode of operation. Therefore, the possibility of a new or different kind of accident from any accident previously evaluated is not created. 3. Does this change involve a significant reduction in a margin of safety? Relative to an untrippable control rod, the ITS requires the unit to be placed into Mode 3 within six hours, where as the CTS could have allowed continued operation with a single untrippable control rod. Multiple misaligned control rods, will also require the unit to be placed into Mode 3 within six hours. By requiring a unit shutdown, in parallel with the initiation of boration to restore SDM, all the appropriate actions necessary to place the unit into a safe condition continue to be required. Relative to single control rod misalignment, all analyses are performed assuming maximum control rod misalignment. Allowing more realistic action for restoration of SDM will continue to be enveloped by the accident analyses. Accordingly, the proposed change does not involve a significant reduction in a margin of safety.

NSHC Number	NSHC Text
L.02 Rev. A	In accordance with the criteria set forth in 10 CFR 50.92, PBNP has evaluated this proposed Technical Specifications change and determined it does not represent a significant hazards consideration. The following is provided in support of this conclusion. 1. Does the change involve a significant increase in the probability or consequences of an accident previously evaluated? The proposed change relaxes the specified testing conditions for the performance of rod drop timing tests from 540 degrees to 500 degrees. The purpose of specifying a minimum temperature for rod drop timing is to establish test conditions representative of an operating reactor. The probability of an accident is not increased by this change as the testing conditions are independent of event probabilities. The test conditions specified are still representative operating conditions. The establishment of this condition will continue to provide adequate assurance of component operability. In providing this assurance, the consequences of analyzed events have not been significantly increased. 2. Does the change create the possibility of a new or different kind of accident from any accident previously evaluated? The proposed change does not require a physical alteration of the plant (no new or different type of equipment will be installed). Testing parameters specified are still representative of plant conditions expected during operation. The proposed conditions will still provide adequate assurance of continued operability, while not establishing any new or unique failure modes. Thus, this change does not create the possibility of a new or different kind of accident from any accident previously evaluated. 3. Does this change involve a significant reduction in a margin of safety? The control rods will continue to be tested under conditions representative of operating conditions. The minimum temperature for testing, while lower than currently allowed, will result in more conservative measured times, as rod drop times are longer at reduced temperatures due to increased coolant density. Therefore, while establishing increased testing flexibility, the proposed change will not result in a significant reduction in a margin of safety.

NSHC Number	NSHC Text
L.03 Rev. A	In accordance with the criteria set forth in 10 CFR 50.92, PBNP has evaluated this proposed Technical Specifications change and determined it does not represent a significant hazards consideration. The following is provided in support of this conclusion. 1. Does the change involve a significant increase in the probability or consequences of an accident previously evaluated? The Technical Specifications will continue to require testing to verify control rod operability. The performance of control rod drop timing with the reactor coolant temperature greater than 500 degrees has been demonstrated to be sufficient to verify control rod operability. There are no credible failure mechanisms that would exist solely under cold conditions during rod drop testing nor has there been an occurrence of a control rod failing to trip under cold conditions alone. Accordingly, the proposed change does not significantly increase the probability or consequences of 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 require a physical alteration of the plant (no new or different type of equipment will be installed) or changes in parameters governing normal plant operation. The proposed change will still provide for testing under conditions representative of reactor operation, and will continue to provide assurance of control rod operability. Thus, this change does not create the possibility of a new or different kind of accident from any accident previously evaluated. 3. Does this change involve a significant reduction in a margin of safety? The testing conditions retained in the Technical Specifications for rod drop testing are adequate to provide sufficient assurance of rod operability. Cold drop testing is a good practice for verification of control rod trippability prior to plant heat up where rod drop timing is performed. There are no credible failure mechanisms that would exist solely under cold conditions nor has there been an occurrence of a control rod failing to trip under cold conditions alone. Satisfactory demonstration of control rod trippability in a hot condition is sufficient to provide adequate assurance of function prior to entry into the Mode of Applicability for control rods. Based on the above, this change does not represent a significant reduction in a margin of safety.
L.04 Rev. B	Not Used.

NSHC Number	NSHC Text
L.05 Rev. A	In accordance with the criteria set forth in 10 CFR 50.92, PBNP has evaluated this proposed Technical Specifications change and determined it does not represent a significant hazards consideration. The following is provided in support of this conclusion. 1. Does the change involve a significant increase in the probability or consequences of an accident previously evaluated? The proposed change involves deletion of a non-routine surveillance from the Technical Specifications which is based on the inoperability of a non-safety related alarm. The rod alignment limit alarm (initiated by the on-line computer) does not provide any safety function nor does it input to any protection circuits. Inoperability of the alarm in and of itself does not lead to a control rod misalignment. Alarm inoperability represents a reduction in monitoring capability for a condition (actual control rod misalignment) which rarely occurs. The on-line computer alarm merely provides a non-safety means of alerting personnel to a condition which does not comply with an LCO requirement. Control rod alignment limits are required to be routinely verified once every 12 hours by the proposed ITS. Deletion of the increased surveillance and conditional frequencies (logged once per hour, after load changes in excess of 10% power, and after rod motion in excess of 30 steps with an inoperable alarm) does not alleviate the responsibility of the license to be vigilant of plant conditions and LCO compliance. The inoperability and increased surveillance frequency are not linked to any accident precursors, and accordingly have no impact on the probability of events. The on-line computer (rod alignment limit alarm) is not assumed to function in the mitigation of any event, nor is the increased surveillance interval assumed. Accordingly the consequences of previously evaluated accidents are also unaffected. Therefore, this change does not involve a significant increase in the probability or consequences of an accident previously evaluated. 2. Does the change create the possibility of a new or different kind of accident from any accident previously evaluated? The proposed change does not require a physical alteration of the plant (no new or different type of equipment will be installed) or changes in parameters governing normal plant operation. LCO compliance is still required. This change deletes non-routine surveillances based on other means which readily alert personnel to the potential of an LCO non-compliance. The proposed change will not impose any different operational configurations. Thus, this change does not create the possibility of a new or different kind of accident from any accident previously evaluated. 3. Does this change involve a significant reduction in a margin of safety? The proposed changes do not alter any assumed conditions or limitation in any previously evaluated accidents. Therefore the margin of safety is unaffected by this change.

NSHC Number	NSHC Text
L.06 Rev. A	In accordance with the criteria set forth in 10 CFR 50.92, PBNP has evaluated this proposed Technical Specifications change and determined that it does not represent a significant hazards consideration. The following is provided in support of this conclusion. 1. Does the change involve a significant increase in the probability or consequences of a accident previously evaluated? The proposed change continues to ensure that rod drop times remain within limits. The capability of control rods to drop in response to a trip signal is not a factor in determining the probability of an accident. The capability of the control rods to drop in response to a valid trip signal is a factor in determining the consequences of the accident. The proposed Specifications require that all applicable surveillances be met in order to consider the control rods operable. One of the surveillances verifies the control rods ability to drop into the core within required time limits. As the Specifications continue to provide assurance that the control rods will drop within time limits as required by the analyses, the consequences of an accident previously evaluated are not increased. 2. Does the change create the possibility of a new or different kind of accident from any accident previously evaluated? The proposed change, in conjunction the Technical Specification requirement that all applicable Surveillances be met in order to consider the control rods operable, ensures operation within the bounds of the existing analyses. No new systems or equipment are being installed, nor is the operation of existing equipment or systems altered by this change. Therefore, this change cannot create a new or different kind of accident form any accident previously evaluated. 3. Does the change involve a significant reduction in a margin of safety? The proposed change does not alter the acceptance limits for rod drop testing. Technical Specifications continue to require operability of the control rods and ensure that the applicable Surveillance is adequate and performed at a periodicity necessary to provide assurance of operability. Rod drop testing will still be required following all maintenance that could affect rod drop times. Therefore, operation in accordance with the proposed change does not result in a reduction of a margin of safety.

NSHC Number	NSHC Text
L.07 Rev. A	In accordance with the criteria set forth in 10 CFR 50.92, PBNP has evaluated this proposed Technical Specifications change and determined it does not represent a significant hazards consideration. The following is provided in support of this conclusion. 1. Does the change involve a significant increase in the probability or consequences of an accident previously evaluated? The proposed change does not involve any physical alteration of plant systems, structures or components, changes in parameters governing normal plant operation, or methods of operation. The proposed change extends the surveillance frequency for rod alignment verifications from "each shift" (nominally 8 hours) to 12 hours. This relaxation is acceptable, because the 12 hour Frequency takes into account other rod position information that is continuously available to the operator in the control room, so that during actual rod motion, deviations can immediately be detected. Therefore, this change does not increase 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 any physical alteration of plant systems, structures or components, nor does it alter parameters governing normal plant operation. The proposed change does not introduce a new mode of operation or alter the method of normal plant operation. Therefore, the possibility of a new or different kind of accident from any accident previously evaluated is not created. 3. Does this change involve a significant reduction in a margin of safety? There are no margins of safety related to safety analyses that are dependent upon the proposed change. The requirements will continue to assure that limiting conditions for the rod alignment verifications are properly maintained. Therefore, this change does not involve a reduction in a margin of safety.

NSHC Number	NSHC Text
LA Rev. A	In accordance with the criteria set forth in 10 CFR 50.92, PBNP has evaluated this proposed Technical Specifications change and determined it does not represent a significant hazards consideration. The following is provided in support of this conclusion. 1. Does the change involve a significant increase in the probability or consequences of an accident previously evaluated? The proposed change relocates requirements from the Technical Specifications to the Bases, FSAR, or other plant controlled documents. The Bases and FSAR will be maintained using the provisions of 10 CFR 50.59. In addition to 10 CFR 50.59 provisions, the Technical Specifications Bases are subject to the change process in the Administrative Controls Chapter of the ITS. Plant procedures and other plant controlled documents are subject to controls imposed by plant administrative procedures, which endorse applicable regulations and standards. Changes to the Bases, FSAR, or other plant controlled documents will be evaluated in accordance with the requirements of the Bases Control Program in Chapter 5.0 of the ITS, 10 CFR 50.59, or plant administrative processes. Therefore, no increase in the probability or consequences of an accident previously evaluated will be allowed. 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 require a physical alteration of the plant (no new or different type of equipment will be installed) or changes in parameters governing normal plant operation. The proposed change will not impose any different requirements and adequate control of the information will be maintained. Thus, this change does not create the possibility of a new or different kind of accident from any accident previously evaluated. 3. Does this change involve a significant reduction in a margin of safety? The proposed change will not reduce a margin of safety because it has no impact on any safety analysis assumptions. In addition, the requirements to be moved from the Technical Specifications to the Bases, FSAR, or other plant controlled documents are as they currently exist. Future changes to the requirements in the Bases, FSAR, or other plant controlled documents will be evaluated in accordance with the requirements of 10 CFR 50.59, the Bases Control Program in Chapter 5.0 of the ITS, or the applicable plant process and no reduction in a margin of safety will be allowed.

No Significant Hazards Considerations - NUREG-1431 Section 3.01.05

24-May-00

NSHC Number	NSHC Text
M Rev. A	In accordance with the criteria set forth in 10 CFR 50.92, PBNP has evaluated this proposed Technical Specifications change and determined it does not represent a significant hazards consideration. The following is provided in support of this conclusion. 1. Does the change involve a significant increase in the probability or consequences of an accident previously evaluated? The proposed change provides more restrictive requirements for operation of the facility. These more stringent requirements do not result in operation that will increase the probability of initiating an analyzed event and do not alter the assumptions relative to the mitigation of an accident or transient event. These more restrictive requirements continue to ensure process variables, structures, systems and components are maintained consistent with the safety analyses. Therefore, this change does not increase 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 require a physical alteration of the plant (no new or different type of equipment will be installed) or changes in parameters governing normal plant operation. The proposed change does impose different requirements. However, these changes are consistent with assumptions made in the safety analysis. Thus, this change does not create the possibility of a new or different kind of accident from any accident previously evaluated. 3. Does this change involve a significant reduction in a margin of safety? The imposition of more restrictive requirements either has no affect on or increases the margin of safety. Each change is providing additional restrictions to enhance plant safety. These changes are consistent with the safety analysis. Therefore, this change does not involve a reduction in a margin of safety.

3.1 REACTIVITY CONTROL SYSTEMS

3.1.4 Rod Group Alignment Limits

LCO 3.1.4 All shutdown and control rods shall be OPERABLE, with individual rod positions within limits.

APPLICABILITY: MODES 1 and 2. B RAI 3.1.5-1

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. One or more rod(s) inoperable.	A.1.1	Verify SDM to be within the limits provided in the COLR.	1 hour
	OR		
	A.1.2	Initiate boration to restore SDM to within limit.	1 hour
	AND		
	A.2	Be in MODE 3.	6 hours B RAI 3.1.5-1
B. One rod not within alignment limits.	B.1	Restore rod to within alignment limits.	1 hour
	OR		
	B.2.1.1	Verify SDM to be within the limits provided in the COLR.	1 hour
		OR	
			(continued)

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
B. (continued)	B.2.1.2 Initiate boration to restore SDM to within limit.	1 hour
	AND	
	B.2.2 Reduce THERMAL POWER to ≤ 75% RTP.	2 hours
	AND	
	B.2.3 Verify SDM to be within the limits provided in the COLR.	Once per 12 hours
	AND	
	B.2.4 Perform SR 3.2.1.1 and SR 3.2.1.2.	72 hours
	AND	
	B.2.5 Perform SR 3.2.2.1.	72 hours
	AND	
	B.2.6 Re-evaluate safety analyses and confirm results remain valid for duration of operation under these conditions.	5 days
C. Required Action and associated Completion Time of Condition B not met.	C.1 Be in MODE 3.	6 hours B RAI 3.1.5-1

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
D. More than one rod not within alignment limit.	D.1.1 Verify SDM to be within the limits provided in the COLR.	1 hour
	OR	
	D.1.2 Initiate boration to restore required SDM to within limit.	1 hour
	AND	
	D.2 Be in MODE 3.	6 hours

B
RAI 3.1.5-1

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.1.4.1	Verify individual rod positions are within the following alignment limits: a. ± 12 steps of demanded position when demand position indication is > 30 steps and < 215 steps; and b. ± 24 steps of demanded position when demand position indication is ≤ 30 steps and ≥ 215 steps.	12 hours
SR 3.1.4.2	Verify rod freedom of movement (trippability) by moving each rod not fully inserted in the core ≥ 10 steps in either direction.	92 days

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.1.4.3	Verify rod drop time of each rod, from the fully withdrawn position, is ≤ 2.2 seconds from the beginning of decay of stationary gripper coil voltage to dashpot entry, with: a. $T_{avg} \geq 500°F$; and b. All reactor coolant pumps operating.	Prior to reactor criticality after each removal of the reactor head

B 3.1 REACTIVITY CONTROL SYSTEMS

B 3.1.4 Rod Group Alignment Limits

BASES

BACKGROUND

The OPERABILITY (i.e., trippability) of the shutdown and control rods is an initial assumption in all safety analyses that assume rod insertion upon reactor trip. Maximum rod misalignment is an initial assumption in the safety analysis that directly affects core power distributions and assumptions of available SDM.

The applicable criteria for these reactivity and power distribution design requirements are FSAR Section 3.2, Reactor Design, FSAR Section 1.3.5, Reactivity Control (Ref. 1 and 2), and 10 CFR 50.46, "Acceptance Criteria for Emergency Core Cooling Systems for Light Water Nuclear Power Plants" (Ref. 3).

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

Limits on rod alignment have been established, and all rod positions are monitored and controlled during power operation to ensure that the power distribution and reactivity limits defined by the design power peaking and SDM limits are preserved.

Rod cluster control assemblies (RCCAs), or rods, are moved by their control rod drive mechanisms (CRDMs). Each CRDM moves its RCCA one step (approximately 5/8 inch) at a time, but at varying rates (steps per minute) depending on the signal output from the Rod Control System.

The RCCAs are divided among control banks and shutdown banks. Each bank may be further subdivided into two groups to provide for precise reactivity control. A group consists of two or more RCCAs that are electrically paralleled to step simultaneously. A bank of RCCAs may consist of one or two groups. When a bank consists of two groups, the groups are moved in a staggered fashion, but always within one step of each other. Control banks A and C and shutdown bank A consist of two groups each while control banks B and D and shutdown bank B consist of a single group.

BASES

BACKGROUND (continued)

The shutdown banks are maintained either in the fully inserted or fully withdrawn position. The control banks are moved in an overlap pattern, using the following withdrawal sequence: When control bank A reaches a predetermined height in the core, control bank B begins to move out with control bank A. Control bank A stops at the position of maximum withdrawal, and control bank B continues to move out. When control bank B reaches a predetermined height, control bank C begins to move out with control bank B. This sequence continues until control banks A, B, and C are at the fully withdrawn position, and control bank D is approximately halfway withdrawn. The insertion sequence is the opposite of the withdrawal sequence. The control rods are arranged in a radially symmetric pattern, so that control bank motion does not introduce radial asymmetries in the core power distributions.

The axial position of shutdown rods and control rods is indicated by two separate and independent systems, which are the Bank Demand Position Indication System (commonly called group step counters) and the Rod Position Indication (RPI) System.

The Bank Demand Position Indication System counts the pulses from the rod control system that moves the rods. There is one step counter for each group of rods. Individual rods in a group all receive the same signal to move and should, therefore, all be at the same position indicated by the group step counter for that group. The Bank Demand Position Indication System is considered highly precise (± 1 step or ± 5/8 inch). If a rod does not move one step for each demand pulse, the step counter will still count the pulse and incorrectly reflect the position of the rod.

The RPI System provides a highly accurate indication of actual rod position, but at a lower precision than the step counters. The RPI is a linear variable differential transformer (LVDT) consisting of primary and secondary coils stacked alternately on a support tube with the control rod drive shaft acting as the core of the transformer. The primary and secondary coils are series connected with the primary coil supplied with AC power from a constant current source. The position of the control rod drive shaft changes the primary to secondary coil magnetic coupling resulting in a variable secondary voltage which is proportional to the position of the drive shaft (control rod). The RPI channel has an indication accuracy of 5% of span (11.5 steps) therefore, the maximum deviation between actual and demanded indication could be 24 steps or approximately 15 inches. Because the rod position indicator system may have a 12 step error when a misalignment of 24 steps is occurring, the Specification allows only an indicated misalignment of 12 steps between 30 and 215 steps. When the bank demand position is greater than or equal to 215 steps, or, less than or equal to 30 steps, the consequences of a misalignment are much less severe. The differential worth of an individual RCCA is less, and the resultant perturbation on power distributions is less than when the bank is in its high differential worth region. At the top and bottom of the core, an indicated 24 step misalignment may be representing an actual misalignment of 36 steps.

BASES

APPLICABLE SAFETY ANALYSES

Control rod misalignment accidents are analyzed in the safety analysis (Ref. 4). The acceptance criteria for addressing control rod inoperability or misalignment are that:

a. There be no violations of:

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

b. The core remains subcritical after accident transients.

Two types of misalignment are distinguished. During movement of a control rod group, one rod may stop moving, while the other rods in the group continue. This condition may cause excessive power peaking. The second type of misalignment occurs if one rod fails to insert upon a reactor trip and remains stuck fully withdrawn. This condition requires an evaluation to determine that sufficient reactivity worth is held in the control rods to meet the SDM requirement, with the maximum worth rod stuck fully withdrawn.

Two types of analysis are performed in regard to static rod misalignment (Ref. 4). With control banks at their insertion limits, one type of analysis considers the case when any one rod is completely inserted into the core. The second type of analysis considers the case of a completely withdrawn single rod from a bank inserted to its insertion limit. Satisfying limits on departure from nucleate boiling ratio in both of these cases bounds the situation when a rod is misaligned from its group by 12 steps.

Another type of misalignment occurs if one RCCA fails to insert upon a reactor trip and remains stuck fully withdrawn. This condition is assumed in the evaluation to determine that the required SDM is met with the maximum worth RCCA also fully withdrawn (Ref. 4).

The Required Actions in this LCO ensure that either deviations from the alignment limits will be corrected or that THERMAL POWER will be adjusted so that excessive local linear heat rates (LHRs) will not occur, and that the requirements on SDM and ejected rod worth are preserved.

Continued operation of the reactor with a misaligned control rod is allowed if the heat flux hot channel factors $F_Q^C(Z)$ and $F_Q^W(Z)$ and the nuclear enthalpy hot channel factor ($F_{\Delta H}^N$) are verified to be within their limits in the COLR and the safety analysis is verified to remain valid. When a control rod is misaligned, the assumptions that are used to determine the rod insertion limits, AFD limits, and quadrant power tilt limits are not preserved. Therefore, the limits may not preserve the

BASES

APPLICABLE SAFETY ANALYSES (continued)

design peaking factors, and $F_Q^C(Z)$, $F_Q^W(Z)$, and $F_{\Delta H}^N$ must be verified directly by incore mapping. Bases Section 3.2 (Power Distribution Limits) contains more complete discussions of the relation of $F_Q^C(Z)$, $F_Q^W(Z)$, and $F_{\Delta H}^N$ to the operating limits.

Shutdown and control rod OPERABILITY and alignment are directly related to power distributions and SDM, which are initial conditions assumed in safety analyses. Therefore they satisfy Criterion 2 of the NRC Policy Statement.

LCO

The limits on shutdown or control rod alignments ensure that the assumptions in the safety analysis will remain valid. The requirements on control rod OPERABILITY ensure that upon reactor trip, the assumed reactivity will be available and will be inserted. The control rod OPERABILITY requirement is satisfied provided the control rod will fully insert within the required rod drop time assumed in the safety analysis. Control rod malfunctions that result in the inability to move a control rod (e.g. lift coil and rod control system logic failures), but do not impact the control rod trippability, do not result in control rod inoperability. The LCO requirements also ensure that the RCCAs and banks maintain the correct power distribution and rod alignment.

The requirement to maintain the rod alignment to within plus or minus 12 steps is conservative. The minimum misalignment assumed in safety analysis is 24 steps (15 inches), and in some cases a total misalignment from fully withdrawn to fully inserted is assumed. Failure to meet the requirements of this LCO may produce unacceptable power peaking factors and LHRs, or unacceptable SDMs, all of which may constitute initial conditions inconsistent with the safety analysis.

APPLICABILITY

The requirements on RCCA OPERABILITY and alignment are applicable in MODES 1 and 2 because these are the only MODES in which neutron (or fission) power is generated, and the OPERABILITY (i.e., trippability) and alignment of rods have the potential to affect the safety of the plant. In MODES 3, 4, 5, and 6, the alignment limits do not apply because the control rods are bottomed and the reactor is shut down and not producing fission power. In the shutdown MODES, the OPERABILITY of the shutdown and control rods has the potential to affect the required SDM, but this effect can be compensated for by an increase in the boron concentration of the RCS. See LCO 3.1.1, "SHUTDOWN MARGIN (SDM)" for SDM in MODE 2 with $k_{eff} < 1.0$, and MODES 3, 4, and 5 and LCO 3.9.1, "Boron Concentration," for boron concentration requirements during refueling. B RAI 3.1.5-1

BASES

ACTIONS

A.1.1 and A.1.2

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

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

A.2

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

The allowed Completion Time is reasonable, based on operating experience, for reaching MODE 3 from full power conditions in an orderly manner and without challenging plant systems. B RAI 3.1.5-1

B.1

When a rod becomes misaligned, it can usually be moved and is still trippable. If the rod can be realigned within the Completion Time of 1 hour, local xenon redistribution during this short interval will not be significant, and operation may proceed without further restriction.

An alternative to realigning a single misaligned RCCA to the group average position is to align the remainder of the group to the position of the misaligned RCCA. However, this must be done without violating the bank sequence, overlap, and insertion limits specified in LCO 3.1.5, "Shutdown Bank Insertion Limits," and LCO 3.1.6, "Control Bank Insertion Limits." The Completion Time of 1 hour gives the operator sufficient time to adjust the rod positions in an orderly manner.

B.2.1.1 and B.2.1.2

With a misaligned rod, SDM must be verified to be within limit or boration must be initiated to restore SDM to within limit.

In many cases, realigning the remainder of the group to the misaligned rod may not be desirable. For example, realigning control bank B to a rod that is misaligned 25 steps from the top of the core would require a

BASES

ACTIONS (continued) significant power reduction, since control bank D must be moved fully in and control bank C must be moved in to approximately 100 to 115 steps.

Power operation may continue with one RCCA misaligned, provided that SDM is verified within 1 hour. The Completion Time of 1 hour represents the time necessary for determining the actual unit SDM and, if necessary, aligning and starting the necessary systems and components to initiate boration.

B.2.2, B.2.3, B.2.4, B.2.5, and B.2.6

For continued operation with a misaligned rod, RTP must be reduced, SDM must periodically be verified within limits, hot channel factors ($F_Q^C(Z)$, $F_Q^W(Z)$, and $F_{\Delta H}^N$) must be verified within limits, and the safety analyses must be re-evaluated to confirm continued operation is permissible.

Reduction of power to 75% RTP ensures that local LHR increases due to a misaligned RCCA will not cause the core design criteria to be exceeded (Ref. 4). The Completion Time of 2 hours gives the operator sufficient time to accomplish an orderly power reduction without challenging the Reactor Protection System.

When a rod is known to be misaligned, there is a potential to impact the SDM. Since the core conditions can change with time, periodic verification of SDM is required. A Frequency of 12 hours is sufficient to ensure this requirement continues to be met.

Verifying that $F_Q^C(Z)$, $F_Q^W(Z)$, and $F_{\Delta H}^N$ are within the required limits ensures that current operation at 75% RTP with a rod misaligned is not resulting in power distributions that may invalidate safety analysis assumptions at full power. The Completion Time of 72 hours allows sufficient time to obtain flux maps of the core power distribution using the incore flux mapping system and to calculate $F_Q^C(Z)$, $F_Q^W(Z)$, and $F_{\Delta H}^N$.

Once current conditions have been verified acceptable, time is available to perform evaluations of accident analysis to determine that core limits will not be exceeded during a Design Basis Event for the duration of operation under these conditions. The accident analyses presented in the FSAR Chapter 14 (Ref. 4) that may be adversely affected will be evaluated to ensure that the analysis results remain valid for the duration of continued operation under these conditions. A Completion Time of 5 days is sufficient time to obtain the required input data and to perform the analysis.

BASES

ACTIONS (continued) C.1

When Required Actions cannot be completed within their Completion Time, the unit must be brought to a MODE or Condition in which the LCO requirements are not applicable. To achieve this status, the unit must be brought to at least MODE 3 within 6 hours, which obviates concerns about the development of undesirable xenon or power distributions. The allowed Completion Time of 6 hours is reasonable, based on operating experience, for reaching MODE 3 from full power conditions in an orderly manner and without challenging the plant systems. B RAI 3.1.5-1

D.1.1 and D.1.2

More than one control rod becoming misaligned from its group average position is not expected, and has the potential to reduce SDM. Therefore, SDM must be evaluated. One hour allows the operator adequate time to determine SDM. Restoration of the required SDM, if necessary, requires increasing the RCS boron concentration to provide negative reactivity, as described in the Bases of LCO 3.1.1. The required Completion Time of 1 hour for initiating boration is reasonable, based on the time required for potential xenon redistribution, the low probability of an accident occurring, and the steps required to complete the action. This allows the operator sufficient time to align the required valves and start the boric acid pumps. Boration will continue until the required SDM is restored.

D.2

If more than one rod is found to be misaligned or becomes misaligned because of bank movement, the unit conditions fall outside of the accident analysis assumptions. Since automatic bank sequencing would continue to cause misalignment, the unit must be brought to a MODE or Condition in which the LCO requirements are not applicable. To achieve this status, the unit must be brought to at least MODE 3 within 6 hours.

The allowed Completion Time is reasonable, based on operating experience, for reaching MODE 3 from full power conditions in an orderly manner and without challenging plant systems. B RAI 3.1.5-1

SURVEILLANCE REQUIREMENTS SR 3.1.4.1

Verification that individual rod positions are within alignment limits at a Frequency of 12 hours provides a history that allows the operator to detect a rod that is beginning to deviate from its expected position.

BASES

SURVEILLANCE REQUIREMENTS (continued)

The specified Frequency takes into account other rod position information that is continuously available to the operator in the control room, so that during actual rod motion, deviations can immediately be detected.

SR 3.1.4.2

Verifying each control rod is OPERABLE would require that each rod be tripped. However, in MODES 1 and 2, tripping each control rod would result in radial or axial power tilts, or oscillations. Exercising each individual control rod every 92 days provides increased confidence that all rods continue to be OPERABLE without exceeding the alignment limit, even if they are not regularly tripped. Moving each control rod by 10 steps will not cause radial or axial power tilts, or oscillations, to occur. The 92 day Frequency takes into consideration other information available to the operator in the control room and SR 3.1.4.1, which is performed more frequently and adds to the determination of OPERABILITY of the rods. Between required performances of SR 3.1.4.2 (determination of control rod OPERABILITY by movement), if a control rod(s) is discovered to be immovable, but remains trippable, the control rod(s) is considered to be OPERABLE. At any time, if a control rod(s) is immovable, a determination of the trippability (OPERABILITY) of the control rod(s) must be made, and appropriate action taken. B RAI 3.1.5-1

SR 3.1.4.3

Verification of rod drop times allows the operator to determine that the maximum rod drop time permitted is consistent with the assumed rod drop time used in the safety analysis. Measuring rod drop times prior to reactor criticality, after reactor vessel head removal, ensures that the reactor internals and rod drive mechanism will not interfere with rod motion or rod drop time, and that no degradation in these systems has occurred that would adversely affect control rod motion or drop time. This testing is performed with all RCPs operating and the average moderator temperature ≥ 500°F to simulate a reactor trip under actual conditions.

This Surveillance is performed during a plant outage, due to the plant conditions needed to perform the SR and the potential for an unplanned plant transient if the Surveillance were performed with the reactor at power.

REFERENCES

1. FSAR, Section 3.2.
2. FSAR, Sections 1.3.5.
3. 10 CFR 50.46.
4. FSAR, Chapter 14.

Description of Changes - NUREG-1431 Section 3.01.06

01-Jun-00

DOC Number	DOC Text

A.01
Rev. A

In the conversion of Point Beach current Technical Specifications (CTS) to the proposed plant specific Improved Technical Specifications (ITS), certain wording preferences or conventions are adopted which do not result in technical changes (either actual or interpretational). Editorial changes, reformatting, and revised numbering are adopted to make the ITS consistent with the Standard Technical Specifications, Westinghouse Plants, NUREG-1431, Revision 1 (i.e., Improved Standard Technical Specifications (ISTS)).

CTS:	ITS:
15.03.10	LCO 3.01.05
15.03.10.D.01	LCO 3.01.05
15.03.10.D.01.a	LCO 3.01.05 COND A
15.03.10.D.01.b	LCO 3.01.05 COND A RA A.2
15.03.10.D.01.c	LCO 3.01.05 COND B LCO 3.01.05 COND B RA B.1
15.04.01 T 15.04.01-01 19	SR 3.01.05.01
15.04.01 T 15.04.01-01 19.A	SR 3.01.05.01

A.02
Rev. B

Not Used.

CTS:	ITS:
N/A	N/A

A.03
Rev. A

The Bases of the current Technical Specifications for this section have been completely replaced by revised Bases that reflect the format and applicable content of PBNP ITS, consistent with the Standard Technical Specifications for Westinghouse Plants, NUREG-1431. The revised Bases are as shown in the PBNP ITS Bases.

CTS:	ITS:
BASES	B 3.01.05

DOC Number	DOC Text
A.04 Rev. A	Note 22 to line item 19 of CTS Table 15.4.1-2 states that shiftly control rod insertion limit channel checks are not required during periods of cold shutdown and refueling, but must be performed prior to reactor criticality if it had not been performed within its previous surveillance interval. This frequency notation is ambiguous in that it does not provide any specific guidance between cold shutdown and reactor critical operations. CTS Specification 15.4.0.1 states that surveillance requirements shall be met when the system or component is required to be operable. The CTS Mode of Applicability for control rod insertion limits has been determined to be equivalent to ITS Mode 1 and 2 with Keff greater than or equal to 1.0 which has been established as the ITS Mode of Applicability as stated in Description of Change A.2 of this LCO. By applying Specification 15.4.0.1, the CTS required mode of performance for this surveillance has been determined to be equivalent to ITS Modes 1 and 2 with Keff greater than or equal to 1.0. ITS SR 3.0.1 establishes the requirement that surveillances must be met when the LCO is applicable. As such, the ITS mode of performance for this surveillance is equivalent to the CTS, making this change administrative. **CTS:** / **ITS:** 15.04.01 T 15.04.01-01 19 (22) / SR 3.01.05.01 15.04.01 T 15.04.01-01 19.A (22) / SR 3.01.05.01
A.05 Rev. A	CTS Table 15.4.1-1 line item 19 requires the performance of a channel check for control rods on a shiftly basis, which has been concluded to be equivalent to the ITS Surveillance Requirements which verify that the shutdown banks are within their insertion limits. The control rod analog and demand position indicators do not provide any protective functions. These channels are used solely for the purpose of verifying that the shutdown bank insertion limits are maintained. A channel check as discussed in CTS Section 15.4.1 is intended to be a simple observation of instrument function, which is fulfilled through verification of these operational parameters. Performance of the proposed ITS surveillances while stated to verify operational limits still encompasses an observation of required channel function and clarifies the intended check. This change is administrative. **CTS:** / **ITS:** 15.04.01 T 15.04.01-01 19 / SR 3.01.05.01 15.04.01 T 15.04.01-01 19.A / SR 3.01.05.01
A.06 Rev. A	The CTS provides an introductory statement (Applicability) which simply states which systems/components are addressed within a given section. This same information while worded differently is contained within the title of each ITS LCO. Accordingly, this change is a change in format with no change in technical requirement. **CTS:** / **ITS:** 15.03.10 APPL / LCO 3.01.05

Description of Changes - NUREG-1431 Section 3.01.06

01-Jun-00

DOC Number	DOC Text
A.07 Rev. A	The CTS provides an introductory statement (Objective) at the beginning of this Section of the Technical Specifications which provide a brief summary of the purpose for this Section. This information is contained in the Bases Section of the ITS. This information does not establish any regulatory requirements for the systems and components addressed within this Section. Accordingly, deletion of this information does not alter any requirement set forth in the Technical Specifications. This change is administrative and consistent with the format and presentation for the ITS as provided in NUREG 1431. **CTS:** 15.03.10 OBJ **ITS:** B 3.01.05
A.08 Rev. A	CTS 15.3.10.A.1 and 2 requires Shutdown Margin (SDM) to be maintained whenever the reactor coolant temperature is less than 350 degrees (15.3.10.A.2) and from 350 degrees to full power (15.3.10.A.1). The requirement to maintain SDM within limits has been moved to several LCOs within the ITS. During critical operation (Mode 1 and Mode 2 with Keff greater than or equal to 1.0) , SDM is assured through the maintenance of rod insertion limits in ITS LCO 3.1.5 and 3.1.6, while in Mode 2 with Keff less than 1.0, and Modes 3, 4, and 5, SDM is assured through the application of ITS LCO 3.1.1. Accordingly, while presented in a different fashion than CTS, the requirement to maintain SDM has been retained in the ITS, making this change administrative. **CTS:** 15.03.10.A.01 **ITS:** LCO 3.01.05

Description of Changes - NUREG-1431 Section 3.01.06

01-Jun-00

DOC Number	DOC Text
L.01 Rev. A	CTS 15.3.10.A.1 requires Shutdown Margin (SDM) to be maintained within limits, with an Action of initiation of boration to restore SDM within 15 minutes, if this limitation is not met. In addition, CTS Action 15.3.10.D.1.a requires Shutdown Margin (SDM) to either be verified to exceed its required value or to be restored by boration within one hour when one or more shutdown bank(s) are found to not be fully withdrawn. Shutdown banks are withdrawn and left in position during reactor critical operation, except during performance of the periodic freedom of movement tests. During performance of this test, the shutdown bank can be readily returned to its required position within the one hour time limit. However, in the unlikely event of a shutdown bank being found outside of its rod insertion limit with SDM not met, the CTS Action to initiate boration within 15 minutes and restore SDM via boration within one hour is not considered to be a viable action. Restoration of SDM would require determination of the SDM deficit, quantification of the amount of boration required, initiation and completion of the boration, and a confirmatory sample to conclude that the required RCS boron concentration was achieved. The proposed ITS will require either verification that SDM is within limits or initiation of boration to restore SDM within one hour. Relaxing the Required Actions to the initiation of boration will allow actions to be initiated to restore SDM (boration if necessary) while increasing available focus to restoring the shutdown bank to its required position, restoring compliance with the LCO and SDM. The CTS and the ITS both require restoration of shutdown bank insertion limits within two hours, which establishes a bounding limit for operation with an insertion limit not met. As such, the maximum time that SDM could be not met is an additional one hour before the initiation of a plant shutdown. The additional one hour allowance is considered acceptable based on the increased focus that will be available to the most appropriate action, which is restoration of the shutdown bank insertion limit.

CTS:	ITS:
15.03.10.A.01	LCO 3.01.05 COND A RA A.1.2
15.03.10.D.01.a	LCO 3.01.05 COND A RA A.1.2

DOC Number	DOC Text
L.02 Rev. A	CTS 15.3.10.D.1 requires the shutdown banks to be fully withdrawn whenever the reactor is critical. Line item 10 of CTS Table 15.4.1-2 requires the control rods to be tested for freedom of movement once every two weeks. The control rod freedom of movement test involves moving several control rods slightly below their rod insertion limits, resulting in entry into the CTS Action which requires the control rods to be returned to their required positions within two hours. The duration that rods are below their insertion limits during the performance of this test is minimal (less than approximately 15 minutes) in comparison to the current two hour restoration time. ITS LCO 3.1.5 is modified by a note which allows the Shutdown Bank Insertion Limit LCO to not be considered applicable during performance of the periodic control rod freedom of movement test. This change is not intended to change the method of surveillance testing but rather reduce the administrative burden associated with tracking entry into and closure of Technical Specification Required Actions when performing routine required Technical Specification Surveillance tests. Elimination of the above referenced administrative burden is a relaxation of current requirements, which is acceptable based on the fact that control rods are typically exercised over a range of travel where the integral rod/bank worth is low, thereby having minimal affect on power distribution and required shutdown margin. Reducing the administrative burden associated with routine testing will allow increased focus on issues of higher safety significance. **CTS:** NEW **ITS:** LCO 3.01.05 NOTE
L.03 Rev. A	CTS Table 15.4.1-1, surveillance frequency S, "each shift", is proposed to become "every 12 hours", in ITS. The nominal Point Beach shift duration is 8 hours. Therefore this change extends the nominal time between performances of these surveillances by 4 hours, resulting in a relaxation of the current requirement. This is acceptable based on other less formal, but more frequent, checks of channels during normal operational use of the displays associated with the LCO required channels, and the low probability of equipment malfunction during the additional (nominal 4 hour) time interval. Proposed ITS SR 3.1.5.1 establishes an acceptable periodicity for verification of that all rods are meet the rod insertion limits. The shutdown banks are manually repositioned by the operator, thus there is no credible means for the shutdown banks to inadvertently violate the insertion limit absent a readily detectable system failure. The operators will be knowledgeable of any change in rod insertion; thus the increased frequency of verification failure of a rod insertion limit alarm is not necessary to ensure the limits are met. **CTS:** 15.04.01 T 15.04.01-01 19 **ITS:** SR 3.01.05.01 **CTS:** 15.04.01 T 15.04.01-01 19.A **ITS:** SR 3.01.05.01 **CTS:** 15.04.01 T 15.04.01-01 S - EACH SHIFT **ITS:** SR 3.01.05.01

Description of Changes - NUREG-1431 Section 3.01.06

01-Jun-00

DOC Number	DOC Text
LA.01 Rev. A	CTS 15.3.10.D contains a requirement to maintain Shutdown Margin within limits (defined in Figure 15.3.10-1) and a requirement to maintain the shutdown banks fully withdrawn (defined as being greater than 225 steps) in addition to referencing the specific Shutdown Margin Limit (SDM) which must be restored if the insertion limits are violated. Rod insertion limits are established primarily to maintain the required SDM which is a cycle specific variable. The SDM limits have been moved to the Core Operating Limits Report. SDM, and control and shutdown rod insertion limits, can be adequately controlled outside of the Technical Specifications since these limitations are calculated using an approved methodology which is ultimately controlled via the methodology's inclusion in the Administrative Control Section of the ITS. Specific Reporting Requirements to notify the NRC when changes are made to the COLR have been proposed consistent with NUREG 1431 and NRC Generic Letter 88-16. This change represents a relaxation of existing requirements. The limits associated with this specification are not required to be in the ITS to provide adequate protection of the public health and safety, since the ITS will still retain a requirement to maintain compliance with the limitation. Changes to the limits will be controlled in accordance with the 10 CFR 50.59 process. Therefore, the level of safety is unaffected by the change.

CTS:	ITS:
15.03.10 F 15.03.10-01	COLR
15.03.10 F 15.03.10-02	COLR
15.03.10.A.01	COLR
15.03.10.D.01.a	LCO 3.01.05 COND A RA A.1.1

DOC Number	DOC Text
M.01 Rev. B	CTS 15.3.10.D.1 states that whenever the reactor is critical that the shutdown banks must be fully withdrawn. This establishes an applicability of reactor critical for this requirement. The proposed ITS for Point Beach specifies an applicability of Modes 1 and 2. Accordingly, the applicability for this requirement in the proposed ITS is more restrictive, since it expands the Mode of applicablility to include reactor subcritical. Line item 19 of table 15.4.1-1, requires a channel check to be performed for the rod position indicators on a shiftly frequency in "all" plant conditions. Table 15.4.1-1 defines "all" plant conditions through reference to Specification 15.1.g, h, and m, which are; 1] Shutdown (Hot, Cold, Refueling, and Shutdown Margin), 2] Power Operations (greater than 2% power), and 3] Low Power Operation (less than or equal to 2% power). As such, defining the applicability of this surveillance in the terms specified in Specification 15.1.g, h, and m are vague and non-prescriptive. Specification 15.4.0.1 states that surveillance requirements shall be met during all times that the system or component is required to be operable. Through applying Specification 15.4.0.1, the applicability of CTS line item 19 of Table 15.4.1-1 would be when the reactor is critical, which is consistent with the safety basis for the Specification. The applicability for this requirement in the proposed ITS is more restrictive, since it expands the Mode of applicablility to include reactor subcritical (Modes 1 and 2).

CTS:	ITS:
15.03.10.D.01	LCO 3.01.05
15.04.01 T 15.04.01-01 19	SR 3.01.05.01
15.04.01 T 15.04.01-01 19.A	SR 3.01.05.01
15.04.01 T 15.04.01-01 ALL	SR 3.01.05.01

A.1

(2) Once per shift check the position of the rods with inoperable RPIs by using excore detectors, or thermocouples, or movable incore detectors;

(3) If the above actions and associated completion times are not met, perform the actions in accordance with TS 15.3.10.B.1.b.

b. If one or more rods with inoperable RPIs have been moved in excess of 24 steps in one direction since the last determination of the rod's position, perform the following actions:

(1) Within four hours check the position of the rods with inoperable RPIs by using excore detectors, or thermocouples, or movable incore detectors;

< See LCO 3.1.8 >

L.2

(2) If the above action and associated completion time is not met, perform the actions in accordance with TS 15.3.10.B.1.b.

Add Note for control rod testing (Insert 3.1.6-02)

c. If bank demand position indication, for one or more banks, is determined to be inoperable, perform the following actions:

(1) Once per shift verify that all RPIs for the affected banks are operable;
AND

(2) Once per shift verify that the most withdrawn rod and the least withdrawn rod of the affected banks are ≤12 steps apart, except when the bank demand position is ≤30 steps or ≥215 steps. In this case, once per shift verify that the most withdrawn rod and the least withdrawn rod of the affected banks are ≤24 steps apart;

LCO 3.1.5

(3) If the above actions and associated completion times are not met, perform the actions in accordance with TS 15.3.10.B.1.b.

D. BANK INSERTION LIMITS

Modes 1 and 2

M.1

LA.1

B

RAI 3.1.6-1

1. ~~When the reactor is critical, the shutdown banks shall be fully withdrawn. Fully withdrawn is defined as a bank position equal to or greater than 225 steps. This definition is applicable to shutdown and control banks.~~

Replace with Insert 3.1.6-01

If this condition is not met, perform the following actions:

the COLR

a. Within one hour verify that the shutdown margin exceeds the applicable value as shown in ~~Figure 15.3.10-2;~~ OR ~~within one hour restore the shutdown margin by boration;~~

L.1

LCO 3.1.5 Cond A RA A.1.1 and A.1.2

initiate boration to restore SDM to within limits.

A.1

TABLE 15.4.1-1 (continued)

NO.	CHANNEL DESCRIPTION	CHECK	CALIBRATE	TEST	PLANT CONDITIONS WHEN REQUIRED	
9.	Steam Generator Flow Mismatch	S(22)	R	Q(1)	ALL	
10.	Steam Generator Pressure	S(16)	R	Q(1)	ALL	
11.	4KV Bus Undervoltage (A01 & A02)					< See LCO 3.3.1 and 3.3.2 >
	-AFW pump actuation	-	R	M(1)	ALL	
	-Reactor Protection actuation	-	R	M(1,2)	ALL	
12.	4KV Bus Underfrequency (A01 & A02)					
	-to Reactor Coolant Pump trip	-	R	-	ALL	
13.	Safeguards Bus Voltage					< See 3.3.5 >
	-Loss of 4KV	S	R	M	ALL	
	-Degraded 4KV	S	R	M	ALL	
	-Loss of 480V	S	R	M	ALL	
14.	120 Vac Instrument Buses	W(6)	-	-	ALL	< See LCO 3.8.9 and 3.8.10 >
15.	Reactor Trip Signal From Turbine					
	-Turbine Autostop	-	-	M(1)	ALL	
	-Turbine Stop Valve	-	-	M(1)	ALL	< See LCO 3.3.1 >
16.	Reactor Trip Signal From SI	-	-	M(1)	ALL	
17.	Feedwater Isolation on SI					
	-MFP Trip on Safety Injection	-	-	R	ALL	< See LCO 3.3.2 and 3.7.3 >
	-MFRV Shutting on Safety Injection	-	-	R	ALL	
18.	Accumulator Level and Pressure	S	R	-	ALL	< See LCO 3.5.1 >
19.	Analog Rod Position	S(8,22)	R	-	ALL	Modes 1 and 2
	-with step counters	S(22)	-	-	ALL	
	-Monitoring by On-Line Computer	(18)	-	-	PWR,HOT S/D	

Replace with Insert 3.1.6-03

A.5/L.3

M.1

B

RAI 3.1.6-1

SR 3.1.5.1

< See LCO 3.1.5>

L.3

NOTATION USED IN TABLE 15.4.1-1

A - Annually
S- Each shift ← L.3
D- Daily
W- Weekly
Q- Quarterly
M- Monthly
P- Prior to reactor criticality if not performed during the previous week.
R- Each refueling interval (18 months)
PWR- Power and Low Power Operation, as defined in Specifications 15.1.h. and 15.1.m.
HOT S/D- Hot Shutdown, as defined in Specification 15.1.g.1.
COLD S/D- Cold Shutdown, as defined in Specification 15.1.g.2.
REF S/D- Refueling Shutdown, as defined in Specification 15.1.g.3.
ALL- All conditions of operation, as defined in Specifications 15.1.g, h and m. ← Mode 1 and 2 ← M.1

Discussed in LCOs which notation is applicable to

< See LCOs; 3.3.1, 3.3.3, and 3.3.2 >

B

RAI 3.1.6-1

NOTES USED IN TABLE 15.4.1-1

(1) Not required during periods of refueling shutdown, but must be performed prior to reactor criticality if it has not been performed during the previous surveillance period.

(2) Tests of the low power trip bistable setpoints which cannot be done during power operations shall be conducted prior to reactor criticality if not done in the previous surveillance interval. ← **< See LCO 3.3.1 >**

(3) Perform test of the isolation valve signal. ← **< See LCO 3.3.2 >**

(4) Perform by means of the moveable incore detector system. ← **< See LCO 3.3.1 >**

(5) Recalibrate if the absolute difference is ≥3 percent. ← **< See Section 3.8 >**

(6) Verification of proper breaker alignment and that the 120 Vac instrument buses are energized. ← **< See Section 3.3 >**

(7) Source check is required prior to initiation of a release. Source check is an assessment of channel response by exposing the detector to a source of increased radiation. Channel check is required shiftly during a release. If monitor or isolation function is discovered inoperable, discontinue release immediately.

(8) Verify that the associated rod insertion limit is not being violated at least once per 4 hours whenever the rod insertion limit alarm for a control bank is inoperable. ← L.3

(9) Test of Narrow Range Pressure, 3.0 psig, -3.0 psig excluded. ← **< See LCO 3.3.2 >**

Justification For Deviations - NUREG-1431 Section 3.01.06

24-May-00

JFD Number	JFD Text	
01 Rev. A	Reference to the General Design Criteria (GDC) of 10 CFR 50 Appendix A has been deleted from the Bases of the Technical Specifications, substituting reference to the appropriate section of the FSAR which specifies the Point Beach design criteria. Point Beach was constructed and licensed prior to the GDC being issued. The Point Beach construction permit was issued prior to the GDCs being issued in 1971. Point Beach was designed and constructed utilizing the 1967 proposed GDCs. Accordingly, reference has been provided to the appropriate criteria and section of the Point Beach FSAR which provides explanation of Point Beach's design basis.	
	ITS:	**NUREG:**
	B 3.01.05	B 3.01.06
02 Rev. A	The brackets have been removed and the proper plant specific information has been provided for reference 3 of the References Section for the Bases of LCO 3.1.6. In addition, the Applicable Safety Analyses Section of the Bases for LCO 3.1.6 has been changed to reference 4 and Reference 4 has been added to the References Section of the Bases to allow the appropriate Section of the FSAR to be listed and referenced. Reference 3 contains the control rod design requirements, while reference 4 contains a broad reference to the Accident Analyses Section of the FSAR which contains the accidents analysis assumption for analyzed events which credit a specific SDM.	
	ITS:	**NUREG:**
	B 3.01.05	B 3.01.06
03 Rev. A	The Bases for Required Action A.1.1 references the Bases for SR 3.1.1.1 to obtain a listing of effects for calculation of SDM when one or more shutdown banks are not within limits. This LCO Action is applicable in Modes 1 and 2 with Keff greater than or equal to 1.0. SR 3.1.1.1 calculates SDM in Mode 2 with Keff < 1.0, and Modes 3, 4, and 5, addressing subcritical conditions. Therefore, the Bases of SR 3.1.1.1 contains effects which are not applicable to an operating reactor. Proposed ITS LCO 3.1.5 Required Action A.1.1 has been modified to list the effects listed from the Bases of SR 3.1.1.1 which are applicable to reactor critical operation.	
	ITS:	**NUREG:**
	B 3.01.05	B 3.01.06
04 Rev. B	Not Used.	
	ITS:	**NUREG:**
	N/A	N/A

Justification For Deviations - NUREG-1431 Section 3.01.06

24-May-00

JFD Number	JFD Text	
05 Rev. A	The Bases for proposed ITS SR 3.1.5.1 has been expanded to include reference to the preferred indication for verifying that shutdown banks are within their insertion limits.	
	ITS:	**NUREG:**
	B 3.01.05	B 3.01.06
06 Rev. A	The proposed Bases has been modified to reflect the Point Beach design. Not all control rod banks consist of two groups of rods as stated in the Bases of NUREG 1431 LCO 3.1.6. Control banks may consist of a single group dependent upon the number of control rods assigned to the bank. Typically control rod banks with four or fewer rods consist of a single group. Any bank that consists of two groups will step the banks within one step of each other.	
	ITS:	**NUREG:**
	B 3.01.05	B 3.01.06

Approved TSTF 136 → 5

3.1 REACTIVITY CONTROL SYSTEMS

3.1.~~6~~ Shutdown Bank Insertion Limits

LCO 3.1.~~6~~ Each shutdown bank shall be within insertion limits specified in the COLR.

5 ← Approved TSTF 136

Approved TSTF 239

APPLICABILITY: MODE 1,
MODE 2 ~~with any control bank not fully inserted~~.

B RAI 3.1.6-1

----------NOTE----------
This LCO is not applicable while performing SR 3.1.~~5~~.2.

Approved TSTF 136 → 4

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more shutdown banks not within limits.	A.1.1 Verify SDM ~~is ≥ [1.6]% Δk/k~~ to be within the limits provided in the COLR.	1 hour
	OR	
	A.1.2 Initiate boration to restore SDM to within limit.	1 hour
	AND	
	A.2 Restore shutdown banks to within limits.	2 hours
B. Required Action and associated Completion Time not met.	B.1 Be in MODE 3.	6 hours

Approved TSTF 9

B RAI 3.1.6-1

BACKGROUND (continued)

Hence, they are not capable of adding a large amount of positive reactivity. Boration or dilution of the Reactor Coolant System (RCS) compensates for the reactivity changes associated with large changes in RCS temperature. The design calculations are performed with the assumption that the shutdown banks are withdrawn first. The shutdown banks can be fully withdrawn without the core going critical. This provides available negative reactivity in the event of boration errors. The shutdown banks are controlled manually by the control room operator. During normal unit operation, the shutdown banks are either fully withdrawn or fully inserted. The shutdown banks must be completely withdrawn from the core, prior to withdrawing any control banks during an approach to criticality. The shutdown banks are then left in this position until the reactor is shut down. They affect core power and burnup distribution, and add negative reactivity to shut down the reactor upon receipt of a reactor trip signal. B RAI 3.1.6-1

APPLICABLE SAFETY ANALYSES

On a reactor trip, all RCCAs (shutdown banks and control banks), except the most reactive RCCA, are assu med to insert into the core. The shutdown banks shall be at or above their insertion limits and available to insert the maximum amount of negative reactivity on a reactor trip signal. The control banks may be partially inserted in the core, as ~~allowed by LCO 3.1.6~~ "Control Bank Insertion Limits." The shutdown bank and control bank insertion limits are established to ensure that a sufficient amount of negative reactivity is available to shut down the reactor and maintain the required SDM (see LCO 3.1.1, "SHUTDOWN MARGIN (SDM) ~~T_{avg} > 200°F," and LCO 3.1.2, "SHUTDOWN MARGIN (SDM) T_{avg} ≤ 200°F"~~) following a reactor trip from full power. The combination of control banks and shutdown banks (less the most reactive RCCA, which is assumed to be fully withdrawn) is sufficient to take the reactor from full power conditions at rated temperature to zero power, and to maintain the required SDM at rated no load temperature (Ref. 3). The shutdown bank insertion limit also limits the reactivity worth of an ejected shutdown rod.

6

Approved TSTF 136

The acceptance criteria for addressing shutdown and control rod bank insertion limits and inoperability or misalignment is that:

BASES

Approved TSTF 136 → 5

APPLICABLE SAFETY ANALYSIS (continued)

a. There be no violations of:

1. specified acceptable fuel design limits, or

2. RCS pressure boundary integrity; and

b. The core remains subcritical after accident transients.

As such, the shutdown bank insertion limits affect safety analysis involving core reactivity and SDM (Ref. ~~3~~ 3). 4

The shutdown bank insertion limits preserve an initial condition assumed in the safety analyses and, as such, satisfy Criterion 2 of the NRC Policy Statement. 2

LCO

The shutdown banks must be within their insertion limits any time the reactor is critical or approaching criticality. This ensures that a sufficient amount of negative reactivity is available to shut down the reactor and maintain the required SDM following a reactor trip. B RAI 3.1.6-1

The shutdown bank insertion limits are defined in the COLR.

APPLICABILITY

The shutdown banks must be within their insertion limits, with the reactor in MODES 1 and 2. ~~The applicability in MODE 2 begins prior to initial control bank withdrawal, during an approach to criticality, and continues throughout MODE 2, until all control bank rods are again fully inserted by reactor trip or by shutdown.~~ This ensures that a sufficient amount of negative reactivity is available to shut down the reactor and maintain the required SDM following a reactor trip. The shutdown banks do not have to be within their insertion limits in MODE 3, unless an approach to criticality is being made. In MODE 3, 4, 5, or 6, the shutdown banks are fully inserted in the core and contribute to the SDM. Refer to LCO 3.1.1 ~~and LCO 3.1.2~~ for SDM requirements in MODES 3, 4, and 5. LCO 3.9.1, "Boron Concentration," ensures adequate SDM in MODE 6. B RAI 3.1.6-1 B RAI 3.1.6-1

Approved TSTF 239 → ~~The applicability in MODE 2 begins prior to initial control bank withdrawal, during an approach to criticality, and continues throughout MODE 2, until all control bank rods are again fully inserted by reactor trip or by shutdown.~~

Approved TSTF 136 → ~~and LCO 3.1.2~~

The Applicability requirements have been modified by a Note indicating the LCO requirement is suspended during

Approved TSTF 136 → 5

BASES

Approved TSTF 136

APPLICABILITY (continued)

4 ~~SR 3.1.6.2~~. This SR verifies the freedom of the rods to move, and requires the shutdown bank to move below the LCO limits, which would normally violate t he LCO.

ACTIONS

A.1.1, A.1.2 and A.2

When one or more shutdown banks is not within insertion limits, 2 hours is allowed to restore the shutdown banks to within the insertion limits. This is necessary because the available SDM may be significantly reduced, with one or more of the shutdown banks not within their insertion limits. Also, verification of SDM or initiation of boration within 1 hour is required, since the SDM in MODES 1 and 2 is ensured by adhering to the control and shutdown bank insertion limits (see LCO 3.1.1). If shutdown banks are not within their insertion limits, then SDM will be verified ~~by performing a reactivity balance calculation, considering the effects listed in the BASES for SR 3.1.1.1.~~

B RAI 3.1.6-1

3 Replace with Insert 3.1.6-02

The allowed Completion Time of 2 hours provides an acceptable time for evaluating and repairing minor problems without allowing the plant to remain in an unacceptable condition for an extended period of time.

B.1

If the shutdown banks cannot be restored to within their insertion limits within 2 hours, the unit must be brought to a MODE where the LCO is not applicable. The allowed Completion Time of 6 hours is reasonable, based on operating experience, for reaching the required MODE from full power conditions in an orderly manner and without challenging plant systems.

Approved TSTF 136

SURVEILLANCE REQUIREMENTS

5 SR 3.1.~~6~~.1

Verification that the shutdown banks are within their insertion limits prior to an approach to criticality ensures that when the reactor is critical, or being taken critical, the shutdown banks will be available to shut down the reactor, and the required SDM will be maintained following a

B RAI 3.1.6-1

Approved TSTF 136 5

BASES

Approved TSTF 136 5

SURVEILLANCE REQUIREMENTS (continued)

reactor trip. This SR and Frequency ensure that the shutdown banks are withdrawn before the control banks are withdrawn during a unit startup.

Since the shutdown banks are positioned manually by the control room operator, a verification of shutdown bank position at a Frequency of 12 hours, after the reactor is taken critical, is adequate to ensure that they are within their insertion limits. Also, the 12 hour Frequency takes into account other information available in the control room for the purpose of monitoring the status of shutdown rods.

B RAI 3.1.6-1

REFERENCES

1. ~~10 CFR 50, Appendix A, GDC 10, GDC 26, and GDC 28.~~
2. 10 CFR 50.46.
3. FSAR, Chapter ~~[15]~~.

FSAR, Section 3.1.

1

Section 3.2

2

4. FSAR, Chapter 14.

Typically, the individual rod position indicators are used to confirm shutdown bank insertion limits.

5

Approved TSTF 136 5

No Significant Hazards Considerations - NUREG-1431 Section 3.01.06

01-Jun-00

NSHC Number	NSHC Text
A Rev. A	In accordance with the criteria set forth in 10 CFR 50.92, PBNP has evaluated this proposed Technical Specifications change and determined it does not represent a significant hazards consideration. The following is provided in support of this conclusion. 1. Does the change involve a significant increase in the probability or consequences of an accident previously evaluated? The proposed change involves reformatting and rewording of the current Technical Specifications. The reformatting and rewording process involves no technical changes to existing requirements. As such, this change is administrative in nature and does not impact initiators of analyzed events or assumed mitigation of accident or transient events. Therefore, this change does not increase 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 require a physical alteration of the plant (no new or different type of equipment will be installed) or changes in parameters governing normal plant operation. The proposed change will not impose any new or eliminate any old requirements. Thus, this change does not create the possibility of a new or different kind of accident from any accident previously evaluated. 3. Does this change involve a significant reduction in a margin of safety? The proposed change will not significantly reduce the margin of safety because it has no impact on any safety analysis assumptions. This change is administrative. As such, there is no technical change to the requirements and, therefore, there is no reduction in the margin of safety.

NSHC Number	NSHC Text
L.01 Rev. A	In accordance with the criteria set forth in 10 CFR 50.92, PBNP has evaluated this proposed Technical Specifications change and determined it does not represent a significant hazards consideration. The following is provided in support of this conclusion. 1. Does the change involve a significant increase in the probability or consequences of an accident previously evaluated? The proposed change will relax the Technical Specification Required Actions requiring initiation of boration versus restoration of SDM. The CTS and the ITS both require restoration of control bank insertion limits within two hours, which establishes a bounding limit for operation with an insertion limit not met. As such, the maximum time that SDM could conceivably not be met is an additional one hour before the initiation of a plant shutdown is required. While SDM and rod position (available worth) are variables assumed in various analyses, the state of not meeting a rod insertion limit is not an accident precursor. The proposed change does not involve any physical alteration of plant systems, structures or components, changes in parameters governing normal plant operation, or methods of operation. Accordingly, there will be no significant change in the probability of accidents previously evaluated. The additional one hour period allowed to be in this condition does not represent an increase in the consequences of accidents previously evaluated, as the plant condition during this extended period is the same as those currently allowed for up to one hour. Accordingly, the consequences are the same during this increased period. Accordingly, this change does not present a significant increase in the consequences of accidents 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 any physical alteration of plant systems, structures or components, nor does it alter parameters governing normal plant operation. The proposed change does not introduce a new mode of operation. Therefore, the possibility of a new or different kind of accident from any accident previously evaluated is not created. 3. Does this change involve a significant reduction in a margin of safety? In the unlikely event of a shutdown bank being found outside of its rod insertion limit with SDM not met, the CTS Action to restore SDM via boration within one hour is not a viable action, which would then result in the initiation of a plant shutdown which is a diversion of resources which should be more appropriately focused on restoration of the shutdown banks insertion limit. The additional one hour allowance is acceptable based on the increased focus that will be available to the most appropriate action which is restoration of the shutdown bank insertion limit. Accordingly, increasing the time allowed before shutdown actions are required to be initiated by an additional hour does not involve a significant reduction in a margin of safety.

NSHC Number	NSHC Text
L.02 Rev. A	In accordance with the criteria set forth in 10 CFR 50.92, PBNP has evaluated this proposed Technical Specifications change and determined it does not represent a significant hazards consideration. The following is provided in support of this conclusion. 1. Does the change involve a significant increase in the probability or consequences of an accident previously evaluated? The Technical Specification control rod freedom of movement test involves moving several control rods slightly below their rod insertion limits, resulting in entry into the CTS Action which requires the control rods to be returned to their required positions within two hours. This change eliminates the need to enter the Technical Specification Required Actions during performance of the test. This change is not intended to alter the method of surveillance testing but rather reduce the administrative burden associated with tracking entry into and closure of Technical Specification Required Actions when performing routine required Technical Specification Surveillance tests. Elimination of administrative burden is not an accident precursor. The proposed change does not involve any physical alteration of plant systems, structures or components, changes in parameters governing normal plant operation, or methods of operation. Accordingly, there will be no significant change in the probability of accidents previously evaluated. Control rods are typically exercised over a range of travel where the integral rod/bank worth is low, thereby having minimal affect on power distribution and required shutdown margin. Therefore, this change does not involve a significant increase in the consequences of accidents 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 any physical alteration of plant systems, structures or components, nor does it alter parameters governing normal plant operation. The proposed change does not introduce a new mode of operation or alter the method of normal plant operation. Therefore, the possibility of a new or different kind of accident from any accident previously evaluated is not created. 3. Does this change involve a significant reduction in a margin of safety? The control rod freedom of movement test involves moving several of the control rods slightly below their rod insertion limits. The duration that rods are below their insertion limits during the performance of this test is minimal (less than approximately 15 minutes). Control rods are typically exercised over a range of travel where the integral rod/bank worth is low, thereby having minimal affect on power distribution and required shutdown margin. This change is not intended to change the method of surveillance testing but rather reduce the administrative burden associated with tracking entry into and closure of Technical Specification Required Actions when performing routine required Technical Specification Surveillance tests. Reducing the administrative burden associated with routine testing will

NSHC Number	NSHC Text
	allow increased focus on issues of higher safety significance. Accordingly, this change does not involve a significant reduction in a margin of safety.
L.03 Rev. A	In accordance with the criteria set forth in 10 CFR 50.92, PBNP has evaluated this proposed Technical Specifications change and determined it does not represent a significant hazards consideration. The following is provided in support of this conclusion. 1. Does the change involve a significant increase in the probability or consequences of an accident previously evaluated? The proposed change does not involve any physical alteration of plant systems, structures or components, changes in parameters governing normal plant operation, or methods of operation. The proposed change extends the surveillance frequency for CHANNEL CHECKS from "each shift" (nominally 8 hours) to 12 hours. This is acceptable because the CHANNEL CHECK supplements less formal, but more frequent, checks of channels during normal operational use of the displays associated with the LCO required channels and because of the unlikelihood of a channel failure during this interval. Therefore, this change does not involve an 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 any physical alteration of plant systems, structures or components, nor does it alter parameters governing normal plant operation. The proposed change does not introduce a new mode of operation or alter the method of normal plant operation. Therefore, the possibility of a new or different kind of accident from any accident previously evaluated is not created. 3. Does this change involve a significant reduction in a margin of safety? There are no margins of safety related to safety analyses that are dependent upon the proposed change. The requirements will continue to assure that limiting conditions for the Control Bank Insertion Limits are properly maintained. Therefore, this change does not involve a reduction in a margin of safety.

No Significant Hazards Considerations - NUREG-1431 Section 3.01.06

01-Jun-00

NSHC Number	NSHC Text
LA Rev. A	In accordance with the criteria set forth in 10 CFR 50.92, PBNP has evaluated this proposed Technical Specifications change and determined it does not represent a significant hazards consideration. The following is provided in support of this conclusion. 1. Does the change involve a significant increase in the probability or consequences of an accident previously evaluated? The proposed change relocates requirements from the Technical Specifications to the Bases, FSAR, or other plant controlled documents. The Bases and FSAR will be maintained using the provisions of 10 CFR 50.59. In addition to 10 CFR 50.59 provisions, the Technical Specifications Bases are subject to the change process in the Administrative Controls Chapter of the ITS. Plant procedures and other plant controlled documents are subject to controls imposed by plant administrative procedures, which endorse applicable regulations and standards. Changes to the Bases, FSAR, or other plant controlled documents will be evaluated in accordance with the requirements of the Bases Control Program in Chapter 5.0 of the ITS, 10 CFR 50.59, or plant administrative processes. Therefore, no increase in the probability or consequences of an accident previously evaluated will be allowed. 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 require a physical alteration of the plant (no new or different type of equipment will be installed) or changes in parameters governing normal plant operation. The proposed change will not impose any different requirements and adequate control of the information will be maintained. Thus, this change does not create the possibility of a new or different kind of accident from any accident previously evaluated. 3. Does this change involve a significant reduction in a margin of safety? The proposed change will not reduce a margin of safety because it has no impact on any safety analysis assumptions. In addition, the requirements to be moved from the Technical Specifications to the Bases, FSAR, or other plant controlled documents are as they currently exist. Future changes to the requirements in the Bases, FSAR, or other plant controlled documents will be evaluated in accordance with the requirements of 10 CFR 50.59, the Bases Control Program in Chapter 5.0 of the ITS, or the applicable plant process and no reduction in a margin of safety will be allowed.

NSHC Number	NSHC Text
M Rev. B	In accordance with the criteria set forth in 10 CFR 50.92, PBNP has evaluated this proposed Technical Specifications change and determined it does not represent a significant hazards consideration. The following is provided in support of this conclusion. 1. Does the change involve a significant increase in the probability or consequences of an accident previously evaluated? The proposed change provides more restrictive requirements for operation of the facility. These more stringent requirements do not result in operation that will increase the probability of initiating an analyzed event and do not alter the assumptions relative to the mitigation of an accident or transient event. These more restrictive requirements continue to ensure process variables, structures, systems and components are maintained consistent with the safety analyses. Therefore, this change does not increase 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 require a physical alteration of the plant (no new or different type of equipment will be installed) or changes in parameters governing normal plant operation. The proposed change does impose different requirements. However, these changes are consistent with assumptions made in the safety analysis. Thus, this change does not create the possibility of a new or different kind of accident from any accident previously evaluated. 3. Does this change involve a significant reduction in a margin of safety? The imposition of more restrictive requirements either has no affect on or increases the margin of safety. Each change is providing additional restrictions to enhance plant safety. These changes are consistent with the safety analysis. Therefore, this change does not involve a reduction in a margin of safety.

3.1 REACTIVITY CONTROL SYSTEMS

3.1.5 Shutdown Bank Insertion Limits

LCO 3.1.5 Each shutdown bank shall be within insertion limits specified in the COLR.

APPLICABILITY: MODES 1 and 2

B
RAI 3.1.6-1

----------NOTE----------
This LCO is not applicable while performing SR 3.1.4.2.

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. One or more shutdown banks not within limits.	A.1.1	Verify SDM to be within the limits provided in the COLR.	1 hour
	OR		
	A.1.2	Initiate boration to restore SDM to within limit.	1 hour
	AND		
	A.2	Restore shutdown banks to within limits.	2 hours
B. Required Action and associated Completion Time not met.	B.1	Be in MODE 3.	6 hours

B
RAI 3.1.6-1

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.1.5.1	Verify each shutdown bank is within the limits specified in the COLR.	12 hours

B 3.1 REACTIVITY CONTROL SYSTEMS

B 3.1.5 Shutdown Bank Insertion Limits

BASES

BACKGROUND

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

The design criteria for reactivity and power distribution are found in FSAR Section 3.1, (Ref. 1), and 10 CFR 50.46, "Acceptance Criteria for Emergency Core Cooling Systems for Light Water Nuclear Power Reactors" (Ref. 2). Limits on control rod insertion have been established, and all rod positions are monitored and controlled during power operation to ensure that the power distribution and reactivity limits defined by the design power peaking and SDM limits are preserved.

The rod cluster control assemblies (RCCAs) are divided among control banks and shutdown banks. Each bank may be further subdivided into one or two groups to provide for precise reactivity control. A group consists of two or more RCCAs that are electrically paralleled to step simultaneously. A bank of RCCAs may consist of one or two groups. When a bank consists of two groups, the groups are moved in a staggered fashion, but always within one step of each other. Control banks A and C and shutdown bank A consist of two groups each while control banks B and D and shutdown bank B consist of a single group. See LCO 3.1.4, "Rod Group Alignment Limits," for control and shutdown rod OPERABILITY and alignment requirements, and LCO 3.1.7, "Rod Position Indication," for position indication requirements.

The control banks are used for precise reactivity control of the reactor. The positions of the control banks are normally automatically controlled by the Rod Control System, but they can also be manually controlled. They are capable of adding negative reactivity very quickly (compared to borating). The control banks must be maintained above designed insertion limits and are typically near the fully withdrawn position during normal full power operations.

Hence, they are not capable of adding a large amount of positive reactivity. Boration or dilution of the Reactor Coolant System (RCS) compensates for the reactivity changes associated with large changes in RCS temperature. The design calculations are performed with the assumption that the shutdown banks are withdrawn first. The shutdown

BASES

BACKGROUND (continued)

banks can be fully withdrawn without the core going critical. This provides available negative reactivity in the event of boration errors. The shutdown banks are controlled manually by the control room operator. During normal unit operation, the shutdown banks are either fully withdrawn or fully inserted. The shutdown banks must be completely withdrawn from the core, prior to withdrawing any control banks during an approach to criticality. The shutdown banks are then left in this position until the reactor is shut down. They affect core power and burnup distribution, and add negative reactivity to shut down the reactor upon receipt of a reactor trip signal.

B RAI 3.1.6-1

APPLICABLE SAFETY ANALYSES

On a reactor trip, all RCCAs (shutdown banks and control banks), except the most reactive RCCA, are assumed to insert into the core. The shutdown banks shall be at or above their insertion limits and available to insert the maximum amount of negative reactivity on a reactor trip signal. The control banks may be partially inserted in the core, as allowed by LCO 3.1.6, "Control Bank Insertion Limits." The shutdown bank and control bank insertion limits are established to ensure that a sufficient amount of negative reactivity is available to shut down the reactor and maintain the required SDM (see LCO 3.1.1, "SHUTDOWN MARGIN (SDM)") following a reactor trip from full power. The combination of control banks and shutdown banks (less the most reactive RCCA, which is assumed to be fully withdrawn) is sufficient to take the reactor from full power conditions at rated temperature to zero power, and to maintain the required SDM at rated no load temperature (Ref. 3). The shutdown bank insertion limit also limits the reactivity worth of an ejected shutdown rod.

The acceptance criteria for addressing shutdown and control rod bank insertion limits and inoperability or misalignment is that:

a. There be no violations of:

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

b. The core remains subcritical after accident transients.

As such, the shutdown bank insertion limits affect safety analysis involving core reactivity and SDM (Ref. 4).

The shutdown bank insertion limits preserve an initial condition assumed in the safety analyses and, as such, satisfy Criterion 2 of the NRC Policy Statement.

BASES

LCO

The shutdown banks must be within their insertion limits any time the reactor is critical or approaching criticality. This ensures that a sufficient amount of negative reactivity is available to shut down the reactor and maintain the required SDM following a reactor trip.

B RAI 3.1.6-1

The shutdown bank insertion limits are defined in the COLR.

APPLICABILITY

The shutdown banks must be within their insertion limits, with the reactor in MODES 1 and 2. This ensures that a sufficient amount of negative reactivity is available to shut down the reactor and maintain the required SDM following a reactor trip. The shutdown banks do not have to be within their insertion limits in MODE 3, unless an approach to criticality is being made. In MODE 3, 4, 5, or 6, the shutdown banks are fully inserted in the core and contribute to the SDM. Refer to LCO 3.1.1 for SDM requirements in MODES 3, 4, and 5. LCO 3.9.1, "Boron Concentration," ensures adequate SDM in MODE 6.

The Applicability requirements have been modified by a Note indicating the LCO requirement is suspended during SR 3.1.4.2. This SR verifies the freedom of the rods to move, and requires the shutdown bank to move below the LCO limits, which would normally violate the LCO.

B RAI 3.1.6-1

ACTIONS

A.1.1, A.1.2 and A.2

When one or more shutdown banks is not within insertion limits, 2 hours is allowed to restore the shutdown banks to within the insertion limits. This is necessary because the available SDM may be significantly reduced, with one or more of the shutdown banks not within their insertion limits. Also, verification of SDM or initiation of boration within 1 hour is required, since the SDM in MODES 1 and 2 is ensured by adhering to the control and shutdown bank insertion limits (see LCO 3.1.1). If shutdown banks are not within their insertion limits, then SDM will be verified by performing a reactivity balance calculation, considering the following listed reactivity effects:

B RAI 3.1.6-1

a. RCS boron concentration;

b. Control bank position;

c. Power defect;

d. Fuel burnup;

e. Xenon concentration; and

f. Samarium concentration.

BASES

ACTIONS (continued)

The allowed Completion Time of 2 hours provides an acceptable time for evaluating and repairing minor problems without allowing the plant to remain in an unacceptable condition for an extended period of time.

B.1

If the shutdown banks cannot be restored to within their insertion limits within 2 hours, the unit must be brought to a MODE where the LCO is not applicable. The allowed Completion Time of 6 hours is reasonable, based on operating experience, for reaching the required MODE from full power conditions in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.1.5.1

Verification that the shutdown banks are within their insertion limits prior to an approach to criticality ensures that when the reactor is critical, or being taken critical, the shutdown banks will be available to shut down the reactor, and the required SDM will be maintained following a reactor trip. This SR and Frequency ensure that the shutdown banks are withdrawn before the control banks are withdrawn during a unit startup. Typically, the individual rod position indicators are used to confirm shutdown bank insertion limits.

Since the shutdown banks are positioned manually by the control room operator, a verification of shutdown bank position at a Frequency of 12 hours, after the reactor is taken critical, is adequate to ensure that they are within their insertion limits. Also, the 12 hour Frequency takes into account other information available in the control room for the purpose of monitoring the status of shutdown rods. B RAI 3.1.6-1

REFERENCES

1. FSAR, Section 3.1.
2. 10 CFR 50.46.
3. FSAR, Section 3.2.
4. FSAR, Chapter 14.

Justification For Deviations - NUREG-1431 Section 3.01.07

14-Jun-00

JFD Number	JFD Text	
01 Rev. A	Reference to the General Design Criteria (GDC) of 10 CFR 50 Appendix A has been deleted from the Bases of the Technical Specifications, substituting reference to the appropriate section of the FSAR which specifies the Point Beach design criteria. Point Beach was constructed and licensed prior to the GDC being issued. The Point Beach construction permit was issued prior to the GDCs being issued in 1971. Point Beach was designed and constructed utilizing the 1967 proposed GDCs. Accordingly, reference has been provided to the appropriate criteria and section of the Point Beach FSAR which provides explanation of Point Beach's design basis.	
	ITS:	**NUREG:**
	B 3.01.06	B 3.01.07
02 Rev. A	The Bases for LCO 3.1.7 provides a description of control bank overlap, which includes specific reference to the position of Control Bank C when Control Bank D begins to move in addition to the fully withdrawn position for the control rods. The C Bank position at which Control Bank D begins to move at Point Beach is 125 steps and the fully withdrawn position for the control rods is 225 steps. These site specific numbers have been substituted for the numbers used in NUREG 1431.	
	ITS:	**NUREG:**
	B 3.01.06	B 3.01.07
03 Rev. A	The Bases for LCO 3.1.7 provides three FSAR Section references (3, 4, and 5) for various analyses and parameters. Reference 3 contains contains a broad reference to the Accident Analyses Section of the FSAR which contains the accident analyses assumptions for analyzed events. Therefore, Reference 4 and 5 are unnecessary.	
	ITS:	**NUREG:**
	B 3.01.06	B 3.01.07
04 Rev. A	The brackets have been removed and the proper plant specific information has been provided.	
	ITS:	**NUREG:**
	B 3.01.06	B 3.01.07

Justification For Deviations - NUREG-1431 Section 3.01.07

14-Jun-00

JFD Number	JFD Text
05 Rev. A	The Bases for Required Action A.1.1 references the Bases for SR 3.1.1.1 to obtain a listing of effects for calculation of SDM when one or more shutdown banks are not within limits. This LCO Action is applicable in Modes 1 and 2 with Keff greater than or equal to 1.0. SR 3.1.1.1 calculates SDM in Mode 2 with Keff < 1.0, and Modes 3, 4, and 5, addressing subcritical conditions. Therefore, the Bases for SR 3.1.1.1 contains effects which are not applicable to an operating reactor. Proposed ITS LCO 3.1.5 Required Action A.1.1 has been modified to list the effects listed from the Bases for SR 3.1.1.1 which are applicable to reactor critical operation. **ITS:** B 3.01.06 **NUREG:** B 3.01.07
06 Rev. A	Figure B 3.1.7-1, "Control Bank Insertion Limits" is provided in the Bases of the ITS as an example of the rod insertion limits in addition to a reference in explaining the concept of bank overlap. This figure has been retained, but as a generic figure for information only to eliminate any possible confusion as to the figure's use. Plant specific information is contained in the COLR. **ITS:** B 3.01.06 **NUREG:** B 3.01.07
07 Rev. A	The Mode of Applicability for NUREG LCO 3.1.7 is Mode 1 and 2 with Keff greater than or equal to 1.0, while the default Required Action (C.1) requires the unit to be placed into Mode 3 within 6 hours. LCO 3.0.2 states that an LCOs Required Actions are no longer applicable if an LCO is met or no longer applicable. Accordingly, the Required Actions are no longer applicable after the unit reaches Mode 2 with Keff less than 1.0. Therefore, the default action has been revised to require the unit to be placed into Mode 2 with Keff less than 1.0 within 6 hours to establish continuity between the General Usage LCOs and the Required Actions for NUREG 1431 LCO 3.1.7. This implements approved TSTF 238, Revision 0. **ITS:** B 3.01.06 **NUREG:** B 3.01.07 LCO 3.01.06 COND C RA C.1 LCO 3.01.07 COND C RA C.1

Justification For Deviations - NUREG-1431 Section 3.01.07

14-Jun-00

JFD Number	JFD Text
08 Rev. B	NUREG 1431 allows control bank insertion, sequence, and overlap limits to be specified in the COLR. NUREG 1431 SR 3.1.7.3 states that control banks which are fully withdrawn from the core are not required to be checked for proper sequence and overlap (as in the fully withdrawn position, overlap and sequence are no longer observable parameters). The CTS defines fully withdrawn for the control rods as being greater than or equal to 225 steps. This is to allow the control rods to be "parked" at this position or higher while meeting the definition of fully withdrawn. Defining fully withdrawn at greater than or equal to 225 steps minimizes control rod cladding wear caused by vibration in the guide card area. Fully withdrawn is not adequately defined in the NUREG. Fully withdrawn is subjective and should be defined and maintained consistent with the control bank insertion limits. Therefore, the definition of "fully withdrawn" is proposed for inclusion into the COLR. The Bases has been slightly modified to reflect this COLR attribute. **ITS:** B 3.01.06 — **NUREG:** B 3.01.07
09 Rev. A	The porposed Bases has been modified to reflect the Point Beach design. Not all control rod banks consist of two groups of rods as stated in the Bases of NUREG 1431 LCO 3.1.6. Control banks may consist of a single group dependent upon the number of control rods assigned to the bank. Typically control rod banks with four or fewer rods consist of a single group. Any bank that consists of two groups will step the banks within one step of each other. **ITS:** B 3.01.06 — **NUREG:** B 3.01.07
10 Rev. A	The Mode of Applicability for NUREG 1431 LCO 3.1.7 is Mode 1 and Mode 2 when Keff is greater than or equal to 1.0, however, the Applicability and Actions Sections of the NUREG Bases do not reflect this completely. As such, the proposed ITS Bases has been changed to correspond to the actual Applicability of the LCO. **ITS:** B 3.01.06 — **NUREG:** B 3.01.07

Approved TSTF 136

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.1.~~7~~ 6 2 Approved TSTF 136	Verify each control bank insertion is within the limits specified in the COLR.	12 hours ~~AND~~ ~~Once within 4 hours and every 4 hours thereafter when the rod insertion limit monitor is inoperable~~ Approved TSTF 110
SR 3.1.~~7~~ 6 3 Approved TSTF 136	Verify sequence and overlap limits specified in the COLR are met for control banks not fully withdrawn from the core.	12 hours B RAI 3.1.7-1

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.1.6.3	Verify sequence and overlap limits specified in the COLR are met for control banks not fully withdrawn from the core.	12 hours B RAI 3.1.7-1

Description of Changes - NUREG-1431 Section 3.01.08

24-May-00

DOC Number	DOC Text

A.01
Rev. A

In the conversion of Point Beach current Technical Specifications (CTS) to the proposed plant specific Improved Technical Specifications (ITS), certain wording preferences or conventions are adopted which do not result in technical changes (either actual or interpretational). Editorial changes, reformatting, and revised numbering are adopted to make the ITS consistent with the Standard Technical Specifications, Westinghouse Plants, NUREG-1431, Revision 1 (i.e., Improved Standard Technical Specifications (ISTS)).

CTS:	ITS:
15.03.10.C	LCO 3.01.07
15.03.10.C NOTE	LCO 3.01.07 NOTE
15.03.10.C.01.A	LCO 3.01.07 COND A
15.03.10.C.01.A.01	LCO 3.01.07 COND A RA A.1.1
15.03.10.C.01.A.02	LCO 3.01.07 COND A RA A.1.2
15.03.10.C.01.A.03	LCO 3.01.07 COND D
15.03.10.C.01.B.01	LCO 3.01.07 COND B RA B.1
15.03.10.C.01.C	LCO 3.01.07 COND C
15.03.10.C.01.C.01	LCO 3.01.07 COND C RA C.1.1
15.03.10.C.01.C.02	LCO 3.01.07 COND C RA C.1.2

A.02
Rev. A

The CTS provides an introductory statement (Applicability) which simply states which systems/components are addressed within a given section. This same information, while worded differently, is contained within the title of each ITS LCO. Accordingly, this change is a change in format with no change in technical requirement.

CTS:	ITS:
15.03.10 APPL	LCO 3.01.07

A.03
Rev. A

The CTS provides an introductory statement (Objective) at the beginning of this Section of the Technical Specifications which provide a brief summary of the purpose for this Section. This information is contained in the Bases Section of the ITS. This information does not establish any regulatory requirements for the systems and components addressed within this Section. Accordingly, deletion of this information does not alter any requirement set forth in the Technical Specifications. This change is administrative and consistent with the format and presentation for the ITS as provided in NUREG 1431.

CTS:	ITS:
15.03.10 OBJ	B 3.01.07

DOC Number	DOC Text
A.04 Rev. A	The Bases of the current Technical Specifications for this section have been completely replaced by revised Bases that reflect the format and applicable content of PBNP ITS, consistent with the Standard Technical Specifications for Westinghouse Plants, NUREG-1431. The revised Bases are as shown in the PBNP ITS Bases. **CTS:** BASES **ITS:** B 3.01.07
A.05 Rev. A	CTS 15.3.10.C.1 requires the rod position indication system and the bank demand position indication system to be operable. This requirement has been retained as ITS LCO 3.1.7. The Bases has been modified to reflect the individual control rod position indications available for verification of control rod position and alignment at Point Beach. The rod position indication system at Point Beach is an analog system which provides individual control rod position indication to three separate control room readouts; analog meters, digital displays, and the plant process computer. Any one of these three indicators can be used for the purpose of verifying control rod position and alignment. The position indication signal to each of these readouts is supplied by a linear variable differential transformer (LVDT) which uses the control rod drive shaft to vary the amount of magnetic coupling between primary and secondary windings of the transformer. This generates an analog output signal proportional to actual control rod position. The analog display meters and the digital displays, provide position readouts in direct proportion to the output signal form the LVDT signal conditioning circuit. The process computer uses a fourth order polynomial to provide a more accurate control rod position readout, compensating for non-linearities in the LVDT system. **CTS:** 15.03.10.C.01 **ITS:** LCO 3.01.07

Description of Changes - NUREG-1431 Section 3.01.08

24-May-00

DOC Number	DOC Text
A.06 Rev. B	The CTS requires control rods with inoperable individual position indicators or demand indicators to be checked for positions or alignment, as appropriate on a periodic basis. In lieu of performing these checks, the CTS allows the affected control rods to be declared misaligned and the Technical Specification Action for misalignment to be entered. Accordingly, if multiple individual position indicators were inoperable or a demand indicator was inoperable and the position verification or alignment checks were not performed, the CTS would ultimately require the unit to be placed into Hot Shutdown within six hours. The ITS will allow multiple control rod position indicators or demand indicators to be inoperable simultaneously as the CTS does and will similarly require the unit to be shutdown within 6 hours if the required actions of ITS LCO 3.1.7 cannot be met. This change is administrative.

CTS:	ITS:
15.03.10.B.01.B.01.F	LCO 3.01.07 COND D LCO 3.01.07 COND D RA D.1
15.03.10.B.01.B.02	DELETED
15.03.10.B.01.B.02.A	DELETED
15.03.10.B.01.B.02.B	LCO 3.01.07 COND D RA D.1
15.03.10.C.01.A.03	LCO 3.01.07 COND D RA D.1
15.03.10.C.01.B.02	LCO 3.01.07 COND D RA D.1
15.03.10.C.01.C.03	LCO 3.01.07 COND D RA D.1

L.01 Rev. A	The frequency for calibration of the individual rod position indicators has been changed from each refueling interval (18 months) to prior to reactor criticality after each removal of the reactor head. Verification of agreement between the individual rod position indicators and the group demand position is required by the ITS once every 12 hours in Modes 1 and 2. This test provides a verification of rod position indicator agreement for the current control rods position. Significant deviation between demanded and indicated position will result in evaluation of control rod position or position indicator operability, with corrective actions to reestablish proper indication or control rod position as necessary. Based on the rod position indicators being verified for agreement on a 12 hour frequency, it is highly unlikely that significant deviations or failures could go undetected. Testing is necessary following removal of the reactor head due to the rod position indicators being disconnected as part of the reactor vessel head package removal. As such, a frequency of prior to reactor criticality after each removal of the reactor vessel head is an acceptable frequency for calibration of the indivdual rod position indicators.

CTS:	ITS:
15.04.01 T 15.04.01-01 19	SR 3.01.07.01
15.04.01 T 15.04.01-01 R - REFUELING	SR 3.01.07.01

DOC Number	DOC Text
L.02 Rev. B	Not Used.

CTS:	ITS:
N/A	N/A

DOC Number	DOC Text
LA.01 Rev. A	The CTS requires the position of control rods with inoperable individual rod position indicators to be verified each shift using the excore detectors, thermocouples, or movable incore detectors. The proposed ITS will continue to require verification of control rod position on a periodic basis, but will move the identification of the specific methods used to licensee control. Specifying the specific methods is a detail which is not directly pertinent to the requirement, but rather describes acceptable methods of compliance. Since these details are not necessary to adequately describe the actual regulatory requirement, they can be moved to licensee controlled documents without an impact on safety. Placing these details in licensee controlled documents provides adequate assurance that they will be maintained. The acceptable methods outlined in the CTS will be moved to the Bases of the proposed ITS, which will be controlled by the Bases Control Process contained in Chapter 5.

CTS:	ITS:
15.03.10.C.01.A.02	B 3.01.07
15.03.10.C.01.B.01	B 3.01.07

DOC Number	DOC Text
M.01 Rev. A	The CTS allows control rods with inoperable individual position indicators or demand indicators to be declared misaligned in lieu of verifying control rod positions or alignment. Under the CTS this could allow continued operation at power levels up to 75% power for a single inoperable rod position indicator. The ITS will similarly allow continued operation with a single rod position indicator inoperable, but will limit the maximum allowable power level without verification of rod positions to 50% power. Reducing the allowable power level to less than or equal to 50% power will place the core into a condition where control rod position does not significantly affect peaking factors and is more restrictive than the CTS.

CTS:	ITS:
15.03.10.B.01.B.01.A	DELETED
15.03.10.B.01.B.01.B	DELETED
15.03.10.B.01.B.01.C	DELETED
15.03.10.B.01.B.01.D	DELETED
15.03.10.B.01.B.01.E	DELETED
15.03.10.B.01.B.01.G	DELETED
15.03.10.C.01.A.03	LCO 3.01.07 COND D RA D.1
15.03.10.C.01.B.02	LCO 3.01.07 COND D RA D.1
15.03.10.C.01.C.03	LCO 3.01.07 COND D RA D.1

Description of Changes - NUREG-1431 Section 3.01.08

24-May-00

DOC Number	DOC Text
M.02 Rev. B	The Mode of Applicability for the rod position indicators has been revised from greater than or equal to 10% power to ITS Modes 1 and 2. The purpose of this LCO is to establish the minimum performance level and preferred methods of determining control rod position. Control rod demand and individual indicated positions are necessary for compliance with the control rod alignment requirements of ITS LCO 3.1.4, 3.1.5, and 3.1.6. As such, the applicability for the control rod position indicators has been revised to be consistent with the limitation it supports.

CTS:	ITS:
15.03.10.C.01	LCO 3.01.07

M.03 Rev. A	Line item 19 of table 15.4.1-1, requires channel calibrations to be performed for the rod position indicators in "all" plant conditions. Table 15.4.1-1 defines "all" plant conditions through reference to Specification 15.1.g, h, and m, which are; 1] Shutdown (Hot, Cold, Refueling, and Shutdown Margin), 2] Power Operations (greater than 2% power), and 3] Low Power Operation (less than or equal to 2% power). As such, defining the applicability of this surveillance in the terms specified in Specification 15.1.g, h, and m are vague and non prescriptive. Specification 15.4.0.1 states that surveillance requirements shall be met during all times that the system or component is required to be operable. Through applying Specification 15.4.0.1, the CTS applicability of line item 19 is the same as CTS 15.3.10.C - Rod Position Indication, during power operation greater than or equal to 10% power. As addressed in Description of Change M.2 of this Section, the ITS Mode of Applicability for the rod position indicators has been changed to be Modes 1 and 2, which would encompass any reactor power level with Keff greater than or equal to 0.99. This change establishes consistency between the Mode of Applicability for the LCO and the Surveillance Requirement.

CTS:	ITS:
15.04.01 T 15.04.01-01 19	SR 3.01.07.01
15.04.01 T 15.04.01-01 ALL	SR 3.01.07.01

A.1

M.1

(c) Verify that the shutdown margin exceeds the applicable value as shown in Figure 15.3.10-2 once per twelve hours;

AND

(d) Within 72 hours verify that measured values of $F_Q(Z)$ are within limits;

AND

(e) Within 72 hours verify that $F^N_{\Delta H}$ is within limits;

(f) If the above actions and associated completion times are not met, be in hot shutdown within the following six hours.

(g) In order to subsequently increase thermal power above 75 percent of rated thermal power with the existing rod misalignment, perform an analysis to determine the hot channel factors and the resulting allowable power level in accordance with TS 15.3.10.E.

A.6

(2) If it has been determined that more than one rod is not within alignment limits and the misalignments are not being caused by malfunctioning rod position indication, perform the following actions:

(a) Within one hour verify that the shutdown margin exceeds the applicable value as shown in Figure 15.3.10-2; OR within one hour restore the shutdown margin by boration;

AND

(b) Be in hot shutdown within six hours.

B RAI 3.1.8-1

A.5

LCO 3.1.7

Required Actions Note

C. ROD POSITION INDICATION

NOTE: Separate entry into TS 15.3.10.C.1.a, b, or c is allowed for each inoperable rod position indicator and each bank of demand position indication.

M.2

Modes 1 and 2

1. During power operation ≥10 percent of rated thermal power, the rod position indication system and the bank demand position indication system shall be operable.

B RAI 3.1.8-2

Cond A

a. If one or more rod position indicators (RPI) are determined to be inoperable, perform the following actions:

Cond A RA A.1.1

(1) Within eight hours verify the position of the rods with inoperable RPIs by using movable incore detectors;

AND

D. Required Action and associated Completion Time not met.	D.1 Be in MODE 3.	6 hours

B RAI 3.1.8-1

A.6

Cond D RA D.1 → Be in MODE 3 within 6 hours ← M.1/A.6

Cond A RA A.1.2 → A.1

(2) Once per shift check the position of the rods with inoperable RPIs by using excore detectors, ~~or thermocouples, or movable incore~~ detectors; (LA.1)

(3) If the above actions and associated completion times are not met, ~~perform the actions in accordance with TS 15.3.10.B.1.b.~~

B RAI 3.1.8-1

Cond B →

b. If one or more rods with inoperable RPIs have been moved in excess of 24 steps in one direction since the last determination of the rod's position, perform the following actions:

Cond B RA B.1 →

(1) Within four hours check the position of the rods with inoperable RPIs by using excore detectors, ~~or thermocouples, or movable incore~~ detectors; (LA.1)

A.1 → Cond C → One or more demand position indicator(s) per bank inoperable for one or more banks.

(2) If the above action and associated completion time is not met, ~~perform the actions in accordance with TS 15.3.10.B.1.b.~~ → Be in MODE 3 within 6 hours ← M.1/A.6

B RAI 3.1.8-1

Cond D RA D.1

c. If bank demand position indication, for one or more banks, is determined to be inoperable, perform the following actions:

Cond C RA C.1.1 →

Cond C RA C.1.2 →

(1) Once per shift verify that all RPIs for the affected banks are operable; AND

(2) Once per shift verify that the most withdrawn rod and the least withdrawn rod of the affected banks are ≤12 steps apart, except when the bank demand position is ≤30 steps or ≥215 steps. In this case, once per shift verify that the most withdrawn rod and the least withdrawn rod of the affected banks are ≤24 steps apart;

B RAI 3.1.8-1

Be in MODE 3 within 6 hours

(3) If the above actions and associated completion times are not met, ~~perform the actions in accordance with TS 15.3.10.B.1.b.~~

D. BANK INSERTION LIMITS

< See LCO 3.1.6 and 3.1.7 >

1. When the reactor is critical, the shutdown banks shall be fully withdrawn. Fully withdrawn is defined as a bank position equal to or greater than 225 steps. This definition is applicable to shutdown and control banks.

If this condition is not met, perform the following actions:

a. Within one hour verify that the shutdown margin exceeds the applicable value as shown in Figure 15.3.10-2; OR within one hour restore the shutdown margin by boration;

M.1/A.6 → Cond D RA D.1

B RAI 3.1.8-1

Justification For Deviations - NUREG-1431 Section 3.01.08

24-May-00

JFD Number	JFD Text
01 Rev. A	NUREG 1431 refers to the equipment used to provide individual rod position indication as being digital, while the equipment installed and used at Point Beach for position indication of individual control rods is analog. Additionally, NUREG 1431 refers to the demand position indicators as "group step counters", while the verbiage used for these indicators at Point Beach is "bank demand step counters". Accordingly, the description of the equipment and terminology used in the proposed ITS has been alter to reflect Point Beach's design.

ITS:	NUREG:
B 3.01.07	B 3.01.08
LCO 3.01.07	LCO 3.01.08
LCO 3.01.07 COND A	LCO 3.01.08 COND A
LCO 3.01.07 COND C RA C.1.1	LCO 3.01.08 COND C RA C.1.1

02 Rev. A	Brackets have been removed and the appropriate plant specific information has been input.

ITS:	NUREG:
B 3.01.07	B 3.01.08
LCO 3.01.07 COND B RA B.1	LCO 3.01.08 COND B RA B.1

JFD Number	JFD Text
03 Rev. A	The acceptance criteria for NUREG 1431 SR 3.1.8.1 requires the rod position indicators to agree within plus or minus 12 steps of their respective bank demand counters over the range of indicated travel. The proposed ITS will require a channel calibration to be performed for each rod position indicator. The RPIs for Point Beach indicate over the entire range of control rod travel, making the NUREG latitude of "indicated travel" unnecessary. The rod position indication system at Point Beach is an analog system which use position signals generated by linear variable differential transformers (LVDTs). The LVDTs generate an output signal proportional to actual control rod position, however, the LVDT output signal is not linear, which in effect is reflected by the variable control rod alignment limits contained in CTS 15.3.10.B.1. CTS 15.3.10.B.1 requires that rod alignment be maintained within plus or minus 24 steps of demanded position for rod positions less than or equal to 30 steps and greater than or equal to 215 steps. Demanded position in between these limits must be within plus or minus 12 steps. Based on the non-linearities inherent to the system, the CTS requirement to perform a channel calibration has been retained. By applying the definition of channel calibration to the individual rod position indicators, adjustment will be required such that the channel will respond within its "required range and accuracy" to a known input. This will allow calibration of the rod position indicators within the limitation of the equipment as currently allowed by the CTS. **ITS:** B 3.01.07 — **NUREG:** B 3.01.08 SR 3.01.07.01 — SR 3.01.08.01

Justification For Deviations - NUREG-1431 Section 3.01.08

24-May-00

JFD Number	JFD Text
04 Rev. A	The CTS requires all shutdown and control rods to be within an alignment limit which is based on demanded control rod position. Between 30 and 215 step of demand, the limit is 12 steps with the limit becoming 24 steps at less than or equal to 30 and greater than or equal to 215 steps of demand. NUREG 1431 requires periodic verification of control rod positions when the rod position indicators are inoperable. The acceptance limits for these periodic verifications is based on a fixed alignment acceptance criteria of 12 steps. The proposed ITS Actions will require control rod alignment to be verified within the variable alignment limit of 12 steps between 30 and 215 steps and 24 steps when less than or equal to 30 steps or greater than or equal to 215 steps. This change is necessary to retain the variable alignment limit contained in the CTS. The variable rod alignment limits is required based on non-linearities that exist in the analog rod position indication system. The expanded limits for rod alignment at the extreme ends of control rod demanded position as allowed by the CTS are acceptable based on the relatively low rod worth and peaking factors in this range. **ITS:** B 3.01.07 — **NUREG:** B 3.01.08 LCO 3.01.07 COND C RA C.1.2 — LCO 3.01.08 COND C RA C.1.2
05 Rev. A	Reference to the General Design Criteria (GDC) of 10 CFR 50 Appendix A has been deleted from the Bases of the Technical Specifications, substituting reference to the appropriate section of the FSAR which specifies the Point Beach design criteria. Point Beach was constructed and licensed prior to the GDC being issued. The Point Beach construction permit was issued prior to the GDCs being issued in 1971. Point Beach was designed and constructed utilizing the 1967 proposed GDCs. Accordingly, reference has been provided to the appropriate criteria and section of the Point Beach FSAR which provides explanation of Point Beach's design basis. **ITS:** B 3.01.07 — **NUREG:** B 3.01.08
06 Rev. A	The Bases for NUREG LCO 3.1.8 Required Action A.2 states that a reduction in thermal power to less than or equal to 50% power places the core into a condition where rod position does not significantly affect core peaking factors. This Bases statement then provides reference to FSAR Section [15] as the basis for this statement. Point Beach's FSAR does not state nor does it reference any analysis as a basis for this conclusion. This Technical Specifications Action is based on the qualitative evaluations concluding that a reduction in core power level establishes an acceptable trade off between core thermal output and control rod position and alignment uncertainties. As such, reference to an FSAR Section has been deleted. **ITS:** B 3.01.07 — **NUREG:** B 3.01.08

Justification For Deviations - NUREG-1431 Section 3.01.08

24-May-00

JFD Number	JFD Text
07 Rev. B	Not Used. **ITS:** B 3.01.07 **NUREG:** B 3.01.08
08 Rev. A	The Bases has been modified to reflect the individual control rod position indications available for verification of control rod position and alignment at Point Beach. The rod position indication system at Point Beach is an analog system which provides individual control rod position indication to three separate control room readouts; analog meters, digital plasma displays, and the plant process computer. Any one of these three indicators can be used for the purpose of verifying control rod position and alignment. The position indication signal to each of these readouts is supplied by a linear variable differential transformers (LVDT) which uses the control rod drive shaft to vary the amount of magnetic coupling between primary and secondary windings of the transformer. This generates an analog output signal proportional to actual control rod position. The analog display meters and the plasma displays, provide position readouts in direct proportion to the output signal form the LVDT signal conditioning circuit. The process computer uses a fourth order polynomial to provide a more accurate control rod position readout, compensating for non-linearities in the LVDT system. **ITS:** B 3.01.07 **NUREG:** B 3.01.08

JFD Number	JFD Text
09 Rev. A	Whenever an individual rod position indicator is inoperable, the Actions contained in NUREG 1431 require the position of each rod with an inoperable position indicator to be verified using the movable incore detectors once per 8 hours. Additionally, the position of each rod with an inoperable position indicator must be verified within 4 hours of any rod with an inoperable indicator being moved in excess of 24 steps in one direction. Whenever an individual rod position indicator is inoperable, the CTS requires the position of each rod with an inoperable position indicator to be verified using the movable incore detectors within 8 hours. Subsequent position verifications are required once per shift and within 4 hours of any rod with an inoperable indicator being moved in excess of 24 steps in one direction using the excore detectors, thermocouples, or movable incore detectors. The proposed ITS will require the position of each rod with an inoperable position indicator to be verified using the movable incore detectors within 8 hours. Subsequent position verifications will be required every 8 hours. Subsequent verifications may be performed using other equipment (e.g. movable incore detectors, thermocouples, excore detectors, etc;) as outlined in the Bases of the ITS. In addition to these verifications, verification of position is also required within 4 hours of movement of any non-indicating rod in excess of 24 steps in one direction. This verification may also be performed using other equipment (e.g. movable incore detectors, thermocouples, excore detectors, etc;) as outlined in the Bases of the ITS. These changes are necessary to reflect the Point Beach licensing basis which allows verification of rod positions using techniques other than the movable incore detectors. Moving the equipment required to perform subsequent position verifications and verification after rod motion is addressed in Discussion of Change LA.01 of this Section.

ITS:	NUREG:
B 3.01.07	B 3.01.08
LCO 3.01.07 COND A RA A.1.1	LCO 3.01.08 COND A RA A.1
LCO 3.01.07 COND A RA A.1.2	N/A
LCO 3.01.07 COND B RA B.1	LCO 3.01.08 COND B RA B.1

10 Rev. B	Not Used.

ITS:	NUREG:
N/A	N/A

JFD Number	JFD Text
11 Rev. A	NUREG 1431 LCO 3.1.8 Condition C, allows a single control rod demand position indicator to be inoperable on one or more rod banks. The proposed ITS will allow one or more bank demand counters to be inoperable on one or more rod banks. Control and shutdown banks A and C consist of two groups of rods each group having its own demand step counter, and control banks B and D and shutdown bank B consist of a single group and therefore, a single step counter. CTS 15.3.10.C.1.b.2, allows the demand position indication to be inoperable for one or more banks. As such, the CTS allows banks which consist of two demand counters to have both counters inoperable. Having one or both step in a bank inoperable is acceptable, based on the compensatory Actions contained in ITS Condition C in combination with the, insertion, alignment, and rod sequence requirements contained in ITS LCOs 3.1.5 and 3.1.6. Condition C Required Action C.1 requires all individual rod position indicators to be verified operable, and Required Action C.2 requires that the rods be verified to be within alignment limits. Verification of actual rod position as required by these Actions, when supplement by the Surveillance contained in proposed ITS LCOs 3.1.5 and 3.1.6, which requires rod insertion limits, control rod sequence, and control rod overlap, provided assurance that control rod position will be maintained within required limits. **ITS:** B 3.01.07 — **NUREG:** B 3.01.08 LCO 3.01.07 COND C — LCO 3.01.08 COND C

JFD Number	JFD Text
12 Rev. A	The Bases of NUREG 1431 LCO 3.1.8 contains a discussion related to the operability of the individual control rod position indicator and bank demand step counters. This Section has been modified to reflect the Point Beach design as follows: The individual rod position indicators should be considered operable based on the indicators being satisfactorily calibrated. Agreement of individual rod position indication with the bank demand indicators, while required to verify alignment limits and as an input into confirming acceptable operation of the rod position indicator, is not a criteria for individual rod position indicator operability. Disagreement between these indicators can be indicative of an actual misalignment, or stuck rod. It is inappropriate to consider the individual rod position indicators inoperable, but rather the reason for disagreement investigated, and the appropriate action taken based on the cause of the disagreement. Failed coils is a criteria for digital systems, as any failed coil in an analog system will result in complete failure of the indicator. The bank demand indicators should not be calibrated solely on information from the individual rod position indicators. Doing so could invalidate the basis for the alignment criteria established in the rod alignment LCO. Calibration of the bank demand counter is accomplished through a simple agreement verification between demanded position and actual rod position. Actual position can be determined through numerous means, from a known fully inserted or withdrawn position, or flux mapping. The methods for establishing agreement between demanded position and actual rod position are details which have been addressed procedurally by the licensee, which are not required in the LCO Bases section. **ITS:** B 3.01.07 **NUREG:** B 3.01.08

JFD Number	JFD Text
13 Rev. A	NUREG 1431 LCO 3.1.8 Condition A, allows one control rod position indicator to be inoperable on one or more rod groups. The proposed ITS will allow one or more rod position indicators to be inoperable on one or more rod groups. CTS 15.3.10.C.1.a, allows one or more rod position indicators to be inoperable. As such, the CTS will allow multiple rod position indicators in the same group to be inoperable. Multiple rod position indicators in the same group inoperable is acceptable, based on the compensatory Actions contained in proposed ITS Conditions A and B which require verification of control rod positions periodically and after significant motion. Verification of actual rod position as required by these Actions, when supplemented by the Surveillance contained in proposed ITS LCOs 3.1.5 and 3.1.6, which requires rod insertion limits, control rod sequence, and control rod overlap, provided assurance that control rod position will be maintained within required limits. As such, the CTS provision which allows multiple control rod position indicators to be inoperable in the same groups has been retained. Approved TSTF 234, Revision 1 has not been incorporated into the proposed ITS. Incorporation of TSTF 234 would not be in accordance with the current licensing basis for PBNP, which provides for one or more inoperable rod position indicators. **ITS:** B 3.01.07 — **NUREG:** B 3.01.08 LCO 3.01.07 COND A — LCO 3.01.08 COND A

3.1 REACTIVITY CONTROL SYSTEMS

3.1.8 Rod Position Indication

Approved TSTF 136

LCO 3.1.8 ~~The [Digital] Rod Position Indication ([D]RPI) System and the Demand Position Indication System shall be OPERABLE.~~

Individual control rod position indication and bank demand indication shall be OPERABLE.

APPLICABILITY: MODES 1 and 2.

RAI 3.1.8-2

ACTIONS

bank

---NOTE---

Separate Condition entry is allowed for each inoperable rod position indicator per group and each demand position indicator per bank.

---

or more RPI(s)

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. One ~~[D]RPI~~ per group inoperable for one or more groups.	~~A.1~~ A.1.1 Insert 3.1.8-01	Verify the position of the rods with inoperable position indicators by using movable incore detectors.	~~Once per~~ 8 hours
	OR A.2	Reduce THERMAL POWER to ≤ 50% RTP.	8 hours
B. One or more rods with inoperable position indicators have been moved in excess of 24 steps in one direction since the last determination of the rod's position.	B.1	Verify the position of the rods with inoperable position indicators ~~by using movable incore detectors~~.	[4] hours
	OR		

RAI 3.1.8-1

(continued)

Approved TSTF 136 → 7

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
B. (continued)	B.2 Reduce THERMAL POWER to ≤ 50% RTP.	8 hours
C. One demand position indicator per bank inoperable for one or more banks. (11 or more bank) (s)	C.1.1 Verify by administrative means all ~~10~~ RPIs for the affected banks are OPERABLE. (1)	Once per 8 hours
	AND	
	C.1.2 ~~Verify the most withdrawn rod and the least withdrawn rod of the affected banks are ≤ 12 steps apart.~~	Once per 8 hours
	OR	
	C.2 Reduce THERMAL POWER to ≤ 50% RTP.	8 hours
D. Required Action and associated Completion Time not met.	D.1 Be in MODE 3.	6 hours

(4) Verify the most withdrawn rod and the least withdrawn rod of the affected banks are: ≤ 12 steps apart when demanded bank position is > 30 and < 215 steps, and ≤ 24 steps apart when demanded bank position is ≤ 30 and ≥ 215 steps.

B RAI 3.1.8-1

BASES

LCO (continued)

Approved TSTF 136

12

Replace with Insert B 3.1.8-3

~~corresponding control rod group is within the assumed values~~ used in the analysis (that specified control rod group insertion limits).

These requirements ensure that control rod position indication during power operation and PHYSICS TESTS is accurate, and that design assumptions are not challenged.

OPERABILITY of the position indicator channels ensures that inoperable, misaligned, or mispositioned control rods can be detected. Therefore, power peaking, ejected rod worth, and SDM can be controlled within acceptable limits.

APPLICABILITY

1

Approved TSTF 136

The requirements on the DRPI and step counters are only applicable in MODES 1 and 2 (consistent with LCO 3.1.5, LCO 3.1.6, and LCO 3.1.7), because these are the only MODES in which power is generated, and the OPERABILITY and alignment of rods have the potential to affect the safety of the plant. In the shutdown MODES, the OPERABILITY of the shutdown and control banks has the potential to affect the required SDM, but this effect can be compensated for by an increase in the boron concentration of the Reactor Coolant System.

4

5

6

Approved TSTF 136

B

RAI 3.1.8-2

ACTIONS

The ACTIONS table is modified by a Note indicating that a separate Condition entry is allowed for each inoperable rod position indicator per group and each demand position indicator per bank. This is acceptable because the Required Actions for each Condition provide appropriate compensatory actions for each inoperable position indicator.

A.1.1

A.1

9/13

Insert B 3.1.8-5

When one DRPI channel per group fails, the position of the rod can still be determined by use of the incore movable detectors. Based on experience, normal power operation does not require excessive movement of banks. If a bank has been significantly moved, the Required Action of B.1 or B.2 below is required. Therefore, verification of RCCA position within the Completion Time of 8 hours is adequate for

B

RAI 3.1.8-2

Approved TSTF 136

7

BASES

ACTIONS (continued)

Insert B 3.1.8-5

Approved TSTF 136

~~allowing continued full power operation, since the probability of simultaneously having a rod significantly out of position and an event sensitive to that rod position is small.~~

A.1.2

Insert B 3.1.8-6

A.2

Reduction of THERMAL POWER to ≤ 50% RTP puts the core into a condition where rod position is not significantly affecting core peaking factors ~~(Ref. 3)~~

The allowed Completion Time of 8 hours is reasonable, based on operating experience, for reducing power to ≤ 50% RTP from full power conditions without challenging plant systems and allowing for rod position determination by Required ~~Action A.1~~ above.

Actions A.1.1 and A.1.2

B.1 and B.2

These Required Actions clarify th at when one or more rods with inoperable position indicators have been moved in excess of 24 steps in one direction, since the position was last determined, the Required Actions of ~~A.1 and A.2~~ are still appropriate but must be initiated promptly under Required Action B.1 to begin verifying that these rods are still properly positioned, relative to their group positions.

A.1.1, A.1.2, and A.2

If, within 40 hours, the rod positions have not been determined, THERMAL POWER must be reduced to ≤ 50% RTP within 8 hours to avoid undesirable power distributions that could result from continued operation at > 50% RTP, if one or more rods are misaligned by more than 24 steps. The allowed Completion Time of 40 hours provides an acceptable period of time to verify the rod positions.

C.1.1 and C.1.2

or more

(s)

With one demand position indicator per bank inoperable, the rod positions can be determined by the DRPI System. Since normal power operation does not require excessive movement of rods, verification by administrative means that the rod position indicators are OPERABLE and the most withdrawn rod

B

RAI 3.1.8-1

Approved TSTF 136

BASES

ACTIONS (continued)

4 → ; ≤ 12 steps apart when demanded bank position is > 30 and < 215 steps, and ≤ 24 steps apart when demanded bank position is ≤ 30 or ≥ 215 steps

7 → Approved TSTF 136

and the least withdrawn rod are ~~≤ 12 steps apart~~ within the allowed Completion Time of once every 8 hours is adequate.

C.2

Reduction of THERMAL POWER to ≤ 50% RTP puts the core into a condition where rod position is not significantly affecting core peaking factor limits ~~(Ref. 3)~~. The allowed Completion Time of 8 hours provides an acceptable period of time to verify the rod positions per Required Actions C.1.1 and C.1.2 or reduce power to ≤ 50% RTP.

6 →

D.1

If the Required Actions cannot be completed within the associated Completion Time, the plant must be brought to a MODE in which the requirement does not apply. To achieve this status, the plant must be brought to at least MODE 3 within 6 hours. The allowed Completion Time is reasonable, based on operating experience, for reaching the required MODE from full power conditions in an orderly manner and without challenging plant systems.

B
RAI 3.1.8-1

Approved TSTF 136 → 7

SURVEILLANCE REQUIREMENTS

3 → Replace with Insert B 3.1.8-7

SR 3.1.8.1

Verification that the DRPI agrees with the demand position within [12] steps ensures that the DRPI is operating correctly. Since the DRPI does not display the actual shutdown rod positions between 18 and 210 steps, only points within the indicated ranges are required in comparison.

Approved TSTF 89 → Replace with Insert B 3.1.8-1

~~The [18 month] Frequency is based on the need to perform this Surveillance under the conditions that apply during a plant outage and the potential for unnecessary plant transients if the SR were performed with the reactor at power. Operating experience has shown these components usually pass the SR when performed at a Frequency of once every [18 months.] Therefore, the Frequency was concluded to be acceptable from a reliability~~

REFERENCES

1. ~~10 CFR 50, Appendix A, GDC 13.~~ ← FSAR, Section 7.1.2.
2. FSAR, Chapter ~~[15]~~ ← 14 ← 2

5 → FSAR, Section 7.1.2.

LCO 3.1.8 BASES INSERTS

Insert B 3.1.8-4:

By determining the non-indicating rod's position initially through use of the incore movable detectors, actual rod position is established with a high degree of certainty.

Insert B 3.1.8-5:

When one or more RPI channel(s) per group fails, the position of the rod can still be determined by use of the incore movable detectors (the incore is not effective for determining rod position until the power level is above approximately 5% RTP). By determining the non-indicating rod's position initially through use of the incore movable detectors, actual rod position is established with a high degree of certainty. Initial verification of RCCA position within the Completion Time of 8 hours is adequate for continued power operation above 50% of RTP, based on meeting the alignment requirements for the controls rod(s) prior to the individual position indicator becoming inoperable and the probability of a control rod becoming significantly out of position coincident with an event sensitive to that rod position is small.

B
RAI 3.1.8-2

Insert B 3.1.8-6:

After the initial position determination performed in Required Action A.1.1 above, Required Action A.1.2 requires periodic position verifications for control rods with inoperable individual position indicators once every 8 hours. Position verification can be performed by use of thermocouples, excore instrumentation, or the movable incore detectors. Based on experience, normal power operation does not require excessive movement of banks. If a bank has been significantly moved, Required Action B.1 below is required. Therefore, verification of RCCA position once every 8 hours is adequate for allowing continued power operation above 50% of RTP, since the probability of undetected rod misalignment and an event sensitive to that rod position is small.

Insert B 3.1.8-7:

A CHANNEL CALIBRATION of the individual rod position indicators is performed to ensure that the rod position indicators respond within the necessary range and accuracy.

No Significant Hazards Considerations - NUREG-1431 Section 3.01.08

24-May-00

NSHC Number	NSHC Text
A Rev. A	In accordance with the criteria set forth in 10 CFR 50.92, PBNP has evaluated this proposed Technical Specifications change and determined it does not represent a significant hazards consideration. The following is provided in support of this conclusion. 1. Does the change involve a significant increase in the probability or consequences of an accident previously evaluated? The proposed change involves reformatting and rewording of the current Technical Specifications. The reformatting and rewording process involves no technical changes to existing requirements. As such, this change is administrative in nature and does not impact initiators of analyzed events or assumed mitigation of accident or transient events. Therefore, this change does not increase 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 require a physical alteration of the plant (no new or different type of equipment will be installed) or changes in parameters governing normal plant operation. The proposed change will not impose any new or eliminate any old requirements. Thus, this change does not create the possibility of a new or different kind of accident from any accident previously evaluated. 3. Does this change involve a significant reduction in a margin of safety? The proposed change will not significantly reduce the margin of safety because it has no impact on any safety analysis assumptions. This change is administrative. As such, there is no technical change to the requirements and, therefore, there is no reduction in the margin of safety.

No Significant Hazards Considerations - NUREG-1431 Section 3.01.08

24-May-00

NSHC Number	NSHC Text
L.01 Rev. A	In accordance with the criteria set forth in 10 CFR 50.92, PBNP has evaluated this proposed Technical Specifications change and determined it does not represent a significant hazards consideration. The following is provided in support of this conclusion. 1. Does the change involve a significant increase in the probability or consequences of an accident previously evaluated? Changing the frequency of performance for a surveillance does not result in any hardware changes, nor does it significantly increase the probability of occurrence for initiation of any analyzed events since the function of the equipment has remained unchanged. Surveillance tests are intended to provide assurance of continued component operability. The frequency of performance of a surveillance does not significantly increase the consequences of an accident as a change in frequency does not change the response of the equipment in performing its specified function. 2. Does the change create the possibility of a new or different kind of accident from any accident previously evaluated? The proposed change does not require a physical alteration of the plant (no new or different type of equipment will be installed) or changes in parameters governing normal plant operation. The proposed change will still ensure compliance with the limiting condition for operation. Thus, this change does not create the possibility of a new or different kind of accident from any accident previously evaluated. 3. Does this change involve a significant reduction in a margin of safety? Agreement between the individual rod position indicators and the group demand position is required by the Technical Specifications to be verified once per 12 hours during unit operation. This test provides a verification of rod position indicator agreement for the current control rods position. Significant deviation between demanded and indicated position will result in evaluation of control rod position or position indicator operability, with corrective actions to reestablish proper indication or control rod position as necessary. Based on the rod position indicators being verified for agreement on a 12 hour frequency, it is highly unlikely that significant deviations or failures could go undetected. Based on the above, it has been concluded that this change does not represent a significant reduction in a margin of safety.
L.02 Rev. B	Not Used.

NSHC Number	NSHC Text
LA Rev. A	In accordance with the criteria set forth in 10 CFR 50.92, PBNP has evaluated this proposed Technical Specifications change and determined it does not represent a significant hazards consideration. The following is provided in support of this conclusion. 1. Does the change involve a significant increase in the probability or consequences of an accident previously evaluated? The proposed change relocates requirements from the Technical Specifications to the Bases, FSAR, or other plant controlled documents. The Bases and FSAR will be maintained using the provisions of 10 CFR 50.59. In addition to 10 CFR 50.59 provisions, the Technical Specifications Bases are subject to the change process in the Administrative Controls Chapter of the ITS. Plant procedures and other plant controlled documents are subject to controls imposed by plant administrative procedures, which endorse applicable regulations and standards. Changes to the Bases, FSAR, or other plant controlled documents will be evaluated in accordance with the requirements of the Bases Control Program in Chapter 5.0 of the ITS, 10 CFR 50.59, or plant administrative processes. Therefore, no increase in the probability or consequences of an accident previously evaluated will be allowed. 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 require a physical alteration of the plant (no new or different type of equipment will be installed) or changes in parameters governing normal plant operation. The proposed change will not impose any different requirements and adequate control of the information will be maintained. Thus, this change does not create the possibility of a new or different kind of accident from any accident previously evaluated. 3. Does this change involve a significant reduction in a margin of safety? The proposed change will not reduce a margin of safety because it has no impact on any safety analysis assumptions. In addition, the requirements to be moved from the Technical Specifications to the Bases, FSAR, or other plant controlled documents are as they currently exist. Future changes to the requirements in the Bases, FSAR, or other plant controlled documents will be evaluated in accordance with the requirements of 10 CFR 50.59, the Bases Control Program in Chapter 5.0 of the ITS, or the applicable plant process and no reduction in a margin of safety will be allowed.

No Significant Hazards Considerations - NUREG-1431 Section 3.01.08

24-May-00

NSHC Number	NSHC Text
M Rev. A	In accordance with the criteria set forth in 10 CFR 50.92, PBNP has evaluated this proposed Technical Specifications change and determined it does not represent a significant hazards consideration. The following is provided in support of this conclusion. 1. Does the change involve a significant increase in the probability or consequences of an accident previously evaluated? The proposed change provides more restrictive requirements for operation of the facility. These more stringent requirements do not result in operation that will increase the probability of initiating an analyzed event and do not alter the assumptions relative to the mitigation of an accident or transient event. These more restrictive requirements continue to ensure process variables, structures, systems and components are maintained consistent with the safety analyses. Therefore, this change does not increase 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 require a physical alteration of the plant (no new or different type of equipment will be installed) or changes in parameters governing normal plant operation. The proposed change does impose different requirements. However, these changes are consistent with assumptions made in the safety analysis. Thus, this change does not create the possibility of a new or different kind of accident from any accident previously evaluated. 3. Does this change involve a significant reduction in a margin of safety? The imposition of more restrictive requirements either has no affect on or increases the margin of safety. Each change is providing additional restrictions to enhance plant safety. These changes are consistent with the safety analysis. Therefore, this change does not involve a reduction in a margin of safety.

3.1 REACTIVITY CONTROL SYSTEMS

3.1.7 Rod Position Indication

LCO 3.1.7 Individual control rod position indication and bank demand indication shall be OPERABLE.

APPLICABILITY: MODES 1 and 2.

B RAI 3.1.8-2

ACTIONS

----------NOTE----------
Separate Condition entry is allowed for each inoperable rod position indicator and each bank demand position indicator per bank.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more RPI(s) per group inoperable for one or more groups.	A.1.1 Verify the position of the rods with inoperable position indicators by using movable incore detectors.	8 hours
	AND	
	A.1.2 Verify the position of the rods with inoperable position indicators.	Once per 8 hours
	OR	
	A.2 Reduce THERMAL POWER to ≤ 50% RTP.	8 hours B RAI 3.1.8-1
B. One or more rods with inoperable position indicators have been moved in excess of 24 steps in one direction since the last determination of the rod's position.	B.1 Verify the position of the rods with inoperable position indicators.	4 hours
	OR	B RAI 3.1.8-1

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
B. (continued)	B.2 Reduce THERMAL POWER to ≤ 50% RTP.	8 hours B RAI 3.1.8-1
C. One or more bank demand position indicator(s) per bank inoperable for one or more banks.	C.1.1 Verify by administrative means all RPIs for the affected banks are OPERABLE.	Once per 8 hours
	AND	
	C.1.2 Verify the most withdrawn rod and the least withdrawn rod of the affected banks are; ≤ 12 steps apart when demanded bank position is > 30 and < 215 steps, and ≤ 24 steps apart when demanded bank position is ≤ 30 and ≥ 215 steps.	Once per 8 hours
	OR	
	C.2 Reduce THERMAL POWER to ≤ 50% RTP.	8 hours B RAI 3.1.8-1
D. Required Action and associated Completion Time not met.	D.1 Be in MODE 3.	6 hours B RAI 3.1.8-1

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.1.7.1	Perform CHANNEL CALIBRATION of each RPI.	Once prior to criticality after each removal of the reactor head.

B 3.1 REACTIVITY CONTROL SYSTEM

B 3.1.7 Rod Position Indication

BASES

BACKGROUND

According to the Point Beach Design Criteria (Ref. 1), instrumentation to monitor variables and systems over their operating ranges during normal operation, anticipated operational occurrences, and accident conditions must be OPERABLE. LCO 3.1.7 is required to ensure OPERABILITY of the control rod position indicators to determine control rod positions and thereby ensure compliance with the control rod alignment and insertion limits.

The OPERABILITY, including position indication, of the shutdown and control rods is an initial assumption in all safety analyses that assume rod insertion upon reactor trip. Maximum rod misalignment is an initial assumption in the safety analysis that directly affects core power distributions and assumptions of available SDM. Rod position indication is required to assess OPERABILITY and misalignment.

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

Limits on control rod alignment and OPERABILITY have been established, and all rod positions are monitored and controlled during power operation to ensure that the power distribution and reactivity limits defined by the design power peaking and SDM limits are preserved.

Rod cluster control assemblies (RCCAs), or rods, are moved out of the core (up or withdrawn) or into the core (down or inserted) by their control rod drive mechanisms. The RCCAs are divided among control banks and shutdown banks. Each bank may be further subdivided into two groups to provide for precise reactivity control.

The axial position of shutdown rods and control rods are determined by two separate and independent systems: the Bank Demand Position Indication System (commonly called bank demand step counters) and the individual analog Rod Position Indication (RPI) System.

BASES

BACKGROUND (continued)

The Bank Demand Position Indication System counts the pulses from the Rod Control System that move the rods. There is one step counter for each group of rods. Individual rods in a group all receive the same signal to move and should, therefore, all be at the same position indicated by the group step counter for that group. The Bank Demand Position Indication System is considered highly precise (± 1 step or ± 5/8 inch). If a rod does not move one step for each demand pulse, the step counter will still count the pulse and incorrectly reflect the position of the rod.

The individual rod position indication system consists of three separate control room readouts; analog meters, digital displays, and the plant process computer. The position indication signal to each of these readouts is supplied by a linear variable differential transmitter (LVDT) which uses the control rod drive shaft to vary the amount of magnetic coupling between primary and secondary windings of the transformer. This generates an analog output signal proportional to actual control rod position. The analog display meters and the digital displays, provide position readouts in direct proportion to the output signal from the LVDT signal conditioning circuit. The process computer applies a polynomial to compensate for non-linearities in the LVDT system, providing for a more accurate position readout. Any one of these three readouts can be used for the purpose of verifying control rod position and alignment. The RPI system has an indication accuracy of 5% of span (11.5 steps); therefore, the maximum deviation between actual and demanded indication could be 24 steps or approximately 15 inches.

APPLICABLE SAFETY ANALYSES

Control and shutdown rod position accuracy is essential during power operation. Power peaking, ejected rod worth, or SDM limits may be violated in the event of a Design Basis Accident (Ref. 2), with control or shutdown rods operating outside their limits undetected. Therefore, the acceptance criteria for rod position indication is that rod positions must be known with sufficient accuracy in order to verify the core is operating within the group sequence, overlap, design peaking limits, ejected rod worth, and with minimum SDM (LCO 3.1.5, "Shutdown Bank Insertion Limits, " and LCO 3.1.6, "Control Bank Insertion Limits"). The rod positions must also be known in order to verify the alignment limits are preserved (LCO 3.1.4, "Rod Group Alignment Limits"). Control rod positions are continuously monitored to provide operators with information that ensures the plant is operating within the bounds of the accident analysis assumptions.

The control rod position indicator channels satisfy Criterion 2 of the NRC Policy Statement. The control rod position indicators monitor control rod position, which is an initial condition of the accident.

BASES

LCO

LCO 3.1.7 specifies that one RPI System and one Bank Demand Position Indication System be OPERABLE for each control rod.

OPERABILITY of the position indicator channels ensures that inoperable, misaligned, or mispositioned control rods can be detected. Therefore, power peaking, ejected rod worth, and SDM can be controlled within acceptable limits.

A deviation of less than the allowable limit, given in LCO 3.1.4, in position indication for a single control rod, ensures high confidence that the position uncertainty of the corresponding control rod group is within the assumed values used in the analysis (that specified control rod group insertion limits).

These requirements ensure that control rod position indication during power operation and PHYSICS TESTS is accurate, and that design assumptions are not challenged.

APPLICABILITY

The requirements on the RPI and step counters are only applicable in MODES 1 and 2 (consistent with LCO 3.1.4, LCO 3.1.5, and LCO 3.1.6), because these are the only MODES in which power is generated, and the OPERABILITY and alignment of rods have the potential to affect the safety of the plant. In the shutdown MODES, the OPERABILITY of the shutdown and control banks has the potential to affect the required SDM, but this effect can be compensated for by an increase in the boron concentration of the Reactor Coolant System. B RAI 3.1.8-2

ACTIONS

The ACTIONS table is modified by a Note indicating that a separate Condition entry is allowed for each inoperable rod position indicator per group and each demand position indicator per bank. This is acceptable because the Required Actions for each Condition provide appropriate compensatory actions for each inoperable position indicator.

A.1.1 B RAI 3.1.8-1

When one or more RPI channel(s) per group fails, the position of the rod can still be determined by use of the incore movable detectors (the incore is not effective for determining rod position until the power level is above approximately 5% RTP). By determining the non-indicating rod's position initially through use of the incore movable detectors, actual rod position is established with a high degree of certainty. Initial verification of RCCA position within the Completion Time of 8 hours is adequate for continued power operation above 50% of RTP, based on meeting the alignment requirements for the controls rod(s) prior to the individual position indicator becoming inoperable and B RAI 3.1.8-2

BASES

ACTIONS (continued) the probability of a control rod becoming significantly out of position coincident with an event sensitive to that rod position is small.

A.1.2

B
RAI 3.1.8-1

After the initial position determination performed in Required Action A.1.1 above, Required Action A.1.2 requires periodic position verifications for control rods with inoperable individual position indicators once every 8 hours. Position verification can be performed by use of thermocouples, excore instrumentation, or the movable incore detectors. Based on experience, normal power operation does not require excessive movement of banks. If a bank has been significantly moved, Required Action B.1 below is required. Therefore, verification of RCCA position once every 8 hours is adequate for allowing continued power operation above 50% of RTP, since the probability of undetected rod misalignment and an event sensitive to that rod position is small.

A.2

B
RAI 3.1.8-1

Reduction of THERMAL POWER to ≤ 50% RTP puts the core into a condition where rod position is not significantly affecting core peaking factors.

The allowed Completion Time of 8 hours is reasonable, based on operating experience, for reducing power to ≤ 50% RTP from full power conditions without challenging plant systems and allowing for rod position determination by Required Actions A.1.1 and A.1.2 above.

B.1 and B.2

These Required Actions clarify that when one or more rods with inoperable position indicators have been moved in excess of 24 steps in one direction, since the position was last determined, the Required Actions of A.1.1, A.1.2 and A.2 are still appropriate but must be initiated promptly under Required Action B.1 to begin verifying that these rods are still properly positioned, relative to their group positions.

If, within 4 hours, the rod positions have not been determined, THERMAL POWER must be reduced to ≤ 50% RTP within 8 hours to avoid undesirable power distributions that could result from continued operation > 50% RTP, if one or more rods are misaligned by more than 24 steps. The allowed Completion Time of 4 hours provides an acceptable period of time to verify the rod positions.

C.1.1 and C.1.2

With one or more demand position indicator(s) per bank inoperable, the

rod positions can be determined by the RPI System. Since normal power operation does not require excessive movement of rods, verification by administrative means that the rod position indicators are OPERABLE and the most withdrawn rod and the least withdrawn rod are; ≤ 12 steps apart when demanded bank position is > 30 and < 215 steps, and ≤ 24 steps apart when demanded bank position is ≤ 30 or ≥ 215 steps within the allowed Completion Time of once every 8 hours is adequate.

C.2

B

RAI 3.1.8-1

Reduction of THERMAL POWER to ≤ 50% RTP puts the core into a condition where rod position is not significantly affecting core peaking factor limits. The allowed Completion Time of 8 hours provides an acceptable period of time to verify the rod positions per Required Actions C.1.1 and C.1.2 or reduce power to ≤ 50% RTP.

D.1

If the Required Actions cannot be completed within the associated Completion Time, the plant must be brought to a MODE in which the requirement does not apply. To achieve this status, the plant must be brought to at least MODE 3 within 6 hours. The allowed Completion Time is reasonable, based on operating experience, for reaching the required MODE from full power conditions in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.1.7.1

A CHANNEL CALIBRATION of the individual rod position indicators is performed to ensure that the rod position indicators respond within the necessary range and accuracy.

This surveillance is performed prior to reactor criticality after each removal of the reactor head as there is potential for unnecessary plant transients if the SR were performed with the reactor at power.

REFERENCES

1. FSAR. Section 7.1.2.

2. FSAR. Chapter 14.

Justification For Deviations - NUREG-1431 Section 3.01.10

01-Jun-00

JFD Number	JFD Text
01 Rev. A	The brackets have been removed and the proper plant specific information has been provided. In some instances, even though the information was designated as being site specific information in the LCO (bracketed), the corresponding Bases information was not bracketed. These cases are self evident, corresponding to the bracketed information in the LCO and have had the appropriate site specific information provided.

ITS:	NUREG:
B 3.01.08	B 3.01.10
LCO 3.01.08	LCO 3.01.10
SR 3.01.08.01	SR 3.01.10.02

02 Rev. A	NUREG 1431 SR 3.1.10.1 requires the performance of a Channel Operational Test (COT) on both the Power Range (PR) and Intermediate Range (IR) channels prior to the initiation of Physics Testing. The Bases states that performance of the COT ensures that the Reactor Trip System (RTS) is properly aligned to provide the required degree of core protection. The Point Beach CTS does not contain any requirements to perform channel testing prior to the performance of physics tests. The PR and IR channels are required to be operable in accordance with proposed ITS LCO 3.3.1 during the performance of Mode 2 Physics Tests. Accordingly, LCO 3.3.1 provides assurance that the channels are aligned and operable by performance of the surveillances associated with this LCO. In addition, channel checks are required on a 12 hour frequency for both the PR and IR channels during Mode 2. Performance of a COT prior to the initiation of physics testing is an unnecessary burden based on other qualitative means available for determining channel alignment (e.g. bi-stable indications, channel checks, and calibrations and functional tests completed within their required frequencies) and the fact that meeting LCO 3.3.1 Surveillance Requirements already provides a high degree of confidence that the PR and IR channels are operable. Based on the above information, NUREG 1431 SR 3.1.10.1 was not adopted as part of the Point Beach conversion to the Improved Technical Specifications. This change is consistent with approved TSTF 13. This change required the surveillances associated with this LCO to be renumbered to maintain continuity in the SR numbering scheme.

ITS:	NUREG:
B 3.01.08	B 3.01.10
N/A	SR 3.01.10.01
SR 3.01.08.01	SR 3.01.10.02
SR 3.01.08.02	SR 3.01.10.03
SR 3.01.08.03	SR 3.01.10.04

Justification For Deviations - NUREG-1431 Section 3.01.10

01-Jun-00

JFD Number	JFD Text
03 Rev. A	NUREG 1431 specifies two tables which specify the location in the FSAR which provides a listing of initial startup tests. This information is bracketed as site specific information. Point Beach lists all of the initial start up tests in a single table of the FSAR. Therefore, reference to only one table (13.2.2-1) is necessary. The proposed ITS have been modified to list only this single table reference. **ITS:** B 3.01.08 **NUREG:** B 3.01.10
04 Rev. B	The Bases of NUREG 1431 LCO 3.1.10 describes the method for performing several physics tests which must be repeated for all control rod banks (both shutdown and control). The NUREG Bases is written in a fashion that could imply performance of these tests for the control banks only. As such, the proposed ITS Bases has been changed to clarify applicability to both the shutdown and control banks. **ITS:** B 3.01.08 **NUREG:** B 3.01.10
05 Rev. A	The wording has been modified to conform with the proposed plant specific Technical Specifications. See ITS LCO Section 3.3.1. **ITS:** B 3.01.08 **NUREG:** B 3.01.10 LCO 3.01.08 LCO 3.01.10
06 Rev. A	The Westinghouse Dynamic Rod Worth Measurement Technique as described in WCAP 13360-P-A is also an approved method for determining control rod worth. This method was approved by the NRC and SER issued on January 5, 1996. A short description of this method has been added as a fourth means of performing the Control Rod Worth Test. When this method is used, it is not necessary to perform the Critical Boron Concentration - Control Rods Inserted as this is information is not used for reactor core design validation per the ANS-19.6.1 requirements. Appropriate bases changes have been made to reflect the use of Dynamic Rod Worth Measurement Technique. **ITS:** B 3.01.08 **NUREG:** B 3.01.10

BASES

BACKGROUND (Continued)

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

The PHYSICS TESTS required for reload fuel cycles (Ref. 4) in MODE 2 are listed below:

a. Critical Boron Concentration – Control Rods Withdrawn;

b. Critical Boron Concentration – Control Rods Inserted;

c. Control Rod Worth;

d. Isothermal Temperature Coefficient (ITC); and

e. Neutron Flux Symmetry.

The first four tests are performed in MODE 2, and the last test can be performed in either MODE 1 or 2. These and other supplementary tests may be required to calibrate the nuclear instrumentation or to diagnose operational problems. These tests may cause the operating controls and process variables to deviate from their LCO requirements during their performance.

1

1

a. The Critical Boron Concentration – Control Rods Withdrawn Test measures the critical boron concentration at hot zero power (HZP). With all rods out, the lead control bank is at or near its fully withdrawn position. HZP is where the core is critical (k_{eff} = 1.0), and the Reactor Coolant System (RCS) is at design temperature and pressure for zero power. Performance of this test should not violate any of the referenced LCOs.

b. The Critical Boron Concentration – Control Rods Inserted Test measures the critical boron concentration at HZP, with a bank having a worth of at least 1% $\Delta k/k$ when fully inserted into the core. This test is used to measure the boron reactivity coefficient. With the core at HZP and all banks fully withdrawn, the boron concentration of the reactor coolant is gradually lowered in a continuous manner. The selected bank is

BASES

BACKGROUND (Continued)

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

B RAI 3.1.10-1

The PHYSICS TESTS required for reload fuel cycles (Ref. 4) in MODE 2 are listed below:

a. Critical Boron Concentration – Control Rods Withdrawn;

b. Critical Boron Concentration – Control Rods Inserted;

c. Control Rod Worth;

d. Isothermal Temperature Coefficient (ITC); and

e. Neutron Flux Symmetry.

The first four tests are performed in MODE 2, and the last test can be performed in either MODE 1 or 2. These and other supplementary tests may be required to calibrate the nuclear instrumentation or to diagnose operational problems. These tests may cause the operating controls and process variables to deviate from their LCO requirements during their performance.

1

1

a. The Critical Boron Concentration – Control Rods Withdrawn Test measures the critical boron concentration at hot zero power (HZP). With all rods out, the lead control bank is at or near its fully withdrawn position. HZP is where the core is critical (k_{eff} = 1.0), and the Reactor Coolant System (RCS) is at design temperature and pressure for zero power. Performance of this test should not violate any of the referenced LCOs.

b. The Critical Boron Concentration – Control Rods Inserted Test measures the critical boron concentration at HZP, with a bank having a worth of at least 1% $\Delta k/k$ when fully inserted into the core. This test is used to measure the boron reactivity coefficient. With the core at HZP and all banks fully withdrawn, the boron concentration of the reactor coolant is gradually lowered in a continuous manner. The selected bank is

B 3.1 REACTIVITY CONTROL SYSTEMS

B 3.1.8 PHYSICS TESTS Exceptions - MODE 2

BASES

BACKGROUND

The primary purpose of the MODE 2 PHYSICS TESTS exceptions is to permit relaxations of existing LCOs to allow certain PHYSICS TESTS to be performed.

Section XI of 10 CFR 50, Appendix B (Ref. 1), requires that a test program be established to ensure that structures, systems, and components will perform satisfactorily in service. All functions necessary to ensure that the specified design conditions are not exceeded during normal operation and anticipated operational occurrences must be tested. This testing is an integral part of the design, construction, and operation of the plant. Requirements for notification of the NRC, for the purpose of conducting tests and experiments, are specified in 10 CFR 50.59 (Ref. 2).

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

a. Ensure that the facility has been adequately designed;

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

c. Verify the assumptions used to predict unit response;

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

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

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

B
RAI 3.1.10-1

PHYSICS TESTS procedures are written and approved in accordance with established formats. The procedures include all information necessary to permit a detailed execution of the testing required to ensure that the design intent is met. PHYSICS TESTS are performed in accordance with these procedures and test results are approved prior to continued power escalation and long term power operation.

BASES

BACKGROUND (continued)

The PHYSICS TESTS required for reload fuel cycles (Ref. 4) in MODE 2 are listed below:

a. Critical Boron Concentration—Control Rods Withdrawn;

b. Critical Boron Concentration—Control Rods Inserted;

c. Control Rod Worth;

d. Isothermal Temperature Coefficient (ITC); and

e. Neutron Flux Symmetry.

The first four tests are performed in MODE 2, and the last test can be performed in either MODE 1 or 2. These and other supplementary tests may be required to calibrate the nuclear instrumentation or to diagnose operational problems. These tests may cause the operating controls and process variables to deviate from their LCO requirements during their performance.

a. The Critical Boron Concentration—Control Rods Withdrawn Test measures the critical boron concentration at hot zero power (HZP). With all rods out, the lead control bank is at or near its fully withdrawn position. HZP is where the core is critical (k_{eff} = 1.0), and the Reactor Coolant System (RCS) is at design temperature and pressure for zero power. Performance of this test should not violate any of the referenced LCOs.

b. The Critical Boron Concentration—Control Rods Inserted Test measures the critical boron concentration at HZP, with a bank having a worth of at least 1% $\Delta k/k$ when fully inserted into the core. This test is used to measure the boron reactivity coefficient. With the core at HZP and all banks fully withdrawn, the boron concentration of the reactor coolant is gradually lowered in a continuous manner. The selected bank is then inserted to make up for the decreasing boron concentration until the selected bank has been moved over its entire range of travel. The reactivity resulting from each incremental bank movement is measured with a reactivity computer. The difference between the measured critical boron concentration with all rods fully withdrawn and with the bank inserted is determined. The boron reactivity coefficient is determined by dividing the measured bank worth by the measured boron concentration difference. Performance of this test could violate LCO 3.1.4, "Rod Group Alignment Limits"; LCO 3.1.5, "Shutdown Bank Insertion Limit"; or LCO 3.1.6, "Control Bank Insertion Limits." When the control rod worth test is performed using the Dynamic Rod Worth Measurement Technique, it is not

BASES

BACKGROUND (continued)

necessary to determine Critical Boron Concentration – Control Rods Inserted as it is not used for design validation per the ANS-19.6.1 requirements.

c. The Control Rod Worth Test is used to measure the reactivity worth of selected control banks. This test is performed at HZP and has four alternative methods of performance. The first method, the Boron Exchange Method, varies the reactor coolant boron concentration and moves the selected control bank in response to the changing boron concentration. The reactivity changes are measured with a reactivity computer. This sequence is repeated for the remaining control rod banks. The second method, the Rod Swap Method, measures the worth of a predetermined reference bank using the Boron Exchange Method above. The reference bank is then nearly fully inserted into the core. The selected bank is then inserted into the core as the reference bank is withdrawn. The HZP critical conditions are then determined with the selected bank fully inserted into the core. The worth of the selected bank is inferred, based on the position of the reference bank with respect to the selected bank. This sequence is repeated as necessary for the remaining control rod banks. The third method, the Boron Endpoint Method, moves the selected control bank over its entire length of travel and then varies the reactor coolant boron concentration to achieve HZP criticality again. The difference in boron concentration is the worth of the selected control bank. This sequence is repeated for the remaining control rod banks. The fourth method, the Westinghouse Dynamic Rod Worth Measurement Technique (Ref. 7), is accomplished by inserting and withdrawing the bank at the maximum stepping speed, without changing boron concentration, and recording the signals on the excore detectors. The recorded signals are processed on a reactivity computer, which solves the inverse point kinetics equation with proper analytical compensation for spatial effects. Performance of this test could violate LCO 3.1.4, LCO 3.1.5, or LCO 3.1.6.

d. The ITC Test measures the ITC of the reactor. This test is performed at HZP and has two methods of performance. The first method, the Slope Method, varies RCS temperature in a slow and continuous manner. The reactivity change is measured with a reactivity computer as a function of the temperature change. The ITC is the slope of the reactivity versus the temperature plot. The test is repeated by reversing the direction of the temperature change, and the final ITC is the average of the two calculated ITCs. The second method, the Endpoint Method, changes the RCS temperature and measures the reactivity at the beginning and end of the temperature change. The ITC is the total reactivity change divided by the total temperature change. The test is repeated by

BASES

BACKGROUND (continued)

reversing the direction of the temperature change, and the final ITC is the average of the two calculated ITCs. Performance of this test could violate LCO 3.4.2, "RCS Minimum Temperature for Criticality."

e. The Flux Symmetry Test measures the degree of azimuthal symmetry of the neutron flux at as low a power level as practical, depending on the test method employed. This test can be performed at HZP (Control Rod Worth Symmetry Method) or at ≤ 30% RTP (Flux Distribution Method). The Control Rod Worth Symmetry Method inserts a control bank, which can then be withdrawn to compensate for the insertion of a single control rod from a symmetric set. The symmetric rods of each set are then tested to evaluate the symmetry of the control rod worth and neutron flux (power distribution). A reactivity computer is used to measure the control rod worths. Performance of this test could violate LCO 3.1.4, LCO 3.1.5, or LCO 3.1.6. The Flux Distribution Method uses the incore flux detectors to measure the azimuthal flux distribution at selected locations with the core at ≤ 30% RTP.

APPLICABLE SAFETY ANALYSES

The fuel is protected by LCOs that preserve the initial conditions of the core assumed during the safety analyses. The methods for development of the LCOs that are excepted by this LCO are described in the Westinghouse Reload Safety Evaluation Methodology Report (Ref. 5). The above mentioned PHYSICS TESTS, and other tests that may be required to calibrate nuclear instrumentation or to diagnose operational problems, may require the operating control or process variables to deviate from their LCO limitations.

The FSAR defines requirements for initial testing of the facility, including PHYSICS TESTS. Table 13.2.2-1 summarize the zero, low power, and power tests. Requirements for reload fuel cycle PHYSICS TESTS are defined in ANSI/ANS-19.6.1-1985 (Ref. 4). Although these PHYSICS TESTS are generally accomplished within the limits for all LCOs, conditions may occur when one or more LCOs must be suspended to make completion of PHYSICS TESTS possible or practical. This is acceptable as long as the fuel design criteria are not violated. When one or more of the requirements specified in LCO 3.1.3, "Moderator Temperature Coefficient (MTC)," LCO 3.1.4, LCO 3.1.5, LCO 3.1.6, and LCO 3.4.2 are suspended for PHYSICS TESTS, the fuel design criteria are preserved as long as the power level is limited to ≤ 5% RTP, the reactor coolant temperature is kept ≥ 530°F, and SDM is within the limits provided in the COLR.

The PHYSICS TESTS include measurement of core nuclear parameters or the exercise of control components that affect process variables. Among the process variables involved are AFD and QPTR,

BASES

APPLICABLE SAFETY ANALYSES (continued)

which represent initial conditions of the unit safety analyses. Also involved are the movable control components (control and shutdown rods), which are required to shut down the reactor. The limits for these variables are specified for each fuel cycle in the COLR. As described in LCO 3.0.7, compliance with Test Exception LCOs is optional, and therefore no criteria of the NRC Policy Statement apply. Test Exception LCOs provide flexibility to perform certain operations by appropriately modifying requirements of other LCOs. A discussion of the criteria satisfied for the other LCOs is provided in their respective bases.

Reference 6 allows special test exceptions (STEs) to be included as part of the LCO that they affect. It was decided, however, to retain this STE as a separate LCO because it was less cumbersome and provided additional clarity.

LCO

This LCO allows the reactor parameters of MTC and minimum temperature for criticality to be outside their specified limits. In addition, it allows selected control and shutdown rods to be positioned outside of their specified alignment and insertion limits. One Power Range Neutron Flux channel may be bypassed, reducing the number of required channels from "4" to "3". Operation beyond specified limits is permitted for the purpose of performing PHYSICS TESTS and poses no threat to fuel integrity, provided the SRs are met.

The requirements of LCO 3.1.3, LCO 3.1.4, LCO 3.1.5, LCO 3.1.6, and LCO 3.4.2 may be suspended and the number of required channels for LCO 3.3.1, "RTS Instrumentation," Functions 2, 5, and 17.d, may be reduced to "3" required channels, during the performance of PHYSICS TESTS provided:

a. RCS lowest loop average temperature is ≥ 530°F;

b. SDM is within the limits provided in the COLR; and

c. THERMAL POWER is ≤ 5% RTP.

APPLICABILITY

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

BASES

ACTIONS

A.1 and A.2

If the SDM requirement is not met, boration must be initiated promptly. A Completion Time of 15 minutes is adequate for an operator to correctly align and start the required systems and components. The operator should begin boration with the best source available for the plant conditions. Boration will be continued until SDM is within limit.

Suspension of PHYSICS TESTS exceptions requires restoration of each of the applicable LCOs to within specification.

B.1

When THERMAL POWER is > 5% RTP, the only acceptable action is to open the reactor trip breakers (RTBs) to prevent operation of the reactor beyond its design limits. Immediately opening the RTBs will shut down the reactor and prevent operation of the reactor outside of its design limits.

C.1

When the RCS lowest T_{avg} is < 530°F, the appropriate action is to restore T_{avg} to within its specified limit. The allowed Completion Time of 15 minutes provides time for restoring T_{avg} to within limits without allowing the plant to remain in an unacceptable condition for an extended period of time. Operation with the reactor critical and with temperature below 530°F could violate the assumptions for accidents analyzed in the safety analyses.

D.1

If the Required Actions cannot be completed within the associated Completion Time, the plant must be brought to a MODE in which the requirement does not apply. To achieve this status, the plant must be brought to at least MODE 3 within an additional 15 minutes. The Completion Time of 15 additional minutes is reasonable, based on operating experience, for reaching MODE 3 in an orderly manner and without challenging plant systems.

BASES

SURVEILLANCE REQUIREMENTS

SR 3.1.8.1

Verification that the RCS lowest loop T_{avg} is ≥ 530°F will ensure that the unit is not operating in a condition that could invalidate the safety analyses. Verification of the RCS temperature at a Frequency of 30 minutes during the performance of the PHYSICS TESTS will ensure that the initial conditions of the safety analyses are not violated.

SR 3.1.8.2

Verification that THERMAL POWER is ≤ 5% RTP will ensure that the plant is not operating in a condition that could invalidate the safety analyses. Verification of THERMAL POWER at a Frequency of 30 minutes during the performance of PHYSICS TESTS will ensure that the initial conditions of the safety analyses are not violated.

SR 3.1.8.3

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

a. RCS boron concentration;

b. Control bank position;

c. RCS average temperature;

d. Fuel burnup based on gross thermal energy generation;

e. Xenon concentration;

f. Samarium concentration;

g. Isothermal temperature coefficient (ITC), when below the Point of Adding Heat (POAH);

h. Moderator Defect, when above the POAH; and

i. Doppler Defect, when above the POAH.

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

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

BASES

REFERENCES

1. 10 CFR 50, Appendix B, Section XI.
2. 10 CFR 50.59.
3. Regulatory Guide 1.68, Revision 2, August, 1978.
4. ANSI/ANS-19.6.1-1985, December 13, 1985.
5. WCAP-9273-NP-A, "Westinghouse Reload Safety Evaluation Methodology Report," July 1985.
6. WCAP-11618, including Addendum 1, April 1989.
7. WCAP-13360-P-A, "Westinghouse Dynamic Rod Worth Measurement Technique," January 1996.

DOC Number	DOC Text
LA.01 Rev. B	The CTS specifies both the power distribution limit and its associated formula within the Technical Specifications. The formulas are contained in WCAP 10216 (P) and WCAP 10216 (P-A), which address the NRC approved methodology for FQ(Z) surveillance testing, while the FQ limits are being proposed for inclusion into the Core Operating Limits Report (COLR). This formula is adequately controlled outside of the Technical Specifications as it is controlled via inclusion in a document which establishes the methodology approved by the NRC for use. FQ(Z) is calculated in accordance with WCAP-10216 P-A, Revision 1A, Relaxation of Constant Axial Offset Control, February 1994 which is controlled via its inclusion into the Administrative Control Section of the ITS (section 5 - COLR). Any changes to the formulas require a revision to the NRC approved methodology which would require a Technical Specification change. Therefore, even though the formulas are removed from the Technical Specifications, control has remained unchanged. The FQ limit and the K(Z) figures (15.3.10-3 and 15.3.10-3a) have been proposed for inclusion into a Core Operating Limits Report (COLR). The Technical Specifications continue to require compliance with the FQ limitation itself, this change simply allows the limits to be changed on a cycle specific basis without requiring prior NRC review and approval of the revised limits. This change will alleviate an unnecessary burden on both the NRC and the licensee with no decrease in the level of safety. The limitations contained within the COLR will be calculated in accordance with an NRC approved methodology (WCAP 10216) which is controlled via its inclusion in the ITS Section 5. Specific Reporting Requirements to notify the NRC when ever changes are made to the COLR limits has been proposed in ITS Section 5 consistent with NUREG 1431 and NRC Generic Letter 88-16. The limits associated with this specification are not required to be in the ITS to provide adequate protection of public health and safety, since the ITS will still retain a requirement to maintain compliance with FQ limit itself, which will be calculated in accordance with an NRC approved methodology. Changes to the COLR limits will be controlled in accordance with the 10 CFR 50.59 process thereby, ensuring that all applicable limits of the safety analysis will continue to be met.

CTS:	ITS:
15.03.10 F 15.03.10-03	COLR
15.03.10.E.01.A	COLR
	COLR

< See Section 3.1 >

A.1

15.3.10 CONTROL ROD AND POWER DISTRIBUTION LIMITS

Applicability

Applies to the operation of the control rods and to core power distribution limits. A.5

Objective

< See Section 3.1 >

To insure (1) core subcriticality after a reactor trip, (2) a limit on potential reactivity insertions from a hypothetical rod cluster control assembly (RCCA) ejection, and (3) an acceptable core power distribution during power operation. A.6

Specification

A. SHUTDOWN MARGIN

1. The shutdown margin shall exceed the applicable value as shown in Figure 15.3.10-2 under all steady-state operating conditions from 350°F to full power. If the shutdown margin is less than the applicable value of Figure 15.3.10-2, within 15 minutes initiate boration to restore the shutdown margin.

2. A shutdown margin of at least 1% Δk/k shall be maintained when the reactor coolant temperature is less than 350°F. If the shutdown margin is less than this limit, within 15 minutes initiate boration to restore the shutdown margin.

B. ROD OPERABILITY AND BANK ALIGNMENT LIMITS

1. During power and low power operation, all shutdown and control rods shall be operable, with all individual indicated rod positions within twelve steps of their bank demand position, except when the bank demand position is ≤30 steps or ≥215 steps. In this case, all individual indicated rod positions shall be within 24 steps of their bank demand position.

If an RCCA does not step in upon demand, up to six hours is allowed to determine whether the problem with stepping is an electrical problem. If the problem cannot be resolved within six hours, the RCCA shall be declared inoperable until it has been verified that it will step in or would drop upon demand.

a. Rod Operability Requirements

(1) If one rod is determined to be untrippable, perform the following actions:

< See Section 3.1 >

< See LCO 3.1.6 and 3.1.7 >

A.1

AND

b. Within two hours fully withdraw the shutdown banks.

c. If the above actions and associated completion times are not met, be in hot shutdown within the following six hours.

2. When the reactor is critical, the control banks shall be inserted no further than the limits shown by the lines on Figure 15.3.10-1. If this condition is not met, perform the following actions:

a. Within one hour verify that the shutdown margin exceeds the applicable value as shown in Figure 15.3.10-2; OR within one hour restore the shutdown margin by boration;

AND

b. Within two hours restore the control banks to within limits.

c. If the above actions and associated completion times are not met, be in hot shutdown within the following six hours.

L.1

Clarify Mode of Applicability to be Mode 1

E. POWER DISTRIBUTION LIMITS

< See LCO 3.2.2 >

L.4

1. Hot Channel Factors

LA.1

be within the limits specified in the COLR.

as approximated by $F_Q^C(Z)$ and $F_Q^W(Z)$,

a. The hot channel factors ~~defined in the basis~~ shall meet the following limits:

	For OFA and Upgraded OFA Fuel	For 422V+ Fuel
for P > 0.5	$F_Q(Z) \leq (2.50)/P \times K(Z)$	$F_Q(Z) \leq (2.60)/P \times K(Z)$
for P < 0.5	$F_Q(Z) \leq 5.00 \times K(Z)$	$F_Q(Z) \leq 5.20 \times K(Z)$
	$F^N_{\Delta H} < 1.70 \times [1 + 0.3\,(1-P)]$	$F^N_{\Delta H} < 1.77 \times [1 + 0.3\,(1-P)]$

B

Amendment 193/198

LA.1

Where P is the fraction of full power at which the core is ~~operating~~, K(Z) is the function in Figure 15.3.10-3 ~~or Figure~~ 15.3.10-3a, as applicable, and Z is the ~~core height~~ location of F_Q.

Cond A - RA A.1

b. If $F_Q(Z)$ exceeds the limit of Specification 15.3.10.E.1.a, within fifteen ~~minutes~~ reduce thermal power ~~until $F_Q(Z)$ limits are satisfied~~;

$F_Q^C(Z)$

(1) ~~After thermal power has been reduced in accordance with Specification 15.3.10.E.1.b, perform the following actions:~~

L.4

A.2

> 1% RTP for each 1% $F_Q^C(Z)$ exceeds it limit

M.1

B

Amendment 193/198

Cond A - RA A.2

Reduce Power Range Neutron Flux-High trip setpoints ≥ 1% for each 1% $F_Q^C(Z)$ exceeds limit.

L.2 → 72 hours → A.1

(a) Within eight hours reduce the full power Power Range Neutron Flux - High trip setpoints by an amount equivalent to the power reduction required in Specification 15.3.10.E.1.b;

AND

M.2

≥ 1% for each 1% $F_Q^C(Z)$ exceeds limit.

(b) Within 72 hours reduce Overpower and Overtemperature ΔT trip setpoints by an amount equivalent to the power reduction required in Specification 15.3.10.E.1.b; ← L.3

AND

Cond A - RA A.3

(c) Verify that $F_Q(Z)$ will be within limits for the increased power level prior to increasing any setpoints that have been reduced and thermal power above the limit specified in Specification 15.3.10.E.1.b.

and $F_Q^W(Z)$

Cond A - Note and RA A.4 Completion Time

A.3

(d) If the above actions and associated completion times are not met, be in low power operation within the following six hours.

Mode 2 ← L.1/L.5

L.4

Cond C and RA C.1

c. If $F^N_{\Delta H}$ exceeds the limit of Specification 15.3.10.E.1.a, within four hours reduce thermal power to restore $F^N_{\Delta H}$ to within limits OR perform the following actions:

(1) Within four hours reduce thermal power to ≤50 percent rated thermal power;

AND

(2) Within eight hours reduce the full power Power Range Neutron Flux - High trip setpoints to ≤58 percent rated thermal power.

In addition to the above actions, the following actions shall also be performed during the subsequent power escalation if $F^N_{\Delta H}$ had exceeded the limit of Specification 15.3.10.E.1.a:

(3) Verify that $F^N_{\Delta H}$ is within limits within 24 hours;

AND

(4) Verify that $F^N_{\Delta H}$ is within limits prior to thermal power exceeding 50 percent of rated thermal power;

AND

(5) Verify that $F^N_{\Delta H}$ is within limits prior to thermal power exceeding 75 percent of rated thermal power;

AND

(6) Verify that $F^N_{\Delta H}$ is within limits within 24 hours after reaching ≥95 percent of rated thermal power.

Insert new Condition B, Note, and Required Action B.1 - B.4 $F_Q^W(Z)$ not within limit

L.4

< See LCO 3.2.2 >

A.4

Power Distribution

During power operation, the global power distribution is limited by TS 15.3.10.E.2, "Axial Flux Difference," and TS 15.3.10.E.3, "Quadrant Power Tilt," which are directly and continuously measured process variables. These specifications, along with TS 15.3.10.D, "Bank Insertion Limits," maintain the core limits on power distributions on a continuous basis.

As a result of the increased peaking factors allowed by the new 422V+ fuel, a new column was added to TS 15.3.10.E.1.a. The full power $F^N_{\Delta H}$ peaking factor design limit (radial peaking factor) for 422V+ fuel will increase to 1.77 from the 1.70 value for the OFA fuel. The maximum $F_Q(Z)$ peaking factor limit (total peaking factor) for 422V+ fuel will increase to 2.60 from the 2.50 value for the OFA fuel. The OFA fuel design will retain the current $F^N_{\Delta H}$ and $F_Q(Z)$ peaking factors of 1.70 and 2.50, respectively. In addition, the K(Z) envelope for the new 422V+ fuel was modified and a new TS figure 15.3.10-3a was developed and inserted in the Technical Specifications. The K(Z) envelope in TS Figure 15.3.10-3 remains for the OFA fuel. B Amendm 193/198

The purpose of the limits on the values of $F_Q(Z)$, the height dependent heat flux hot channel factor, is to limit the local peak power density. The value of $F_Q(Z)$ varies along the axial height (Z) of the core.

$F_Q(Z)$ is defined as the maximum local fuel rod linear power density divided by the average fuel rod linear power density, assuming nominal fuel pellet and fuel rod dimensions. Therefore, $F_Q(Z)$ is a measure of the peak fuel pellet power within the reactor core.

$F_Q(Z)$ varies with fuel loading patterns, control bank insertion, fuel burnup, and changes in axial power distribution. $F_Q(Z)$ is measured periodically using the incore detector system. These measurements are generally taken with the core at or near steady state conditions.

The purpose of the limits on $F^N_{\Delta H}$, the nuclear enthalpy rise hot channel factor, is to ensure that the fuel design criteria are not exceded and the accident analysis assumptions remain valid. The design limits on local and integrated fuel rod peak power density are expressed in terms of hot channel factors. Control of the core power distribution with respect to these factors ensures that local conditions in the fuel rods and coolant channels do not challenge core integrity at any location during either normal operation or a postulated accident analyzed in the safety analyses.

$F^N_{\Delta H}$, Nuclear Enthalpy Rise Hot Channel Factor, is defined as the ratio of the integral of linear power along a fuel rod to the average fuel rod power. Imposed limits pertain to the maximum $F^N_{\Delta H}$ in the core, that is the fuel rod with the highest integrated power. It should be noted that $F^N_{\Delta H}$ is based on an integral and is used as such in the DNB calculations. Local heat flux is obtained by using hot channel and adjacent channel explicit power shapes which take into account variations in horizontal (x-y) power shapes throughout the core. Thus, the horizontal power shape at the point of maximum heat flux is not necessarily directly related to $F^N_{\Delta H}$.

< See LCO 3.2.2 >

A.4

< See LCO 3.2.2 >

$F^N_{\Delta H}$ is sensitive to fuel loading patterns, bank insertion, and fuel burnup. $F^N_{\Delta H}$ typically increases with control bank insertion and typically decreases with fuel burnup.

$F^N_{\Delta H}$ is not directly measurable but is inferred from a power distribution map obtained with the movable incore detector system. Specifically, the results of the three dimensional power distribution map are analyzed by a computer to determine $F^N_{\Delta H}$. This factor is calculated at least monthly. However, during power operation, the global power distribution is monitored by TS 15.3.10.E.2, "Axial Flux Difference," and TS 15.3.10.E.3, "Quadrant Power Tilt," which address directly and continuously measured process variables.

It has been determined that, provided the following conditions are observed, the hot channel factor limits will be met:

1. Control rods in a single bank move together with no individual rod insertion differing by more than 24 steps from the bank demand position, when the bank demand position is between 30 steps and 215 steps. A misalignment of 36 steps is allowed when the bank position is less than or equal to 30 steps, or, when the bank position is greater than or equal to 215 steps, due to the small worth and consequential effects of an individual rod misalignment.

2. Control rod banks are sequenced with overlapping banks as described in Figure 15.3.10-1.

3. Control bank insertion limits are not violated.

4. Axial power distribution control procedures, which are given in terms of flux difference control and control bank insertion limits, are observed. Flux difference refers to the difference in signals between the top and bottom halves of two-section excore neutron detectors. The flux difference is a measure of the axial offset which is defined as the difference in normalized power between the top and bottom halves of the core.

The permitted relaxation of $F^N_{\Delta H}$ allows radial power shape changes with rod insertion to the insertion limits. It has been determined that provided the above four conditions are observed, these hot channel factor limits are met. In Specification 15.3.10.E.1.a, F_Q is arbitrarily limited for $p \leq 0.5$.

The upper bound envelope F_Q (defined in 15.3.10.E) times the normalized peaking factor axial dependence of Figure 15.3.10-3 for OFA and Upgraded OFA fuel and Figure 15.3.10.3a for 422V+ fuel (consistent with the Technical Specifications on power distribution control as given in Section 15.3.10) was used in the large and small break LOCA analyses. The envelope was determined based on allowable power density distributions at full power restricted to axial flux difference (ΔI) values consistent with those in Specification 15.3.10.E.2.

< See LCO 3.2.2 >

B

Amendm 193/198

A.4

The results of the analyses based on this upper bound envelope indicate a peak clad temperature of less than the 2200°F limit. When an F_Q measurement is taken, both experimental error and manufacturing tolerance must be taken into account. Five percent is the appropriate allowance for a full core map taken with the moveable incore detector flux mapping system and three percent is the appropriate allowance for manufacturing tolerance.

In the design limit of $F^N_{\Delta H}$, there is eight percent allowance for uncertainties which means that normal operation of the core is expected to result in a design $F^N_{\Delta H} \leq 1.70/1.08$ for OFA and Upgraded OFA and 1.77/1.08 for 422V+ fuel. The logic behind the larger uncertainty in this case is as follows:

(a) Normal perturbations in the radial power shape (i.e. rod misalignment) affect $F^N_{\Delta H}$, in most cases without necessarily affecting F_Q.

(b) While the operator has a direct influence on F_Q through movement of rods, and can limit it to the desired value, he has no direct control over $F^N_{\Delta H}$.

(c) An error in the predictions for radial power shape which may be detected during startup physics tests can be compensated for in F_Q by tighter axial control; but compensation for $F^N_{\Delta H}$ is less readily available.

Measurements of the hot channel factors are required as part of startup physics tests, at least each full power month operation, and whenever abnormal power distribution conditions require a reduction of core power to a level based upon measured hot channel factors. The incore map taken following initial loading provides confirmation of the basic nuclear design bases including proper fuel loading patterns. The periodic monthly incore mapping provides additional assurance that the nuclear design bases remain inviolate and identify operational anomalies which would, otherwise, affect these bases.

The measured hot channel factors are increased as follows:

(a) The measurement of total peaking factor, F_Q^{meas}, shall be increased by three percent to account for manufacturing tolerance and further increased by five percent to account for measurement error.

(b) The measurement of enthalpy rise hot channel factor, $F^N_{\Delta H}$ shall be increased by four percent to account for measurement error.

Axial Power Distribution

The limits on axial flux difference (AFD) assure that the axial power distribution is maintained such that the $F_Q(Z)$ upper bound envelope of F_Q^{LIMIT} times the normalized axial peaking factor [K(Z)] is not exceeded during either normal operation or in the event of xenon redistribution following power changes. This ensures that the power distributions assumed in the large and small break LOCA analyses will bound those that occur during plant operation.

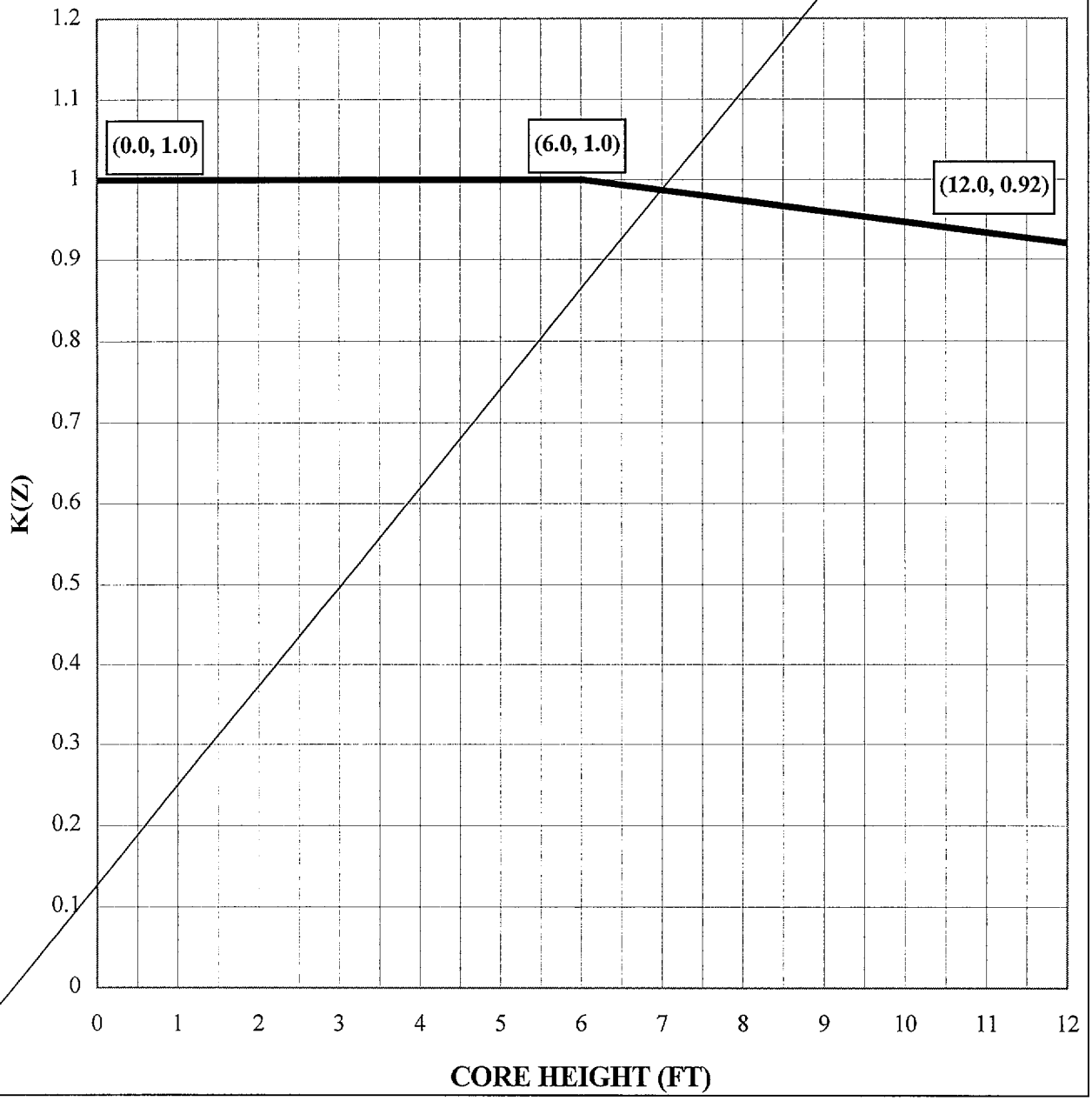
< See LCO 3.2.2 >

LA.1

FIGURE 15.3.10-3

POINT BEACH UNITS 1 AND 2
HOT CHANNEL FACTOR NORMALIZED OPERATING ENVELOPE FOR OFA AND UPGRADED OFA FUEL

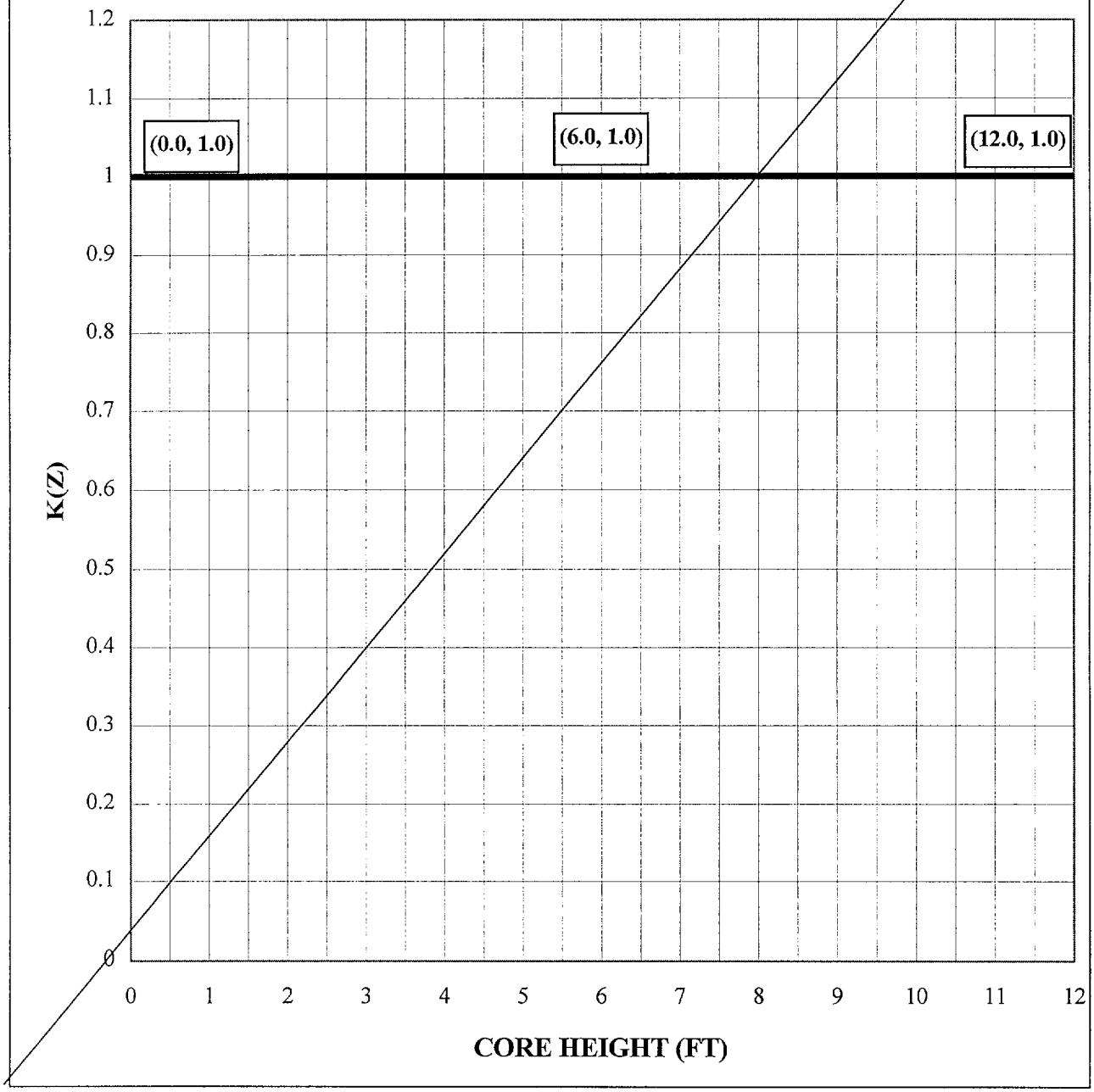
FIGURE 15.3.10-3A



B
Amendmen
193/198

LA.1

POINT BEACH UNITS 1 AND 2
HOT CHANNEL FACTOR NORMALIZED OPERATING ENVELOPE FOR 422V+ FUEL



B
Amendment 193/198

A.1

TABLE 15.4.1-2 (Continued)

30.	Pressurizer Heaters < See Section 3.4 >	Verify that 100 KW of	Quarterly
31.	CVCS Charging Pumps < See LCO 3.5.2 >	Verify operability	Quarterly
32.	Potential Dilution in Alarm < See Section 3.3 >	Verify operability of	Prior to placing plant in Progress cold shutdown.
~~33.~~	~~Core Power Distribution~~ < See LCOs: 3.3.2, 3.4.16, 3.5.4, 3.7.18, and 3.4.13 >	~~Perform power distribution maps using movable incore detector system to confirm hot channel factors.~~ LA.2	~~Monthly (20)~~ See Insert 3.2.1-2
34.	Shutdown Margin < See Section 3.1 >	Perform shutdown margin	Daily (21)

(1) Required only during periods of power operation. < See Section 3.4 >
(2) Q determination will be started when the gross activity analysis of a filtered sample indicates ³10mCi/cc and will be redetermined if the primary coolant gross radioactivity of a filtered sample increases by more than 10mCi/cc.
(3) Drop test shall be conducted at rated reactor coolant flow. Rods shall be dropped under both cold and hot condition, but cold drop tests need not be timed. < See LCO 3.1.5 >
(4) Drop tests will be conducted in the hot condition for rods on which maintenance was performed.
(5) As accessible without disassembly of rotor.
(6) Not required during periods of refueling shutdown.
(7) At least once per week during periods of refueling shutdown. < See LCO 3.4.16 >
(8) At least three times per week (with maximum time of 72 hours between refueling shutdown.
(9) Not required during periods of cold or refueling shutdown, but m < See LCOs 3.3.1, 3.6.3 > been performed during the previous surveillance period.
(10) Sample to be taken after a minimum of 2 EFPD and 20 days power operation since the reactor was last subcritical for 48 hours or longer.
(11) An approximately equal number of ~~valves shall be tested each~~ refueling outage such that all valves will be tested within a five year period. If any valve fails its tests, an additiona < See LCO 3.4.16 > < See LCOs 3.7.1, 3.4.10 > originally tested shall be tested. If any of the additional tested valves fail, all remaining valves shall be tested.
(12) The specified buses shall be determined energized in the required manner at least once per shift by verifying correct static transfer switch alignment and indicated voltag < See Section 3.8 >
(13) Not required if the block valve is shut to isolate a PORV that is inoperable for reasons other than excessive seat leakage. < See LCO 3.4.11 >
(14) Only applicable when the overpressure mitigation system is in service. < See LCO 3.4.12 >
(15) Required to be performed only if conditions will be established, as defined in Spec where the PORVs are used for low temperature overpressure protection. The test must be performed prior to establishing these conditions.

Associated Specification removed with Unit 1 Amendment 176 and Unit 2 Amendment 180

< See LCO 3.4.12 >

TABLE 15.4.1-2 (Continued)

A.1

(16) Test valve operation in accordance with the inservice test requirements of the ASME Boiler and Pressure Vessel Code, Section XI.

(17) Operability of charging pumps is verified by ensuring that the pumps develop the required flowrate, as specified by the In-Service Test Program.

(18) Not required to be performed if the reactor is subcritical.

(19) Required only when the BAST(s) are relied upon as a source of borated water.

(20) Perform during power operation at effective full power monthly intervals. Following a refueling shutdown, a power distribution map shall be performed prior to exceeding 90% of rated thermal power.

(21) Only applicable during low-power physics testing.

Replace with Insert 3.2.1-2

< See LCO 3.5.2 >

< See LCO 3.1.1 and 3.1.10>

< See LCO 3.1.5 >

INSERT 3.2.1-1:

CONDITION	REQUIRED ACTION		COMPLETION TIME
--------NOTE-------- Required Action B.4 shall be completed whenever this condition is entered. ---------- B. $F_Q^W(Z)$ not within limits.	B.1	Reduce AFD limits ≥ 1% for each 1% $F_Q^W(Z)$ exceeds limit.	4 hours
	AND		
	B.2	Reduce Power Range Neutron Flux-High trip setpoints ≥ 1% for each 1% that the maximum allowable power of the AFD limits is reduced.	72 hours
	AND		
	B.3	Reduce Overpower T trip setpoints ≥ 1% for each 1% that the maximum allowable power of the AFD limits is reduced.	72 hours
	AND		
	B.4	Perform SR 3.2.1.1 and SR 3.2.1.2.	Prior to increasing THERMAL POWER above the maximum allowable power of the AFD limits.

L.4

INSERT 3.2.1-2:

SURVEILLANCE REQUIREMENTS

----------NOTE----------
During power escalation at the beginning of each cycle, THERMAL POWER may be increased until an equilibrium power level has been achieved, at which a power distribution map is obtained.

M.3

SURVEILLANCE	FREQUENCY
SR 3.2.1.1 Verify $F_Q^C(Z)$ is within limit.	Once after each refueling prior to THERMAL POWER exceeding 75% RTP AND Once within 12 hours after achieving equilibrium conditions after exceeding, by ≥ 10% RTP, the THERMAL POWER at which $F_Q^C(Z)$ was last verified AND 31 EFPD thereafter

L.4

(continued)

A.1

CTS INSERTS

INSERT 3.2.1-2 (continued):

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.2.1.2	----------NOTE---------- If $F_Q^W(Z)$ measurements indicate maximum over z $\left[\frac{F_Q^C(Z)}{K(Z)} \right]$ has increased since the previous evaluation of $F_Q^C(Z)$: a. Increase $F_Q^W(Z)$ by the greater of a factor of 1.02 or by an appropriate factor specified in the COLR and reverify $F_Q^W(Z)$ is within limits; or b. Repeat SR 3.2.1.2 once per 7 EFPD until two successive flux maps indicate that the maximum over z $\left[\frac{F_Q^C(Z)}{K(Z)} \right]$ has not increased. ---------- Verify $F_Q^W(Z)$ is within limit.	Once after each refueling prior to THERMAL POWER exceeding 75% RTP AND Once within 12 hours after achieving equilibrium conditions after exceeding, by ≥ 10% RTP, the THERMAL POWER at which $F_Q^W(Z)$ was last verified AND 31 EFPD thereafter

L.4

M.3

A.1

Justification For Deviations - NUREG-1431 Section 3.02.04

24-May-00

JFD Number	JFD Text
01 Rev. A	As discussed in Description of Change L.04 of LCO 3.2.1, the surveillance methodology at Point Beach, by which FQ is determined uses Relaxed Axial Offset Control. Under this methodology, FQ is approximated using the limits FCQ(Z) and FWQ(Z), which are verified in Surveillance Requirements SR 3.2.1.1 and 3.2.1.2 in the proposed ITS. FCQ(Z) is the actual measured heat flux at equilibrium conditions, corrected for measurement and manufacturing tolerances, and FWQ(Z) is FCQ(Z) corrected for projected worst case heat flux redistributions. Verification of the steady state limit alone does not ensure that the transient limit is met. Therefore, adding a requirement to verify that the transient limit (FWQ(Z) - SR 3.2.1.2) is met as well as the steady state limit (FCQ(Z) - SR 3.2.1.1) is consistent with verification that the overall FQ limit is maintained as required by the CTS. This change implements approved TSTF 314.

ITS:	NUREG:
B 3.02.04	B 3.02.04
LCO 3.02.04 COND A RA A.3	LCO 3.02.04 COND A RA A.3
LCO 3.02.04 COND A RA A.6	LCO 3.02.04 COND A RA A.6

02 Rev. A	The NUREG Bases state that the acceptance criteria for energy deposition due to a rod ejection accident shall be no greater than 280 cal/gm in accordance with Regulatory Guide 1.77. The acceptance criteria for Point Beach is 225 cal/gm for unirradiated and 200 cal/gm for irradiated fuel. These acceptance limits have been previously approved by the NRC (SER dated May 8, 1989, Amendment No. 120/123 to the Point Beach Technical Specifications) and are referenced in FSAR Section 14.2.6. The proposed Bases has been modified accordingly to reflect Point Beach's licensing basis.

ITS:	NUREG:
B 3.02.04	B 3.02.04
LCO 3.02.04 COND A RA A.1	LCO 3.02.04 COND A RA A.1
LCO 3.02.04 COND A RA A.2	LCO 3.02.04 COND A RA A.2

03 Rev. B	NUREG Bases 3.2.4 contains information specific as to location of incore thimbles and to the types of incore flux maps that can be performed. The specific thimble location is for three and four loop plant designs and is, therefore, not applicable to the PBNP design. This information has been replaced with the applicable PBNP plant specific information.

ITS:	NUREG:
B 3.02.04	B 3.02.04

04 Rev. A	Reference 2 has been changed from "RG 1.77, Rev. [0], May 1974," to "FSAR, Section 14.2.6." This provides the appropriate reference for PBNP.

ITS:	NUREG:
B 3.02.04	B 3.02.04

Justification For Deviations - NUREG-1431 Section 3.02.04

24-May-00

JFD Number	JFD Text	
05 Rev. A	The GDC 26 reference has been modified to the appropriate reference for Point Beach.	
	ITS:	**NUREG:**
	B 3.02.04	B 3.02.04

BASES

SURVEILLANCE REQUIREMENTS (continued)

Approved TSTF 110

~~detection.~~ For

~~When the QPTR alarm is inoperable, the Frequency is increased to 12 hours. This Frequency is adequate to detect any relatively slow changes in QPTR, because for~~ those causes of QPT that occur quickly (e.g., a dropped rod), there typically are other indications of abnormality that prompt a verification of core power tilt.

Approved TSTF 109

SR 3.2.4.2

until 12 hours after

not

This Surveillance is modified by a Note, which states that it is required ~~only when~~ the input from one or more Power Range Neutron Flux channels are inoperable and the THERMAL POWER is ~~≥~~ 75% RTP.

>

Approved TSTF 241

With an NIS power range channel inoperable, tilt monitoring for a portion of the reactor core becomes degraded. Large tilts are likely detected with the remaining channels, but the capability for detection of small power tilts in some quadrants is decreased. Performing SR 3.2.4.2 at a Frequency of 12 hours provides an accurate alternative means for ensuring that any tilt remains within its limits.

For purposes of monitoring the QPTR when on e power range channel is inoperable, the moveable incore detectors are used to confirm that the normalized symmetric power distribution is consistent with the indicated QPTR and any previous data indicating a tilt. ~~The incore detector monitoring is performed with a full incore flux map or two sets of four thimble locations with quarter core symmetry. The two sets of four symmetric thimbles is a set of eight unique detector locations. These locations are C -8, E-5, E-11, H-3, H-13, L-5, L-11, and N-8 for three and four loop cores.~~

3

Insert B3.2.4-3

B

RAI 3.2.4-1

Approved TSTF 241

incore monitoring of

The symmetric thimble flux map can be used to generate symmetric thimble "tilt." This can be compared to a reference symmetric thimble tilt, from the most recent full core flux map, to generate an incore QPTR. Therefore, QPTR can be used to confirm that QPTR is within limits.

With one NIS channel inoperable, the indicated tilt may be changed from the value indicated with all four channels

LCO 3.2.4 BASES INSERTS

Insert B 3.2.4-1:

The maximum allowable power level initially determined by Required Action A.1 may be affected by subsequent determinations of QPTR. Increases in QPTR would require power reduction within 2 hours of QPTR determination, if necessary to comply with the decreased maximum allowable power level. Decreases in QPTR would allow increasing the maximum allowable power level up to the revised limit.

Insert B 3.2.4-2:

Note 2 states that if required Action A.5 is performed, then required Action A.6 shall be performed. Required Action A.5 normalizes the excore detectors to restore QPTR to within limits, which restores compliance with LCO 3.2.4. Thus, Note 2 prevents exiting the Actions prior to completing flux mapping to verify peaking factors, per Required Action A.6.

Insert B 3.2.4-3:

The incore detector monitoring is performed with a full incore flux map or three sets of four thimble locations with quarter core symmetry. The three sets of four symmetric thimbles is one of two sets of twelve unique detector locations. These locations are M7, G6, G4, G11, J10, E10, J3, H3, B6, L4, F12, F8, or M7, I7, D5, G11, G9, E10, J12, C3, C8, L9, H1, and F8.

B

RAI 3.2.4-1

B 3.2 POWER DISTRIBUTION LIMITS

B 3.2.4 QUADRANT POWER TILT RATIO (QPTR)

BASES

BACKGROUND

The QPTR limit ensures that the gross radial power distribution remains consistent with the design values used in the safety analyses. Precise radial power distribution measurements are made during startup testing, after refueling, and periodically during power operation.

The power density at any point in the core must be limited so that the fuel design criteria are maintained. Together, LCO 3.2.3, "AXIAL FLUX DIFFERENCE (AFD)," LCO 3.2.4, and LCO 3.1.6, "Control Rod Insertion Limits," provide limits on process variables that characterize and control the three dimensional power distribution of the reactor core. Control of these variables ensures that the core operates within the fuel design criteria and that the power distribution remains within the bounds used in the safety analyses.

APPLICABLE SAFETY ANALYSES

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

a. During a large break loss of coolant accident, the peak cladding temperature must not exceed 2200°F (Ref. 1);

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

c. During an ejected rod accident, the energy deposition to the fuel must not exceed 225 cal/gm for unirradiated and 200 cal/gm for irradiated fuel (Ref. 2); and

d. The control rods must be capable of shutting down the reactor with a minimum required SDM with the highest worth control rod stuck fully withdrawn (Ref. 3).

The LCO limits on the AFD, the QPTR, the Heat Flux Hot Channel Factor ($F_Q(Z)$), the Nuclear Enthalpy Rise Hot Channel Factor ($F^N_{\Delta H}$), and control bank insertion are established to preclude core power distributions that exceed the safety analyses limits.

The QPTR limits ensure that $F^N_{\Delta H}$ and $F_Q(Z)$ remain below their limiting values by preventing an undetected change in the gross radial power distribution.

BASES

APPLICABLE SAFETY ANALYSES (continued)	In MODE 1, the $F^N_{\Delta H}$ and $F_Q(Z)$ limits must be maintained to preclude core power distributions from exceeding design limits assumed in the safety analyses. The QPTR satisfies Criterion 2 of the NRC Policy Statement.
LCO	The QPTR limit of 1.02, at which corrective action is required, provides a margin of protection for both the DNB ratio and linear heat generation rate contributing to excessive power peaks resulting from X-Y plane power tilts. A limiting QPTR of 1.02 can be tolerated before the margin for uncertainty in $F_Q(Z)$ and $F^N_{\Delta H}$ is possibly challenged.
APPLICABILITY	The QPTR limit must be maintained in MODE 1 with THERMAL POWER > 50% RTP to prevent core power distributions from exceeding the design limits. Applicability in MODE 1 ≤ 50% RTP and in other MODES is not required because there is either insufficient stored energy in the fuel or insufficient energy being transferred to the reactor coolant to require the implementation of a QPTR limit on the distribution of core power. The QPTR limit in these conditions is, therefore, not important. Note that the $F^N_{\Delta H}$ and $F_Q(Z)$ LCOs still apply, but allow progressively higher peaking factors at 50% RTP or lower.
ACTIONS	A.1 With the QPTR exceeding its limit, a power level reduction of 3% RTP for each 1% by which the QPTR exceeds 1.00 is a conservative tradeoff of total core power with peak linear power. The Completion Time of 2 hours allows sufficient time to identify the cause and correct the tilt. Note that the power reduction itself may cause a change in the tilted condition. The maximum allowable power level initially determined by Required Action A.1 may be affected by subsequent determinations of QPTR. Increases in QPTR would require power reductions within 2 hours of QPTR determination, if necessary to comply with the decreased maximum allowable power level. Decreases in QPTR would allow increasing the maximum allowable power level and increasing power up to the revised limit.

BASES

ACTIONS (continued) A.2

After completion of Required Action A.1, the QPTR alarm may still be in its alarmed state. As such, any additional changes in the QPTR are detected by requiring a check of the QPTR once per 12 hours thereafter. A 12 hour Completion Time is sufficient because any additional change in QPTR would be relatively slow.

A.3

The peaking factors $F^N_{\Delta H}$ and $F_Q(Z)$, as approximated by $F^C_Q(Z)$ and $F^W_Q(Z)$, are of primary importance in ensuring that the power distribution remains consistent with the initial conditions used in the safety analyses. Performing SRs on $F^N_{\Delta H}$ and $F_Q(Z)$ within the Completion Time of 24 hours after achieving equilibrium conditions from a thermal power reduction per Required Action A.1 ensures that these primary indicators of power distribution are within their respective limits. Equilibrium conditions are achieved when the core is sufficiently stable at intended operating conditions to support flux mapping. A Completion Time of 24 hours after achieving equilibrium conditions from a thermal power reduction per Required Action A.1 takes into consideration the rate at which peaking factors are likely to change, and the time required to stabilize the plant and perform a flux map. If these peaking factors are not within their limits, the Required Actions of these Surveillances provide an appropriate response for the abnormal condition. If the QPTR remains above its specified limit, the peaking factor surveillances are required each 7 days thereafter to evaluate $F^N_{\Delta H}$ and $F_Q(Z)$ with changes in power distribution. Relatively small changes are expected due to either burnup and xenon redistribution or correction of the cause for exceeding the QPTR limit.

A.4

Although $F^N_{\Delta H}$ and $F_Q(Z)$ are of primary importance as initial conditions in the safety analyses, other changes in the power distribution may occur as the QPTR limit is exceeded and may have an impact on the validity of the safety analysis. A change in the power distribution can affect such reactor parameters as bank worths and peaking factors for rod malfunction accidents. When the QPTR exceeds its limit, it does not necessarily mean a safety concern exists. It does mean that there is an indication of a change in the gross radial power distribution that requires an investigation and evaluation that is accomplished by examining the incore power distribution. Specifically, the core peaking factors and the quadrant tilt must be evaluated because they are the factors that best characterize the core power distribution.

BASES

ACTIONS (continued) This re-evaluation is required to ensure that, before increasing THERMAL POWER to above the limit of Required Action A.1, the reactor core conditions are consistent with the assumptions in the safety analyses.

A.5

If the QPTR has exceeded the 1.02 limit and a re-evaluation of the safety analysis is completed and shows that safety requirements are met, the excore detectors are normalized to restore QPTR to within limits prior to increasing THERMAL POWER to above the limit of Required Action A.1. Normalization is accomplished in such a manner that the indicated QPTR following normalization is near 1.00. This is done to detect any subsequent significant changes in QPTR.

Required Action A.5 is modified by two Notes. Note 1 states that the QPT is not restored to within limits until after the re-evaluation of the safety analysis has determined that core conditions at RTP are within the safety analysis assumptions (i.e., Required Action A.4). Note 2 states that if required Action A.5 is performed, then required Action A.6 shall be performed. Required Action A.5 normalizes the excore detectors to restore QPTR to within limits, which restores compliance with LCO 3.2.4. Thus, Note 2 prevents exiting the Actions prior to completing flux mapping to verify peaking factors, per Required Action A.6. These Notes are intended to prevent any ambiguity about the required sequence of actions.

A.6

Once the flux tilt is restored to within limits (i.e., Required Action A.5 is performed), it is acceptable to return to full power operation. However, as an added check that the core power distribution is consistent with the safety analysis assumptions, Required Action A.6 requires verification that $F_Q(Z)$, as approximated by $F_Q^C(Z)$ and $F_Q^W(Z)$, and $F_{\Delta H}^N$ are within their specified limits within 24 hours of achieving equilibrium conditions at RTP. As an added precaution, if the core power does not reach equilibrium conditions at RTP within 24 hours, but is increased slowly, then the peaking factor surveillances must be performed within 48 hours after increasing thermal power above the limit of Required Action A.1. These Completion Times are intended to allow adequate time to increase THERMAL POWER to above the limit of Required Action A.1, while not permitting the core to remain with unconfirmed power distributions for extended periods of time.

BASES

ACTIONS (continued)

Required Action A.6 is modified by a Note that states that the peaking factor surveillances may only be done after the excore detectors have been normalized to restore QPTR to within limits (i.e., Required Action A.5). The intent of this Note is to have the peaking factor surveillances performed at operating power levels, which can only be accomplished after the excore detectors are normalized to restore QPTR to within limits and the core returned to power.

B.1

If Required Actions A.1 through A.6 are not completed within their associated Completion Times, the unit must be brought to a MODE or condition in which the requirements do not apply. To achieve this status, THERMAL POWER must be reduced to < 50% RTP within 4 hours. The allowed Completion Time of 4 hours is reasonable, based on operating experience regarding the amount of time required to reach the reduced power level without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.2.4.1

SR 3.2.4.1 is modified by two Notes. Note 1 allows QPTR to be calculated with three power range channels if THERMAL POWER is ≤ 75% RTP and the input from one Power Range Neutron Flux channel is inoperable. Note 2 allows performance of SR 3.2.4.2 in lieu of SR 3.2.4.1.

This Surveillance verifies that the QPTR, as indicated by the Nuclear Instrumentation System (NIS) excore channels, is within its limits. The Frequency of 7 days takes into account other information and alarms available to the operator in the control room.

For those causes of QPT that occur quickly (e.g., a dropped rod), there typically are other indications of abnormality that prompt a verification of core power tilt.

SR 3.2.4.2

This Surveillance is modified by a Note, which states that it is not required until 12 hours after the input from one or more Power Range Neutron Flux channels are inoperable and the THERMAL POWER is > 75% RTP.

BASES

SURVEILLANCE REQUIREMENTS (continued)

With an NIS power range channel inoperable, tilt monitoring for a portion of the reactor core becomes degraded. Large tilts are likely detected with the remaining channels, but the capability for detection of small power tilts in some quadrants is decreased. Performing SR 3.2.4.2 at a Frequency of 12 hours provides an accurate alternative means for ensuring that any tilt remains within its limits.

For purposes of monitoring the QPTR when one power range channel is inoperable, the moveable incore detectors are used to confirm that the normalized symmetric power distribution is consistent with the indicated QPTR and any previous data indicating a tilt. The incore detector monitoring is performed with a full incore flux map or three sets of four thimble locations with quarter core symmetry. The three sets of four symmetric thimbles is one of two sets of twelve unique detector locations. These locations are M7, G6, G4, G11, J10, E10, J3, H3, B6, L4, F12, F8, or M7, I7, D5, G11, G9, E10, J12, C3, C8, L9, H1, and F8. B RAI 3.2.4-1

The symmetric thimble flux map can be used to generate symmetric thimble "tilt." This can be compared to a reference symmetric thimble tilt, from the most recent full core flux map, to generate an incore QPTR. Therefore, incore monitoring of QPTR can be used to confirm that QPTR is within limits.

With one NIS channel inoperable, the indicated tilt may be changed from the value indicated with all four channels OPERABLE. To confirm that no change in tilt has actually occurred, which might cause the QPTR limit to be exceeded, the incore result may be compared against previous flux maps either using the symmetric thimbles as described above or a complete flux map. Nominally, quadrant tilt from the Surveillance should be within 2% of the tilt shown by the most recent flux map data.

REFERENCES

1. 10 CFR 50.46.
2. FSAR, Section 14.2.6.
3. FSAR, Chapter 3.

Justification For Deviations - NUREG-1431 Section 3.05.01

08-Jun-00

JFD Number	JFD Text
07 Rev. A	The NUREG Bases for LCO 3.5.1 contains several statements regarding the automatic open feature for the Safety Injection Accumulators, which is interlocked with Permissive P-11 at approximately 1830 psig in the Pressurizer. This feature is not in cluded in the design of Point Beach. Discussion regarding this feature, its association with IEEE 279, and any reference to these items (both in the Bases discussions and the References section) have been deleted. Deletion of the reference to IEEE 279 has resulted in a renumbering of several references within the Bases. These changes are administrative. This change is also consistent with TSTF 316, Revision 0. **ITS:** B 3.05.01 — **NUREG:** B 3.05.01
08 Rev. B	SR 3.5.1.2 and the Bases have been modified regarding the units for Accumulator volume. NUREG 1431 specifies units for Accumulator level in terms of gallons or percentage of indicated volume. The analysis and CTS for Point Beach are in units of cubic feet; therefore, the proposed ITS values have been indicated in cubic feet. The Bases has also been modified to more explicitly reflect Point Beach's basis for the accumulator upper concentration limit, because there is no hot leg recirculation mode which is usually the basis for the upper limit. **ITS:** B 3.05.01 — **NUREG:** B 3.05.01 SR 3.05.01.02 — SR 3.05.01.02
09 Rev. A	The Bases for NUREG 1431 was written to reflect a Westinghouse four loop plant which has four Accumulators. Point Beach is a two loop plant with two Accumulators, therefore, the Bases for LCO 3.5.1 has been rephrased in various locations to reflect a total of two Accumulators consistent with Point Beach's design. **ITS:** B 3.05.01 — **NUREG:** B 3.05.01

JFD Number	JFD Text
16 Rev. B	NUREG 1431 SR 3.5.1.4 requires verification of accumulator boron concentration within 6 hours of an increase in accumulator level from sources other than the refueling water storage tank. Sampling of the affected accumulator after an increase in volume is necessary to determine whether the level increase has resulted in a reduction in accumulator boron concentration to below the required limit. It is not necessary to verify boron concentration if the added water inventory is from the refueling water storage tank, because the water contained in the refueling water storage tank is within the accumulator boron concentration requirements. The proposed Point Beach ITS will allow 24 hours to complete the above mentioned sampling. The STS frequency of 6 hours for obtaining a accumulator boron concentration sample was based on a Westinghouse 4 loop reactor design that has line taps outside containment to obtain accumulator boron samples. However, PBNP does not have this design attribute. A containment entry must be made at PBNP to obtain the necessary accumulator boron samples. Therefore, the additional time is necessary to allow coordination between Radiation Protection, Operations, and Chemistry for the containment entry evolution to obtain samples and perform boron titration's. If an unplanned volume addition was required during a back shift, additional personnel could conceivably be required to be called in to support this evolution, thereby necessitating additional time. Twenty-four hours is a reasonable period of time to perform a containment entry and obtain sample results under these conditions. This time frame is also acceptable based on rationale presented for restoring accumulators boron concentration to within limits (72 hours), as contain in Required Action A.1 of LCO 3.5.1 of NUREG 1431. It should be noted that being in this surveillance requirement would be a rare occurrence at PBNP, since accumulator make up is almost always performed from the RWST and it is usually less than 5% of indicated level (surveillance would not be applicable within these conditions). **ITS:** B 3.05.01 — SR 3.05.01.04 **NUREG:** B 3.05.01 — SR 3.05.01.04
17 Rev. A	The Frequency for SR 3.5.1.4 contains an exception for sampling Accumulator boron concentration, whenever accumulator level has been increased using make up from the RWST. This exception is allowable when the RWST and Accumulator boron concentration limits are the same, however, the Accumulator and RWST boron concentration limits for Point Beach differ (Accumulator - 2600-3100 ppm, RWST – 2700-3200 ppm). Therefore, this provision has been modified to allow the exception only when RWST boron concentration has been verified by its most recent sample to be within the accumulator boron concentration limits. **ITS:** B 3.05.01 — SR 3.05.01.04 **NUREG:** B 3.05.01 — SR 3.05.01.04

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.5.1.1	Verify each accumulator isolation valve is fully open.	12 hours
SR 3.5.1.2	Verify borated ~~water volume in each accumulator is ≥ [7853 gallons ( )% and ≤ 8171 gallons ( )%].~~ ≥ 1100 ft^3 and ≤ 1136 ft^3.	12 hours 8 B Errata #60
SR 3.5.1.3	Verify nitrogen cover pressure in each accumulator is ~~≥ [385] psig and ≤ [481] psig.~~ ≥ 700 psig and ≤ 800 psig.	12 hours 4
SR 3.5.1.4	Verify boron concentration in each accumulator is ≥ ~~[1900]~~ 2600 ppm and ≤ ~~[2100]~~ 3100 ppm. 4	31 days AND -----NOTE------ Only required to be performed for affected accumulators ---------- Once within ~~6~~ 24 hours after each solution volume increase of ≥ ~~[[ ] gallons, ( )%~~ 5% of indicated level that is not the result of addition from the refueling water storage tank 16 5 4 (continued)

17 with boron concentration ≥ 2600ppm and ≤ 3100ppm

3.5 EMERGENCY CORE COOLING SYSTEMS (ECCS)

3.5.1 Accumulators

LCO 3.5.1 Two Safety Injection Accumulators shall be OPERABLE.

APPLICABILITY: MODES 1 and 2,
MODE 3 with RCS pressure > 1000 psig.

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. One accumulator inoperable due to boron concentration not within limits.	A.1	Restore boron concentration to within limits.	72 hours
B. Power not removed from one or both accumulator isolation valve(s) for maintenance or testing.	B.1	Remove power from accumulator isolation valve(s).	4 hours
C. One accumulator inoperable for reasons other than Conditions A or B.	C.1	Restore accumulator to OPERABLE status.	1 hour
D. Required Action and associated Completion Time of Condition A, B, or C not met.	D.1	Be in MODE 3.	6 hours
	AND		
	D.2	Reduce RCS pressure to ≤ 1000 psig.	12 hours
E. Two accumulators inoperable for reasons other than Condition B.	E.1	Enter LCO 3.0.3.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.5.1.1	Verify each accumulator isolation valve is fully open.	12 hours
SR 3.5.1.2	Verify borated water volume in each accumulator is ≥ 1100 ft^3 and ≤ 1136 ft^3.	12 hours B Errata #60
SR 3.5.1.3	Verify nitrogen cover pressure in each accumulator is ≥ 700 psig and ≤ 800 psig.	12 hours
SR 3.5.1.4	Verify boron concentration in each accumulator is ≥ 2600 ppm and ≤ 3100 ppm.	31 days AND ------NOTE------ Only required to be performed for affected accumulators ---------- Once within 24 hours after each solution volume increase of ≥ 5% of indicated level that is not the result of addition from the refueling water storage tank with boron concentration ≥ 2600 ppm and ≤ 3100 ppm
SR 3.5.1.5	Verify power is removed from each accumulator isolation valve operator when RCS pressure is > 1000 psig.	31 days

Description of Changes - NUREG-1431 Section 3.05.02

24-May-00

DOC Number	DOC Text
A.01 Rev. A	In the conversion of Point Beach current Technical Specifications (CTS) to the proposed plant specific Improved Technical Specifications (ITS), certain wording preferences or conventions are adopted which do not result in technical changes (either actual or interpretational). Editorial changes, reformatting, and revised numbering are adopted to make the ITS consistent with the Standard Technical Specifications, Westinghouse Plants, NUREG-1431, Revision 1 (i.e., Improved Standard Technical Specifications (ISTS)).

CTS:	ITS:
15.03.03.A.02	LCO 3.05.02 COND B LCO 3.05.02 COND B RA B.1
15.03.03.A.03	LCO 3.05.02 COND B LCO 3.05.02 COND B RA B.1
15.04.05	LCO 3.05.02
15.04.05.I.A.02	SR 3.05.02.03 SR 3.05.02.04

A.02
Rev. A

The CTS states that the ECCS systems and components listed in Specification 15.3.3.A.1.c, f, and g (SI pumps, SI discharge isolation valves, associated valves, interlocks, and piping) are required to be operable prior to the reactor being made critical. However, the Actions contained in CTS 15.3.3.A.2 will place the unit in cold shutdown if the SI pump/subsystem is not returned to an operable status within its allowed out of service time, implying an Applicability of Modes 1, 2, 3, and 4 (ITS Modes). CTS 15.3.15.B modifies the requirement of CTS 15.3.3.A.1.c (required number of SI pumps) by stating that one of the two high head Safety Injection pumps are to be rendered inoperable whenever LTOP is required (less than 355 degrees).

Proposed LCO 3.5.2 requires two trains of ECCS (SI and an RHR subsystem) to be operable in Modes 1, 2, and 3 (greater than or equal to 350 degrees) and allows one of the two SI pumps to be inoperable under specific constraints (up to four hours after entry into Mode 3 or until all RCS cold leg temperatures exceed 375 degrees, whichever comes first). As such, in Modes 1, 2, and 3 when RCS temperature is greater than or equal to 355 degrees, the number of required SI pumps/subsystems is the same in the proposed ITS as the CTS. The four hour and 375 degree SI pump restoration/disablement constraints are discussed in Description of Changes M.2 and L.1 of this section. The Mode 4 ECCS aspects of this LCO are discussed in LCO section 3.5.3.

CTS:	ITS:
15.03.03.A.01	LCO 3.05.02

Description of Changes - NUREG-1431 Section 3.05.02

24-May-00

DOC Number	DOC Text
A.03 Rev. A	CTS 15.3.3.A.1.c and d require two safety injection (SI) and two residual heat removal (RHR) pumps to be operable, while CTS 15.4.2.B.3 and CTS 15.4.5.II.A.1 requires these pumps to be tested in accordance with ASME Section XI. ITS LCO 3.5.2 requires two ECCS trains to be operable, and ITS SR 3.5.2.2 requires each ECCS pump to be tested in accordance with the Inservice Testing Program. Accordingly, while presented in a different fashion than the CTS, the ITS continue to require two SI and RHR pumps to be operable, and require both SI and RHR pumps to be testing in accordance with the ASME code.

CTS:	ITS:
15.03.03.A.01.C	SR 3.05.02.02
15.03.03.A.01.D	SR 3.05.02.02
15.04.02.B.03	SR 3.05.02.02
15.04.05.II.A.01	SR 3.05.02.02

A.04 Rev. A	CTS 15.3.3.A.1.f requires the isolation valves in the discharge header of the SI system to be in the open position. CTS 15.4.5.II.B.2 requires each manual, power operated and automatic valve necessary to ensure system operability, that is not locked, sealed, or otherwise secured in position, to be verified to be in its correct position at least once every 31 days. ITS SR 3.5.2.1 requires each manual, power operated and automatic valve in the ECCS flowpath, that is not locked, sealed, or otherwise secured in position, to be verified to be in its correct position at least once every 31 days. All surveillance requirements associated with an LCO are required to be met in order to fulfill the LCO. Accordingly, there is no change in requirement, making this change administrative.

CTS:	ITS:
15.03.03.A.01.F	SR 3.05.02.01
15.04.05.II.B.02	SR 3.05.02.01

A.05 Rev. A	CTS 15.3.3.A.1.g, requires the interlocks associated with the ECCS pumps and valves which are required to function during accident conditions to be operable. CTS 15.4.5.I.A.1 requires the performance of a system test during reactor shutdowns for major refueling outages, which ultimately verifies that all components received their actuation signal and actuate to their correct positions. ITS SR 3.5.2.3 requires each ECCS automatic valve in the flowpath that is not locked, sealed, or otherwise secured in position, to be tested to ensure it actuates to its correct position on an actual or simulated actuation signal. Similarly, SR 3.5.2.4 requires each ECCS pump to be tested to ensure it actuates on an actual or simulated actuation signal. Accordingly, both the CTS and ITS require these attributes, making this change administrative, consistent with the format and presentation of NUREG 1431.

CTS:	ITS:
15.03.03.A.01.G	SR 3.05.02.03 SR 3.05.02.04

DOC Number	DOC Text
A.06 Rev. A	CTS 15.3.3.A.2.b, 15.3.3.A.2.c, 15.3.3.A.3.a, and 15.3.3.A.3.c allow 72 hours to restore an inoperable SI pump, RHR pump, or valve affecting an SI or RHR subsystem to operable status providing the redundant pumps and valves are operable. ITS LCO 3.5.2 Condition A allows 72 hours for an ECCS Train to be returned to operable status, prior to requiring a unit shutdown. The CTS verbiage requiring the redundant pumps and valves have been omitted as this requirement is inherent to Condition A which only provides Actions for a single inoperable ECCS Train. Accordingly, the CTS and ITS requirement are the same, making this change administrative.

CTS:	ITS:
15.03.03.A.02.B	DELETED
15.03.03.A.02.C	DELETED
15.03.03.A.03.A	DELETED
15.03.03.A.03.C	DELETED

DOC Number	DOC Text
A.07 Rev. A	CTS 15.3.3.A.2.b, 15.3.3.A.2.c, 15.3.3.A.3.a, 15.3.3.A.3.b, and 15.3.3.A.3.c, define components within the safety injections and residual heat removal systems which are allowed to be inoperable for up to 72 hours before requiring a unit shutdown. Similarly, ITS LCO 3.5.2 Condition A and Required Action A.1 allows one train of ECCS to be inoperable for up to 72 hours before requiring a unit shutdown. The CTS details which define the components that may be inoperable (pumps, valves, and heat exchangers) are attributes of an ECCS Train which are addressed by the Surveillance Requirements associated with LCO 3.5.2, or by the definition of operability. The inoperability of an ECCS Train's component which results in a train of ECCS becoming inoperable, requires entry into Condition A. Therefore, these details are not necessary to define the entry conditions for the ITS Condition and can be removed from the Technical Specifications without changing the meaning, usage, or intent of this Condition.

CTS:	ITS:
15.03.03.A.02.B	LCO 3.05.02 COND A RA A.1
15.03.03.A.02.C	LCO 3.05.02 COND A RA A.1
15.03.03.A.03.A	LCO 3.05.02 COND A LCO 3.05.02 COND A RA A.1
15.03.03.A.03.B	LCO 3.05.02 COND A LCO 3.05.02 COND A RA A.1
15.03.03.A.03.C	LCO 3.05.02 COND A LCO 3.05.02 COND A RA A.1

Description of Changes - NUREG-1431 Section 3.05.02

24-May-00

DOC Number	DOC Text
A.08 Rev. A	The Bases of the current Technical Specifications for this LCO have been completely replaced by the revised Bases reflecting the format and applicable content of the Improved Technical Specifications for Point Beach. The proposed Bases are based on NUREG 1431 Rev. 1. The proposed Bases for this LCO are consistent and supportive of the proposed LCO, and accordingly is administrative.

CTS:	ITS:
BASES	B 3.05.02

DOC Number	DOC Text
A.09 Rev. A	CTS 15.4.5.II.A.1 requires the SI and RH pumps to be tested in accordance with the Inservice Testing Program, while CTS 15.4.5.II.A.2 specifies that the pumps must start and reach their required developed head on the full flow test line. ITS SR 3.5.2.2 requires the SI and RH pumps to be tested in accordance with the Inservice Testing Program, and that the pumps develop greater than or equal to their required head at the test flow point. The CTS requirement only states to use the full flow test line not a specific test configuration/point. Accordingly, the CTS and the ITS item are the same, making this change administrative.

CTS:	ITS:
15.04.05.II.A.02	SR 3.05.02.02

DOC Number	DOC Text
A.10 Rev. A	The CTS provides an introductory statement (Applicability) which simply states which systems/components are addressed within a given section. This same information while worded differently is contained within the title of each ITS LCO. Accordingly, this change is a change in format with no change in technical requirement.

CTS:	ITS:
15.03.03 APPL	LCO 3.05.02
15.04.05 APPL	LCO 3.05.02

DOC Number	DOC Text
A.11 Rev. A	The CTS provides an introductory statement (Objective) at the beginning of this Section of the Technical Specifications which provide a brief summary of the purpose for this Section. This information is contained in the Bases Section of the ITS. This information does not establish any regulatory requirements for the systems and components addressed within this Section. Accordingly, deletion of this information does not alter any requirement set forth in the Technical Specifications. This change is administrative and consistent with the format and presentation for the ITS as provided in NUREG 1431.

CTS:	ITS:
15.03.03 OBJ	B 3.05.02
15.04.05 OBJ	B 3.05.02

DOC Number	DOC Text
L.01 Rev. A	The CTS states that the ECCS systems and components listed in Specification 15.3.3.A.1.c, f, and g (SI pumps, SI discharge isolation valves, associated valves, interlocks, and piping) are required to be operable prior to the reactor being made critical. However, the Actions contained in CTS 15.3.3.A.2 will place the unit in cold shutdown if the SI pump/subsystem is not returned to an operable status within its allowed out of service time, implying an Applicability of Modes 1, 2, 3, and 4 (ITS Modes). CTS 15.3.15.B modifies the requirement of CTS 15.3.3.A.1.c (required number of SI pumps) by stating that one of the two high head Safety Injection pumps to be rendered inoperable whenever LTOP is required (less than 355 degrees). The ECCS flow paths consist of piping, valves, heat exchangers, and pumps necessary to provide water from the RWST into the RCS during the injection phase and from the containment sump into the RCS during the recirculation phase following the accidents described in this LCO. The major components of each subsystem are the RHR pumps, heat exchangers, and the SI pumps. Each of the two subsystems consists of two 100% capacity trains that are interconnected and redundant such that either train is capable of supplying 100% of the flow required to mitigate the accident consequences. Proposed LCO 3.5.2 requires two trains of ECCS (SI and an RHR subsystem) to be operable in Modes 1, 2, and 3 (greater than or equal to 350 degrees); allows both SI flow pump flow paths to be isolated in order to perform pressure isolation valve testing (up to two hours); and, allows one of the two SI pumps to be incapable of injection under specific constrains (up to four hours after entry into Mode 3 or until all RCS cold leg temperatures exceed 375 degrees, whichever comes first). In Modes 1, 2, and 3 when RCS temperature is greater than or equal to 355 degrees, the number of SI pumps/subsystems is the same in the proposed ITS as the CTS as stated in Description of Change A.2 of this section. LCO 3.5.2 allows both SI flow paths to be isolated for a limited time (2 hours) to perform pressure isolation valve testing. This short period of time provides for only a short additional time (1 hour) before a plant shutdown must be commenced because of two trains of ECCS being inoperable as would be required by LCO 3.0.3 (CTS 15.3.0.B). This limited period allows testing to be performed without having to place the plant unnecessarily into the cold shutdown condition. During the testing the valves are capable of being opened from the control room in a short period of time if the ECCS system is called upon to perform its function. The additional 1 hour of inoperability allowed by this change makes this a less restrictive change. This requirement also supports an orderly transition from hot shutdown to hot stanby (ITS). In Mode 3 with RCS temperature between 355 and 375 degrees, the CTS requires both SI pumps to be operable, therefore, the latitude established in the ITS to heatup to 375 degrees prior to restoring the second SI pump is a relaxation. With the LTOP arming temperature in excess of the Mode 3 boundary temperature, an exception is necessary to allow transition into Mode 3 without reliance upon an Action. When increasing RCS temperature, this exception is necessary as LCO 3.0.4 would preclude entry into Mode 3 due to the LCO Actions not allowing indefinite operation in Mode 1, 2, or 3. When decreasing RCS temperature the provision of Note 2 will allow an SI pump to be rendered inoperable pursuant to LCO 3.4.12, when the RCS

DOC Number	DOC Text
	temperature in at least one cold leg is less than or equal to 375° F, for a period not to exceed 4 hours. This will allow transition into MODE 4 without requiring entry into the Conditions and Required Actions of LCO 3.5.2. This temperature band (20 degrees, between 355 and 375 degrees) was chosen to be controllable without the use of extraordinary measures, providing an allowance to perform an orderly and controlled restoration/disablement of the second SI pump based on LTOP concerns. The 375 degree limit is sufficiently above the LTOP enable temperature to allow for both enabling and disabling injection capability of the SI pump in support of heat up and cooldown operations minimizing the potential for inappropriate actions that could result as a result of a exceedingly tight temperature control band.

CTS:	ITS:
15.03.03.A.01	LCO 3.05.02
15.03.03.A.01.C	LCO 3.05.02
NEW	LCO 3.05.02 APPL NOTE 1
	LCO 3.05.02 APPL NOTE 2

L.02 Rev. B	CTS 15.4.5.II.A.2 specifies the conditions and manner in which ECCS pump testing must be conducted (run for at least 15 minutes using the full flow test line). This requirement to run the pumps for at least 15 minutes is an arbitrary requirement with no fundamental safety basis. Therefore, these details are being removed. The proposed ITS specifies the safety objective that must be fulfilled by the surveillance tests, while leaving the details associated with testing methods and acceptance verifications to licensee control. These type of details are better suited for procedural control and are not required to be in the ITS to provide adequate protection to the public health and safety. Changes to plant procedures and other plant controlled documents are subject to controls imposed by plant administrative procedures, which endorse applicable regulations and standards.

CTS:	ITS:
15.04.05.II.A.02	DELETED

DOC Number	DOC Text
L.03 Rev. B	CTS 15.4.5.I.A.1.a specifies the conditions and manner in which ECCS testing must be conducted (RCS pressure less than or equal to 350 psig and RCS temperature less than or equal to 350 degrees, pump breakers may be placed into the "test" position), while CTS 15.4.5.I.A.2 specifies the manner in which equipment operation is to be verified (control board indication and visual indications). These details have been deleted from the Technical Specification, and moved to licensee control as these details are not necessary to describe the actual regulatory requirement (i.e. verification that equipment actuates to its correct position). These details are not required to be in the ITS to provide adequate protection of public health and safety, as the regulatory requirement (verification of equipment actuation) is being maintained in the Technical Specifications. Changes to plant procedures and other plant controlled documents are subject to controls imposed by plant administrative procedures, which endorse applicable regulations and standards.

CTS:	ITS:
15.04.05.I.A.01	DELETED
15.04.05.I.A.01.A	DELETED
15.04.05.I.A.02	DELETED

LA.01 Rev. A	CTS 15.3.3.A.1.e and CTS 15.3.3.A.1.g list the RHR Heat Exchangers, valves, interlocks, and piping associated with the SI and RHR Systems required to be operable to fulfill the ECCS LCO requirement. These are all attributes associated with system design and configuration, which are adequately captured through application of the definition of operability, and accordingly are still encompassed within the LCOs for ECCS. As such, these details are not required to be in the ITS to provide adequate protection of public health and safety. These attributes are discussed within the Bases for the proposed Point Beach ITS, but have been deleted from the Technical Specifications, changes to these details will be controlled in accordance with the provisions of the Bases Control Program described in Chapter 5 of the Technical Specifications.

CTS:	ITS:
15.03.03.A.01.E	DELETED
	DELETED

LA.02 Rev. B	Not Used.

CTS:	ITS:
N/A	N/A

LA.03 Rev. B	Not Used.

CTS:	ITS:
N/A	N/A

Description of Changes - NUREG-1431 Section 3.05.02

24-May-00

DOC Number	DOC Text
M.01 Rev. A	The CTS contains a provision exempting the requirement to maintain the Safety Injection and Residual Heat Removal Systems operable during low power physics testing. This provision has been deleted in the proposed Technical Specifications. Low power physics testing in the Improved Technical Specifications is a subset of Mode 2. While Mode 2 is typically a non limiting Mode, the operability requirements of the Safety Injection and Residual Heat Removal Systems are independent of physics testing, accordingly this provision has been deleted. This change represent a more restrictive changes as it involves the deletion of a flexibility that currently exists.

CTS:	ITS:
15.03.03.A.01	DELETED

DOC Number	DOC Text
M.02 Rev. A	The CTS states that the ECCS systems and components listed in Specification 15.3.3.A.1.c, f, and g (SI pumps, SI discharge isolation valves, associated valves, interlocks, and piping) are required to be operable prior to the reactor being made critical. However the Actions contained in CTS 15.3.3.A.2 will place the unit in cold shutdown if the SI pump/subsystem is not returned to an operable status within its allowed out of service time, implying an Applicability of Modes 1, 2, 3, and 4 (ITS Modes). CTS 15.3.15.B modifies the requirement of CTS 15.3.3.A.1.c (required number of SI pumps) by stating that one of the two high head Safety Injection pumps are to be rendered inoperable whenever LTOP is required (less than 355 degrees). The ECCS flow paths consist of piping, valves, heat exchangers, and pumps necessary to provide water from the RWST into the RCS during the injection phase and from the containment sump into the RCS during the recirculation phase following the accidents described in this LCO. The major components of each subsystem are the RHR pumps, heat exchangers, and the SI pumps. Each of the two subsystems consists of two 100% capacity trains that are interconnected and redundant such that either train is capable of supplying 100% of the flow required to mitigate the accident consequences. Proposed LCO 3.5.2 requires two trains of ECCS (SI and an RHR subsystem) to be operable in Modes 1, 2, and 3 (greater than or equal to 350 degrees) and allows one of the two SI pumps to be inoperable under specific constraints (up to four hours after entry into Mode 3 or until all RCS cold leg temperatures exceed 375 degrees, whichever comes first). As such, in Modes 1, 2, and 3 when RCS temperature is greater than or equal to 355 degrees, the number of SI pumps/subsystems is the same in the proposed ITS as the CTS as stated in Description of Change A.2 of this section. In Mode 3 with RCS temperature less than 355 degrees, the CTS allows unlimited operation, the ITS, however, becomes more restrictive by imposing a four hour time constraint for restoring the second SI pump to operable status after entry into Mode 3. This limitation is a more restrictive change imposed to assure timely restoration of the second SI pump, consistent with the objective of maintaining single failure protection in Modes 1, 2, and 3 for this system. The 375 degree SI pump restoration/disablement constraint is discussed in Description of Changes L.1 of this section. The Mode 4 ECCS aspects of this LCO are discussed in LCO section 3.5.3.

CTS:	ITS:
15.03.03.A.01.C	LCO 3.05.02
15.03.15.B.01	LCO 3.05.02 APPL NOTE 2

DOC Number	DOC Text
M.03 Rev. A	CTS 15.3.3.A.2, CTS 15.3.3.A.2.b, and CTS 15.3.3.A.2.c requires a unit to be placed into hot shutdown within 6 hours and cold shutdown within 36 hours if a single SI pump (other pump is operable) or valve affecting a single SI subsystem (duplicative function still operable) is not restored to operable status in 72 hours. ITS LCO 3.5.2 Condition A allows 72 hours for an SI subsystem to be inoperable, prior to requiring the unit to be placed into Mode 3 within 6 hours and Mode 4 within 12 hours. Accordingly, the CTS does not specify a specific time limit for placing the unit into Mode 4. The addition of this time limit is a more restrictive change, necessary to establish a bounding limit to exit the Mode of Applicability when an SI subsystem is inoperable. The CTS action to place the unit in cold shutdown is covered in the Description of Changes associated with LCO 3.5.3. **CTS:** NEW **ITS:** LCO 3.05.02 COND B RA B.2
M.04 Rev. A	The Actions contained in 15.3.3.A.3 for a single inoperable RHR subsystem will place the unit in hot shutdown within 6 hours if the RHR System is not returned to an operable status within its allowed out of service time. CTS 15.3.3.A.3 will then require the unit to be maintained greater than 350 degrees (ITS Mode 3). Proposed LCO 3.5.2 Action B will require the unit to be placed into Mode 3 within 6 hours and Mode 4 within 12 hours. The requirement to place the unit in mode 4 within 12 hours is a more restrictive requirement, as the CTS would allow indefinite operation greater than 350 degrees. This requirement places the unit in a Mode of Applicability for which redundant RHR ECCS subsystems are not required. Maintaining redundancy in heat removal is not addressed within this LCO as the Mode of Applicability for this LCO is above the cut in temperature limit for the RHR system. The heat removal aspects are discussed in Section 3.4. **CTS:** NEW **ITS:** LCO 3.05.02 COND B RA B.2
M.05 Rev. A	CTS 15.4.5.I.A.1 and 15.4.5.I.A.2 require Safety Injection System tests to be performed during reactor shutdowns for major fuel loadings. These tests are intended to ensure that all components receive their Safety Injection (SI) signal, appropriate pump motor breakers open and close as well as verifying that all valves actuate and travel to their correct position. The proposed ITS for Point Beach (SR 3.5.2.3 and 4) will require each ECCS pump and each automatic valve in the flowpath that is not locked, sealed, or otherwise secured in position, either start or actuates to its correct position, as applicable, on an actual or simulated actuation signal at least once every 18 months. The CTS and the ITS impose the same testing, but the CTS does not define a specific frequency of performance for these Surveillance, but rather an evolution, which can vary significantly from outage to outage with no bounding limit. Accordingly, the adoption of a bounding frequency (18 months) is a more restrictive change. **CTS:** 15.04.05.I.A.01 **ITS:** SR 3.05.02.03 SR 3.05.02.04

DOC Number	DOC Text
M.06 Rev. A	CTS 15.4.b.II.B.1 requires a visual inspection of each containment sump suction inlet and strainer to verify that there is no restriction or evidence of structural distress or abnormal corrosion at least each refueling outage. The proposed ITS for Point Beach both require a visual inspection of each ECCS train containment sump suction inlet for restriction by debris and inspection of the debris screen for structural distress and abnormal corrosion at least once per 18 months. The CTS requirement is the same as the proposed ITS with the exception of the specified frequency. The CTS does not define a specific frequency of performance for these Surveillance, but rather an evolution, which can vary significantly from outage to outage with no bounding limit. Accordingly, the adoption of a bounding frequency (18 months) is a more restrictive change.

CTS:	ITS:
15.04.05.II.B.01	SR 3.05.02.05

DOC Number	DOC Text
M.07 Rev. A	CTS 15.3.3.A.2 and 15.3.3.A.3 provide separate Conditions and Required Actions for inoperable residual heat removal and safety injection system components. In having separate conditions, simultaneous inoperability of a safety injection and residual heat removal subsystem is allowed. The Point Beach ECCS piping design is capable of supporting cross-train operation of the residual heat removal and safety injection subsystems. However, cross-train operation in the post accident recirculation mode of operation requires local valve manipulations which is not currently addressed by the emergency operating procedures. As such, ITS LCO 3.5.2 only provides Required Actions for a single Train of ECCS being inoperable, thereby limiting the simultaneous inoperability of the safety injection and residual heat removal subsystems to the same ECCS Train.

CTS:	ITS:
15.03.03.A.02	LCO 3.05.02 COND A
15.03.03.A.03	LCO 3.05.02 COND A

DOC Number	DOC Text
R.01 Rev. A	Wisconsin Electric Power Company has utilized the selection criteria provided in the 10 CFR 50.36.ii, and has concluded that the Boric Acid System can be relocated to licensee control. The basis for this conclusion is as follows: The boric acid system ensures negative reactivity control is available for normal operation (normal makeup and chemical shim reactivity control) and provides an alternate method for borating the reactor coolant system. However, this system is not assumed to mitigate any design basis accident or transient. Other systems (e.g., SI pumps) and other borated water sources (RWST) are assumed in the safety analysis. Comparison to Screening Criteria: 1. The boric acid system is not used for, nor capable of, detecting a significant abnormal degradation of the reactor coolant pressure boundary prior to a design basis accident (DBA). 2. The boric acid system is not used to indicate status of, or monitor a process variable, design feature, or operating restriction that is an initial condition of a DBA or transient. 3. The boric acid system is not part of a primary success path in the mitigation of a DBA or transient. 4. As discussed in Section 4.0 (Appendix A, pages A-8 and A-10) and summarized in Table 1 of WCAP-11618, the loss of the boric acid system was found to be a non-significant risk contributor to core damage frequency and offsite releases. Wisconsin Electric Power Company has reviewed this evaluation and considers it applicable to Point Beach Station. The Point Beach IPE confirms this judgment for Point Beach Station. Boration, when needed, can be accomplished with the RWST, as well as the boric acid system.

CTS:	ITS:
15.03.02 T 15.03.02-01	TRM 3.05.01
15.03.02.A	TRM 3.05.01 TRM 3.05.01
15.03.02.B	TRM 3.05.01 TRM 3.05.01
15.03.02.B.01	TRM 3.05.01 TRM 3.05.01
15.03.02.B.02	TRM 3.05.01 TRM 3.05.01
15.03.02.B.03	TRM 3.05.01 TRM 3.05.01
15.03.02.B.03.A	TRM 3.05.01

Description of Changes - NUREG-1431 Section 3.05.02

24-May-00

DOC Number	DOC Text
15.03.02.B.03.A	TRM 3.05.01
15.03.02.B.03.B	TRM 3.05.01
	TRM 3.05.01
15.03.02.C	TRM 3.05.01
	TRM 3.05.01
15.03.02.C.01	TRM 3.05.01
	TRM 3.05.01
15.03.02.C.02	TRM 3.05.01
	TRM 3.05.01
15.03.02.C.03	TRM 3.05.01
	TRM 3.05.01
15.03.02.C.03.A	TRM 3.05.01
	TRM 3.05.01
15.03.02.C.03.B	TRM 3.05.01
	TRM 3.05.01
15.03.02.D	TRM 3.05.01
	TRM 3.05.01
15.03.02.D.01	TRM 3.05.01
	TRM 3.05.01
15.03.02.D.02	TRM 3.05.01
	TRM 3.05.01
15.03.02.D.03	TRM 3.05.01
	TRM 3.05.01
15.04.01 T 15.04.01-01 21	TRM 3.03.01 T 3.03.01-01 04
15.04.01 T 15.04.01-01 22	TRM 3.03.01 T 3.03.01-01 05
15.04.01 T 15.04.01-01 23	TRM 3.03.01 T 3.03.01-01 06
15.04.01 T 15.04.01-02 04	TRM 3.05.01
15.04.01 T 15.04.01-02 20	TRM 3.05.01
15.04.01 T 15.04.01-02 20 (19)	TRM 3.05.01
15.04.01 T 15.04.01-02 31	TRM 3.05.01
15.04.01 T 15.04.01-02 31 (17)	TRM 3.05.01
BASES	TRM 3.05.01

15.4.5 EMERGENCY CORE COOLING SYSTEM AND CONTAINMENT COOLING SYSTEM TESTS

Applicability:

A.10

< See Section 3.6 >

Applies to testing of the Emergency Core Cooling System and the Containment Cooling System.

Objective:

To verify that the subject systems will respond promptly and perform their design functions, if required.

A.11

Specification:

I. System Tests and Surveillances

M.5

A. Safety Injection System

18 months

SR 3.5.2.3
SR 3.5.2.4

1. System tests shall be performed during reactor shutdowns for major fuel reloading. The test shall be performed in accordance with the following procedure:

   a. With the Reactor Coolant System pressure less than or equal to 350 psig and temperature less than or equal to 350°F, a test safety injection signal will be applied to initiate operation of the system. The motor breakers for the safety injection and residual heat removal pumps may be placed in the "test" position or racked in and operable for this test.

2. The test will be considered satisfactory if control board indication and visual observations indicate that all components have received the safety injection signal in the proper sequence and timing.

L.3

A.1

< See Section 3.6 >

A.9

SR 3.5.2.2

containment spray pumps shall be tested in accordance with the Inservice Test Program.

2. Acceptable levels of performance shall be that the pumps start, reach their required developed head at, ~~and operate for at least fifteen minutes on the full-flow test lines.~~ L.2 B Errata #50

B. Other

18 months M.6

SR 3.5.2.5

1. ~~At least every refueling~~, verify by visual inspection each containment sump suction inlet is not restricted by debris and the debris strainers show no evidence of structural distress or abnormal corrosion.

A.4

SR 3.5.2.1

2. Verify each manual, power operated, and automatic valve necessary to insure system operability in the emergency core cooling and containment spray systems that is not locked, sealed, or otherwise secured in position, is in the correct position at least once every 31 days.

< See Section 3.6 >

Basis

A.8

The Safety Injection System and the Containment Spray System are principal plant Safety Systems that are normally inoperative during reactor operation. Complete systems tests cannot be performed when the reactor is operating because a safety injection signal causes containment isolation and a Containment Spray System test requires the system to be temporarily disabled. The method of assuring operability of these systems is therefore to combine systems tests to be performed during refueling shutdowns, with more frequent component tests, which can be performed during reactor operation.

No Significant Hazards Considerations - NUREG-1431 Section 3.05.02

24-May-00

NSHC Number	NSHC Text
A Rev. A	In accordance with the criteria set forth in 10 CFR 50.92, PBNP has evaluated this proposed Technical Specifications change and determined it does not represent a significant hazards consideration. The following is provided in support of this conclusion. 1. Does the change involve a significant increase in the probability or consequences of an accident previously evaluated? The proposed change involves reformatting and rewording of the current Technical Specifications. The reformatting and rewording process involves no technical changes to existing requirements. As such, this change is administrative in nature and does not impact initiators of analyzed events or assumed mitigation of accident or transient events. Therefore, this change does not increase 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 require a physical alteration of the plant (no new or different type of equipment will be installed) or changes in parameters governing normal plant operation. The proposed change will not impose any new or eliminate any old requirements. Thus, this change does not create the possibility of a new or different kind of accident from any accident previously evaluated. 3. Does this change involve a significant reduction in a margin of safety? The proposed change will not significantly reduce the margin of safety because it has no impact on any safety analysis assumptions. This change is administrative. As such, there is no technical change to the requirements and, therefore, there is no reduction in the margin of safety.

NSHC Number	NSHC Text
L.01 Rev. A	1. Does the change involve a significant increase in the probability or consequences of an accident previously evaluated? The addition of a short period of time to allow both SI pump flow paths to be isolated provides for a short period of time to perform pressure isolation valve testing when the system is otherwise required to be operable. During this time, the isolation valves are under direct control from the control room such that system function can be restored expeditiously if called upon. This short period of inoperability cannot increase the probability of an accident and since system function can be restored directly by the control room the consequences of an accident previously evaluated will not be significantly increased. The addition of a temperature limitation to the ECCS LCO for Low Temperature Overpressure Protection (LTOP) will allow the flexibility necessary to enable and disable Safety Injection (SI) pumps prior to entry into and exiting the LTOP Mode of Applicability during Mode 3. This change does not result in any hardware changes. SI pumps status/availability is assumed relative to the initial conditions and function in the mitigation of analyzed events, but is not assumed to be an initiator of any analyzed event. The change will allow operation for a limited period of time (four hours) within a small temperature band (355 to 375 degrees) to restore or disable the second SI pump for Low Temperature Overpressure concerns. Core energy levels are not significantly increased through the added 20 degree allowance (355 to 375 degrees). Therefore, the consequences of an event will not be significantly affected. Therefore, the proposed change does not increase 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 result in any physical alterations to the plant (no new or different type of equipment will be installed). The short period of time (2 hours) for SI flow path inoperability ensures system is restored expeditiously. During the time of inoperability, operators can directly restore the flow path from the control room if called upon. The additional 20 degree temperature band in which the second SI pump may be inoperable is a flexibility required for the enable and disable a second SI pump for LTOP concerns which does not introduce any new or significant challenges to plant operation. This provision allows an appropriate bound for either restoration or disablement of a second pump during a plant heatup and cooldown operations, which is necessary for LTOP concerns. The proposed change will limit unit operation above 350 degrees to a maximum of four hours. Thus, this change does not create the possibility of a new or different kind of accident from any accident previously evaluated. 3. Does this change involve a significant reduction in a margin of safety? During the short period of time two SI flow paths can be isolated, the flow paths are under operator control and can be unisolated from the control room expeditiously if called upon.

NSHC Number	NSHC Text
	This ensures the SI system can perform its function as designed. Therefore, a significant reduction in a margin of safety cannot result. The additional 20 degree temperature band in which the second SI pump may be inoperable is a flexibility required for the enable and disable a second SI pump for LTOP concerns. This limitation allows an appropriate bound for restoration or disablement of a second pump during a plant heatup or cooldown, and is acceptable based upon the reduced thermal energy in the core and sufficient time for manual actuation of the remaining ECCS pumps if necessary. The proposed change will limit unit operation above 350 degrees in this configuration to four hours. Accordingly, this change will have an insignificant effect on margin of safety.
L.02 Rev. B	In accordance with the criteria set forth in 10 CFR 50.92, PBNP has evaluated this proposed Technical Specifications change and determined it does not represent a significant hazards consideration. The following is provided in support of this conclusion. 1. Does the change involve a significant increase in the probability or consequences of an accident previously evaluated? The proposed change does not involve any physical alteration of plant systems, structures or components, changes in parameters governing normal plant operation, or methods of operation. The proposed change results in the deletion of details which are not necessary to describe the actual regulatory requirement, or provide adequate protection of the public health and safety. Accordingly, there will be no significant change in the probability or consequences of accidents 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 any physical alteration of plant systems, structures or components, nor does it alter parameters governing normal plant operation. The proposed change does not introduce a new mode of operation. Therefore, the possibility of a new or different kind of accident from any accident previously evaluated is not created. 3. Does this change involve a significant reduction in a margin of safety? The deletion of details which are not necessary to describe the actual regulatory requirement, or provide adequate protection of the public health and safety, does not result in a significant reduction in the margin of safety.

No Significant Hazards Considerations - NUREG-1431 Section 3.05.02

24-May-00

NSHC Number	NSHC Text
L.03 Rev. B	In accordance with the criteria set forth in 10 CFR 50.92, PBNP has evaluated this proposed Technical Specifications change and determined it does not represent a significant hazards consideration. The following is provided in support of this conclusion. 1. Does the change involve a significant increase in the probability or consequences of an accident previously evaluated? The proposed change does not involve any physical alteration of plant systems, structures or components, changes in parameters governing normal plant operation, or methods of operation. The proposed change results in the deletion of details which are not necessary to describe the actual regulatory requirement, or provide adequate protection of the public health and safety. Accordingly, there will be no significant change in the probability or consequences of accidents 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 any physical alteration of plant systems, structures or components, nor does it alter parameters governing normal plant operation. The proposed change does not introduce a new mode of operation. Therefore, the possibility of a new or different kind of accident from any accident previously evaluated is not created. 3. Does this change involve a significant reduction in a margin of safety? The deletion of details which are not necessary to describe the actual regulatory requirement, or provide adequate protection of the public health and safety, does not result in a significant reduction in the margin of safety.

NSHC Number	NSHC Text
LA Rev. A	In accordance with the criteria set forth in 10 CFR 50.92, PBNP has evaluated this proposed Technical Specifications change and determined it does not represent a significant hazards consideration. The following is provided in support of this conclusion. 1. Does the change involve a significant increase in the probability or consequences of an accident previously evaluated? The proposed change relocates requirements from the Technical Specifications to the Bases, FSAR, or other plant controlled documents. The Bases and FSAR will be maintained using the provisions of 10 CFR 50.59. In addition to 10 CFR 50.59 provisions, the Technical Specifications Bases are subject to the change process in the Administrative Controls Chapter of the ITS. Plant procedures and other plant controlled documents are subject to controls imposed by plant administrative procedures, which endorse applicable regulations and standards. Changes to the Bases, FSAR, or other plant controlled documents will be evaluated in accordance with the requirements of the Bases Control Program in Chapter 5.0 of the ITS, 10 CFR 50.59, or plant administrative processes. Therefore, no increase in the probability or consequences of an accident previously evaluated will be allowed. 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 require a physical alteration of the plant (no new or different type of equipment will be installed) or changes in parameters governing normal plant operation. The proposed change will not impose any different requirements and adequate control of the information will be maintained. Thus, this change does not create the possibility of a new or different kind of accident from any accident previously evaluated. 3. Does this change involve a significant reduction in a margin of safety? The proposed change will not reduce a margin of safety because it has no impact on any safety analysis assumptions. In addition, the requirements to be moved from the Technical Specifications to the Bases, FSAR, or other plant controlled documents are as they currently exist. Future changes to the requirements in the Bases, FSAR, or other plant controlled documents will be evaluated in accordance with the requirements of 10 CFR 50.59, the Bases Control Program in Chapter 5.0 of the ITS, or the applicable plant process and no reduction in a margin of safety will be allowed.

NSHC Number	NSHC Text
M Rev. A	In accordance with the criteria set forth in 10 CFR 50.92, PBNP has evaluated this proposed Technical Specifications change and determined it does not represent a significant hazards consideration. The following is provided in support of this conclusion. 1. Does the change involve a significant increase in the probability or consequences of an accident previously evaluated? The proposed change provides more restrictive requirements for operation of the facility. These more stringent requirements do not result in operation that will increase the probability of initiating an analyzed event and do not alter the assumptions relative to the mitigation of an accident or transient event. These more restrictive requirements continue to ensure process variables, structures, systems and components are maintained consistent with the safety analyses. Therefore, this change does not increase 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 require a physical alteration of the plant (no new or different type of equipment will be installed) or changes in parameters governing normal plant operation. The proposed change does impose different requirements. However, these changes are consistent with assumptions made in the safety analysis. Thus, this change does not create the possibility of a new or different kind of accident from any accident previously evaluated. 3. Does this change involve a significant reduction in a margin of safety? The imposition of more restrictive requirements either has no affect on or increases the margin of safety. Each change is providing additional restrictions to enhance plant safety. These changes are consistent with the safety analysis. Therefore, this change does not involve a reduction in a margin of safety.

No Significant Hazards Considerations - NUREG-1431 Section 3.05.02

24-May-00

NSHC Number	NSHC Text
R Rev. A	In accordance with the criteria set forth in 10 CFR 50.92, PBNP has evaluated this proposed Technical Specifications change and determined it does not represent a significant hazards consideration. The following is provided in support of this conclusion. 1. Does the change involve a significant increase in the probability or consequences of an accident previously evaluated? The proposed change relocates requirements and surveillances for structures, systems, components or variables which did not meet the criteria for inclusion in Technical Specifications as identified in the 10CFR 50.36 Technical Specification Selection Criteria. The affected structures, systems, components or variables are not assumed to be initiators of analyzed events and are not assumed to mitigate accident or transient events. The requirements and surveillances for these affected structures, systems, components or variables will be relocated from the Technical Specifications to an appropriate administratively controlled document and maintained pursuant to 10CFR 50.59. Therefore, this change does not increase 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 require a physical alteration of the plant (no new or different type of equipment will be installed) or change in parameters governing normal plant operation. The proposed change will not impose any different requirements and adequate control of information will be maintained. Thus, this change does not create the possibility of a new or different kind of accident from any accident previously evaluated. 3. Does this change involve a significant reduction in a margin of safety? The proposed change will not reduce a margin of safety because it has no impact on any safety analysis assumptions. In addition, the affected requirement will be relocated to an owner controlled document for which future changes will be evaluated pursuant to the requirements of 10CFR 50.59. Therefore, this change does not involve a reduction in a margin of safety.

Description of Changes - NUREG-1431 Section 3.05.03

24-May-00

DOC Number	DOC Text

A.01
Rev. A

In the conversion of Point Beach current Technical Specifications (CTS) to the proposed plant specific Improved Technical Specifications (ITS), certain wording preferences or conventions are adopted which do not result in technical changes (either actual or interpretational). Editorial changes, reformatting, and revised numbering are adopted to make the ITS consistent with the Standard Technical Specifications, Westinghouse Plants, NUREG-1431, Revision 1 (i.e., Improved Standard Technical Specifications (ISTS)).

CTS:	ITS:
15.03.03 APPL	LCO 3.05.03
15.04.05.II.A.01	SR 3.05.03.01 - SR 3.05.02.02
15.04.05.II.A.02	SR 3.05.03.01 - SR 3.05.02.02

A.02
Rev. A

The CTS states that the ECCS systems and components listed in Specification 15.3.3.A.1.c, d, e, f, and g (SI pumps, RHR pumps and heat exchangers, SI discharge isolation valves, associated valves, interlocks, and piping) are required to be operable prior to the reactor being made critical.

However the Actions contained in CTS 15.3.3.A.2 for the SI pump/subsystem will place the unit in cold shutdown if the pump/subsystem is not returned to an operable status within its allowed out of service time, implying an Applicability of Modes 1, 2, 3, and 4 (ITS Modes). CTS 15.3.15.B modifies the requirement of CTS 15.3.3.A.1.c (required number of SI pumps) by stating that one of the two high head Safety Injection pumps are to be rendered inoperable whenever LTOP is required (less than 355 degrees). Accordingly, the CTS requires only one SI pump/subsystem to be operable in Mode 4 (200 - 350 degrees).

Similarly, the Actions contained in CTS 15.3.3.A.3 for a single inoperable RHR subsystem allow the unit to be maintained greater than 350 degrees (ITS Mode 3) indefinitely, as long as the RHR subsystem is not being relied upon for redundancy in decay heat removal. The CTS does not contain any Actions for the RHR subsystem below 350 degrees unless it is being relied upon for redundancy in decay heat removal. Accordingly, the CTS allows indefinite operation in the equivalent to ITS Mode 4 with only a single RHR pump/subsystem operable.

ITS LCO 3.5.3 requires a single train of ECCS (one SI and one RHR subsystem) to be operable in Mode 4 as the CTS does, making the Applicability for a single pump system an administrative change. However, the CTS does not require the SI and RHR subsystems to be associated with the same ECCS train. Requiring both ECCS pump systems to be associated with the same train is addressed in Description of Change M.4 of this LCO.

CTS:	ITS:
15.03.03.A.01	LCO 3.05.03
15.03.15.B.01	LCO 3.05.03

Description of Changes - NUREG-1431 Section 3.05.03

24-May-00

DOC Number	DOC Text
A.03 Rev. A	The Bases of the current Technical Specifications for this LCO have been completely replaced by the revised Bases reflecting the format and applicable content of the Improved Technical Specifications for Point Beach. The proposed Bases are based on NUREG 1431 Rev. 1. The proposed Bases for this LCO are consistent and supportive of the proposed LCO, and accordingly is administrative.

CTS:	ITS:
BASES	B 3.05.03

DOC Number	DOC Text
A.04 Rev. A	CTS 15.4.2.B.3 and CTS 15.4.5.II.A.1 require the SI and RHR pumps to be tested in accordance with ASME Section XI. ITS SR 3.5.3.1 references the performance of SR 3.5.2.2 which requires each ECCS pump to be tested in accordance with the Inservice Testing Program. Accordingly, while presented in a different fashion than the CTS, the ITS will continue to require ASME pump testing.

CTS:	ITS:
15.04.02.B.03	SR 3.05.03.01 - SR 3.05.02.02
15.04.05.II.A.01	SR 3.05.03.01 - SR 3.05.02.02

DOC Number	DOC Text
A.05 Rev. A	NUREG 1431 contains a note as part of SR 3.5.3.1 which allows an RHR train to be considered operable during alignment and operation for decay heat removal, providing the train is capable of being manually realigned to the ECCS mode of operation. This note has been moved to the LCO Section of the proposed ITS in accordance with TSTF 90 which has been approved for incorporation into Revision 2 of NUREG 1431. The CTS, while not explicitly stated in the same fashion, recognizes this allowance through the application of LCO 15.3.3.A.3, which provide LCO Requirements and Actions for redundant heat removal trains in Mode 4. Accordingly, while presented differently, this latitude exists in the CTS making this change administrative.

CTS:	ITS:
NEW	LCO 3.05.03 NOTE

DOC Number	DOC Text
A.06 Rev. A	The CTS provides an introductory statement (Applicability) which simply states which systems/components are addressed within a given section. This same information while worded differently is contained within the title of each ITS LCO. Accordingly, this change is a change in format with no change in technical requirement.

CTS:	ITS:
15.03.03 APPL	LCO 3.05.03
15.04.05 APPL	LCO 3.05.03

Description of Changes - NUREG-1431 Section 3.05.03

24-May-00

DOC Number	DOC Text
A.07 Rev. A	The CTS provides an introductory statement (Objective) at the beginning of this Section of the Technical Specifications which provide a brief summary of the purpose for this Section. This information is contained in the Bases Section of the ITS. This information does not establish any regulatory requirements for the systems and components addressed within this Section. Accordingly, deletion of this information does not alter any requirement set forth in the Technical Specifications. This change is administrative and consistent with the format and presentation for the ITS as provided in NUREG 1431.

CTS:	ITS:
15.03.03 OBJ	DELETED
15.04.05 OBJ	DELETED

A.08 Rev. A	CTS 15.3.3.A.1.f requires the isolation valves in the discharge header of the SI system to be in the open position. CTS 15.4.5.II.B.2 requires each manual, power operated and automatic valve necessary to ensure system operability, that is not locked, sealed, or otherwise secured in position, to be verified to be in its correct position at least once every 31 days. ITS SR 3.5.3.1 referencing SR 3.5.2.1 requires each manual, power operated and automatic valve in the ECCS flowpath, that is not locked, sealed, or otherwise secured in position, to be verified to be in its correct position at least once every 31 days. All surveillance requirements associated with an LCO are required to be met in order to fulfill the LCO. Accordingly, there is no change in requirement, making this change administrative.

CTS:	ITS:
15.03.03.A.01.F	SR 3.05.03.01 - SR 3.05.02.01
15.04.05.II.B.02	SR 3.05.03.01 - SR 3.05.02.01

A.09 Rev. A	CTS 15.3.3.A.1.g, requires the interlocks associated with the ECCS pumps and valves which are required to function during accident conditions to be operable. CTS 15.4.5.I.A.1 requires the performance of a system test during reactor shutdowns for major refueling outages, which ultimately verifies that all components received their actuation signal and actuate to their correct positions. ITS SR 3.5.3.1 referencing SR 3.5.2.3 requires each ECCS automatic valve in the flowpath that is not locked, sealed, or otherwise secured in position, to be tested to ensure it actuates to its correct position on an actual or simulated actuation signal. Similarly, SR 3.5.3.1 referencing 3.5.2.4 requires each ECCS pump to be tested to ensure it actuates on an actual or simulated actuation signal. Accordingly, both the CTS and ITS require these attributes, making this change administrative, consistent with the format and presentation of NUREG 1431.

CTS:	ITS:
15.03.03.A.01.G	SR 3.05.03.01 - SR 3.05.02.03
	SR 3.05.03.01 - SR 3.05.02.04

DOC Number	DOC Text
L.01 Rev. A	The CTS states that the ECCS systems and component listed in Specification 15.3.3.A.1.d, e, and g (RHR System, associated valves, interlocks and piping) are required to be operable prior to the reactor being made critical. CTS 15.3.3.A.1.d and e specify that two RHR pumps and heat exchangers are required to be operable. However, the Actions contained in CTS 15.3.3.A.3 for a single inoperable RHR subsystem allow the unit to be maintained greater than 350 degrees indefinitely, as long as the RHR subsystem is not being relied upon for redundancy in decay heat removal. The CTS does not provide any explicit Actions below 350 degrees unless an RHR system is being relied upon to provide redundancy in decay heat removal; therefore, entry into 15.3.0.B is required if one or both RHR loops are inoperable below 350 degrees when RHR is not being relied upon for heat removal. Entry in 15.3.0.b requires the unit to be placed into cold shutdown within 37 hours. Proposed LCO 3.5.3 will require only one train of ECCS to be operable in Mode 4 (200 to 350 degrees) which is consistent with the CTS Actions which allows a single train of RHR to be inoperable, while remaining in Mode 3 indefinitely. Proposed Condition A of LCO 3.5.3, provides a Required Action which addresses the inoperability of the ECCS function of the RHR system, recognizing its shared function relative to residual heat removal. The appropriate action in this condition is to initiate actions to restore the RHR train to operable status immediately as with no RHR subsystem available, there may be no means available to achieve a cold shutdown condition in a controlled manner making it appropriate to maintain hot conditions to maintaining the availability of the steam generators as a heat sink. Further, the core heat removal aspects of the RHR system is addressed specifically in Section 3.4 of this conversion package.

CTS:	ITS:
15.03.03.A.03	LCO 3.05.03 COND A LCO 3.05.03 COND A RA A.1
15.03.03.A.03.A	LCO 3.05.03 COND A LCO 3.05.03 COND A RA A.1
15.03.03.A.03.B	LCO 3.05.03 COND A LCO 3.05.03 COND A RA A.1
15.03.03.A.03.C	LCO 3.05.03 COND A LCO 3.05.03 COND A RA A.1

DOC Number	DOC Text
L.02 Rev. B	CTS 15.4.5.II.A.2 specifies the conditions and manner in which ECCS pump testing must be conducted (run for at least 15 minutes using the full flow test line). This requirement to run the pumps for at least 15 minutes is an arbitrary requirement with no fundamental safety basis. Therefore, these details are being removed. The proposed ITS specifies the safety objective that must be fulfilled by the surveillance tests, while leaving the details associated with testing methods and acceptance verifications to licensee control. These type of details are better suited for procedural control and are not required to be in the ITS to provide adequate protection to the public health and safety. Changes to plant procedures and other plant controlled documents are subject to controls imposed by plant administrative procedures, which endorse applicable regulations and standards.

CTS:	ITS:
15.04.05.II.A.02	DELETED

DOC Number	DOC Text
L.03 Rev. B	CTS 15.4.5.I.A.1.a specifies the conditions and manner in which ECCS testing must be conducted (RCS pressure less than or equal to 350 psig and RCS temperature less than or equal to 350 degrees, pump breakers may be placed into the "test" position), and CTS 15.4.5.I.A.2 specifies the manner in which equipment operation is to be verified (control board indication and visual indications). These details have been deleted from the Technical Specification, and moved to licensee control. These details are not necessary to describe the actual regulatory requirement (i.e. verification that equipment actuates to it correct position). These performance and acceptance details are not required to be in the ITS to provide adequate protection of public health and safety, as the actual requirement (verification of equipment actuation) has been maintained in the Technical Specifications. Changes to plant procedures and other plant controlled documents are subject to controls imposed by plant administrative procedures, which endorse applicable regulations and standards.

CTS:	ITS:
15.04.05.I.A.01	DELETED
15.04.05.I.A.01.A	DELETED
15.04.05.I.A.02	DELETED

DOC Number	DOC Text	
LA.01 Rev. A	CTS 15.3.3.A.1.e and CTS 15.3.3.A.1.g list the RHR Heat Exchangers, valves, and piping associated with the SI and RHR Systems required to be operable to fulfill the ECCS LCO requirement. These are all attributes associated with system design and configuration, which are adequately captured through application of the definition of operability, and accordingly are still encompassed within the LCOs for ECCS. As such, these details are not required to be in the ITS to provide adequate protection of public health and safety. These attributes are discussed within the Bases for the proposed Point Beach ITS, but have been deleted from the Technical Specifications, changes to these details will be controlled in accordance with the provisions of the Bases Control Program described in Chapter 5 of the Technical Specifications.	
	CTS:	**ITS:**
	15.03.03.A.01.E	DELETED
	15.03.03.A.01.G	DELETED
LA.02 Rev. B	Not Used.	
	CTS:	**ITS:**
	N/A	N/A
LA.03 Rev. B	Not Used.	
	CTS:	**ITS:**
	N/A	N/A

DOC Number	DOC Text
M.01 Rev. A	The CTS states that the ECCS systems and components listed in Specification 15.3.3.A.1.c, f, and g (two SI pumps, SI discharge isolation valves, associated valves, interlocks, and piping) are required to be operable prior to the reactor being made critical. CTS 15.3.15.B modifies the requirement of CTS 15.3.3.A.1.c (required number of SI pumps) by stating that one of the two high head Safety Injection pumps are to be rendered inoperable whenever LTOP is required (less than 355 degrees). CTS Action 15.3.3.A.2 is contingent upon the redundant pump/function being operable as stated in CTS 15.3.3.A.2.b and c therefore the CTS does not provide an explicit action for the only required train of SI being inoperable, which results in entry into CTS 15.3.0.B. CTS 15.3.0.B requires the unit to be placed into cold shutdown within 37 hours. Proposed LCO 3.5.3 requires one train of ECCS (SI subsystem) to be operable in Mode 4, as the CTS does. Proposed Condition and Required Actions B and C of LCO 3.5.3 explicitly address the inoperability of the Safety Injection subsystem in Mode 4. The proposed actions will require the unit to be placed into Mode 5 within 24 hours if the required train of SI is not restored to operable status within one hour. This change will require the unit to be placed into cold shutdown in a shorter time frame than the CTS, and is, therefore, more restrictive. The one hour completion time for restoration of an inoperable subsystem ensures that prompt action is taken to restore the subsystem. Requiring the unit to be placed into Mode 5 is appropriate as ECCS is not required in that Mode. This change will require the unit to be placed into cold shutdown in a shorter time frame, and, as such, is more restrictive.

CTS:	ITS:
15.03.03.A.02	LCO 3.05.03 COND B LCO 3.05.03 COND B RA B.1 LCO 3.05.03 COND C LCO 3.05.03 COND C RA C.1
15.03.03.A.02.B	LCO 3.05.03 COND B LCO 3.05.03 COND B RA B.1 LCO 3.05.03 COND C LCO 3.05.03 COND C RA C.1
15.03.03.A.02.C	LCO 3.05.03 COND B LCO 3.05.03 COND B RA B.1 LCO 3.05.03 COND C LCO 3.05.03 COND C RA C.1

DOC Number	DOC Text
M.02 Rev. A	CTS 15.4.b.II.B.1 requires a visual inspection of each containment sump suction inlet and strainer to verify that there is no restrictions or evidence of structural distress or abnormal corrosion at least each refueling outage. The proposed ITS for Point Beach both require a visual inspection of each ECCS train containment sump suction inlet for restriction by debris and inspection of the debris screen for structural distress and abnormal corrosion at least once per 18 months. The CTS requirement is the same as the proposed ITS with the exception of the specified frequency. The CTS does not define a specific frequency of performance for these Surveillance, but rather an evolution, which can vary significantly from outage to outage with no bounding limit. Accordingly, the adoption of a bounding frequency (18 months) is a more restrictive change.

CTS:	ITS:
15.04.05.II.B.01	SR 3.05.03.01 - SR 3.05.02.05

DOC Number	DOC Text
M.03 Rev. A	CTS 15.4.5.I.A.1 and 15.4.5.I.A.2 require Safety Injection System tests to be performed during reactor shutdowns for major fuel loadings. These tests are intended to ensure that all components receive their Safety Injection (SI) signal, appropriate pump motor breakers open and close as well as verifying that all valves actuate and travel to their correct position. The proposed ITS for Point Beach (SR 3.5.3.1 reference to SR 3.5.2.3 and SR 3.5.3.1 reference to SR 3.5.2.4) will require each ECCS pump and each automatic valve in the flowpath that is not locked, sealed, or otherwise secured in position, either start or actuates to its correct position, as applicable, on an actual or simulated actuation signal at least once every 18 months. The CTS and the ITS impose the same testing, but the CTS does not define a specific frequency of performance for these Surveillance, but rather an evolution, which can vary significantly from outage to outage with no bounding limit. Accordingly, the adoption of a bounding frequency (18 months) is a more restrictive change.

CTS:	ITS:
15.04.05.I.A.01	SR 3.05.03.01 - SR 3.05.02.03 SR 3.05.03.01 - SR 3.05.02.04
15.04.05.I.A.02	SR 3.05.03.01 - SR 3.05.02.03 SR 3.05.03.01 - SR 3.05.02.04

DOC Number	DOC Text
M.04 Rev. A	As addressed in Description of Change A.2 of this LCO, the CTS requires one SI and one RHR pump system to be operable in the equivalent of ITS Mode 4. However, the CTS does not require the SI and RHR pump systems to be associated with the same train of ECCS. The Point Beach ECCS piping design is capable of supporting cross-train operation of the residual heat removal and safety injection subsystems. However, cross-train operation in the post accident recirculation mode of operation requires local valve manipulations which is not currently addressed by the emergency operating procedures. As such, ITS LCO 3.5.3 will require a single Train of ECCS to be operable, thereby requiring the SI and RHR pump systems to be in the same ECCS Train.

CTS:	ITS:
15.03.03.A.01.C	LCO 3.05.03
15.03.03.A.01.D	LCO 3.05.03

15.4.5 EMERGENCY CORE COOLING SYSTEM AND CONTAINMENT COOLING SYSTEM TESTS

Applicability:

A.6

< See Section 3.6 >

Applies to testing of the Emergency Core Cooling System and the Containment Cooling System.

Objective:

To verify that the subject systems will respond promptly and perform their design functions, if required.

A.7

Specification:

I. System Tests and Surveillances

M.3

A. Safety Injection System

18 months

SR 3.5.3.1
SR 3.5.2.3
SR 3.5.2.4

1. System tests shall be performed during reactor shutdowns for major fuel reloading. The test shall be performed in accordance with the following procedure:

   a. With the Reactor Coolant System pressure less than or equal to 350 psig and temperature less than or equal to 350°F, a test safety injection signal will be applied to initiate operation of the system. The motor breakers for the safety injection and residual heat removal pumps may be placed in the "test" position or racked in and operable for this test.

2. The test will be considered satisfactory if control board indication and visual observations indicate that all components have received the safety injection signal in the proper sequence and timing.

L.3

A.1

< See Section 3.6 >

A.1

SR 3.5.3.1 -
SR 3.5.2.2

containment spray pumps shall be tested in accordance with the Inservice Test Program.

2. Acceptable levels of performance shall be that the pumps start, reach their required developed head at, ~~and operate for at least fifteen minutes on the full-flow test lines.~~ L.2 B Errata #50

B. Other

18 months M.2

SR 3.5.3.1 -
SR 3.5.2.5

1. ~~At least every refueling~~, verify by visual inspection each containment sump suction inlet is not restricted by debris and the debris strainers show no evidence of structural distress or abnormal corrosion.

A.1

SR 3.5.3.1 -
SR 3.5.2.1

2. Verify each manual, power operated, and automatic valve necessary to insure system operability in the emergency core cooling and containment spray systems that is not locked, sealed, or otherwise secured in position, is in the correct position at least once every 31 days.

< See Section 3.6 >

Basis

A.3

The Safety Injection System and the Containment Spray System are principal plant Safety Systems that are normally inoperative during reactor operation. Complete systems tests cannot be performed when the reactor is operating because a safety injection signal causes containment isolation and a Containment Spray System test requires the system to be temporarily disabled. The method of assuring operability of these systems is therefore to combine systems tests to be performed during refueling shutdowns, with more frequent component tests, which can be performed during reactor operation.

No Significant Hazards Considerations - NUREG-1431 Section 3.05.03

24-May-00

NSHC Number	NSHC Text
A Rev. A	In accordance with the criteria set forth in 10 CFR 50.92, PBNP has evaluated this proposed Technical Specifications change and determined it does not represent a significant hazards consideration. The following is provided in support of this conclusion. 1. Does the change involve a significant increase in the probability or consequences of an accident previously evaluated? The proposed change involves reformatting and rewording of the current Technical Specifications. The reformatting and rewording process involves no technical changes to existing requirements. As such, this change is administrative in nature and does not impact initiators of analyzed events or assumed mitigation of accident or transient events. Therefore, this change does not increase 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 require a physical alteration of the plant (no new or different type of equipment will be installed) or changes in parameters governing normal plant operation. The proposed change will not impose any new or eliminate any old requirements. Thus, this change does not create the possibility of a new or different kind of accident from any accident previously evaluated. 3. Does this change involve a significant reduction in a margin of safety? The proposed change will not significantly reduce the margin of safety because it has no impact on any safety analysis assumptions. This change is administrative. As such, there is no technical change to the requirements and, therefore, there is no reduction in the margin of safety.

No Significant Hazards Considerations - NUREG-1431 Section 3.05.03

24-May-00

NSHC Number	NSHC Text
L.01 Rev. A	In accordance with the criteria set forth in 10 CFR 50.92, PBNP has evaluated this proposed Technical Specifications change and determined it does not represent a significant hazards consideration. The following is provided in support of this conclusion. 1. Does the change involve a significant increase in the probability or consequences of an accident previously evaluated? This change does not result in any hardware changes. The components covered by this Technical Specification are not assumed to be initiators of any analyzed event. The Shutdown Actions, Mode of Applicabilities, and required number of ECCS components are not precursors to any analyzed events. Therefore the probability associated with analyzed events is unchanged. The proposed Applicabilities, minimum equipment requirements and shutdown actions are based on stable unit conditions associated with MODE 4, the reduced thermal energy in the core, sufficient time for manual actuation of the remaining ECCS pumps to mitigate a Design Basis Accidents as necessary, and the assumption that single failures in the ECCS system are not assumed below Mode 3. As such, there is no significant increase in the consequence 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 require a physical alteration of the plant (no new or different type of equipment will be installed) or changes in parameters governing normal plant operation. The proposed change establishes Applicabilities, Required Actions, and complements of components reflective of assumptions made in the Accident Analysis. As such, this change does not create the possibility of a new or different kind of accident from any accident previously evaluated. 3. Does this change involve a significant reduction in a margin of safety? This change proposes equipment Applicabilities, minimum numbers of components/subsystems and shutdown actions based on stable unit conditions associated with MODE 4, the reduced thermal energy in the core, and sufficient time for manual actuation of the remaining ECCS pumps, assuming no single failure within the ECCS subsystem. In addition, this change is reflective of assumptions made in the accident analysis for Point Beach. Therefore, this change does not involve a significant reduction in a margin of safety.

NSHC Number	NSHC Text
L.02 Rev. B	In accordance with the criteria set forth in 10 CFR 50.92, PBNP has evaluated this proposed Technical Specifications change and determined it does not represent a significant hazards consideration. The following is provided in support of this conclusion. 1. Does the change involve a significant increase in the probability or consequences of an accident previously evaluated? The proposed change does not involve any physical alteration of plant systems, structures or components, changes in parameters governing normal plant operation, or methods of operation. The proposed change results in the deletion of details which are not necessary to describe the actual regulatory requirement, or provide adequate protection of the public health and safety. Accordingly, there will be no significant change in the probability or consequences of accidents 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 any physical alteration of plant systems, structures or components, nor does it alter parameters governing normal plant operation. The proposed change does not introduce a new mode of operation. Therefore, the possibility of a new or different kind of accident from any accident previously evaluated is not created. 3. Does this change involve a significant reduction in a margin of safety? The deletion of details which are not necessary to describe the actual regulatory requirement, or provide adequate protection of the public health and safety, does not result in a significant reduction in the margin of safety.

No Significant Hazards Considerations - NUREG-1431 Section 3.05.03

24-May-00

NSHC Number	NSHC Text
L.03 Rev. B	In accordance with the criteria set forth in 10 CFR 50.92, PBNP has evaluated this proposed Technical Specifications change and determined it does not represent a significant hazards consideration. The following is provided in support of this conclusion. 1. Does the change involve a significant increase in the probability or consequences of an accident previously evaluated? The proposed change does not involve any physical alteration of plant systems, structures or components, changes in parameters governing normal plant operation, or methods of operation. The proposed change results in the deletion of details which are not necessary to describe the actual regulatory requirement, or provide adequate protection of the public health and safety. Accordingly, there will be no significant change in the probability or consequences of accidents 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 any physical alteration of plant systems, structures or components, nor does it alter parameters governing normal plant operation. The proposed change does not introduce a new mode of operation. Therefore, the possibility of a new or different kind of accident from any accident previously evaluated is not created. 3. Does this change involve a significant reduction in a margin of safety? The deletion of details which are not necessary to describe the actual regulatory requirement, or provide adequate protection of the public health and safety, does not result in a significant reduction in the margin of safety.

NSHC Number	NSHC Text
LA Rev. A	In accordance with the criteria set forth in 10 CFR 50.92, PBNP has evaluated this proposed Technical Specifications change and determined it does not represent a significant hazards consideration. The following is provided in support of this conclusion. 1. Does the change involve a significant increase in the probability or consequences of an accident previously evaluated? The proposed change relocates requirements from the Technical Specifications to the Bases, FSAR, or other plant controlled documents. The Bases and FSAR will be maintained using the provisions of 10 CFR 50.59. In addition to 10 CFR 50.59 provisions, the Technical Specifications Bases are subject to the change process in the Administrative Controls Chapter of the ITS. Plant procedures and other plant controlled documents are subject to controls imposed by plant administrative procedures, which endorse applicable regulations and standards. Changes to the Bases, FSAR, or other plant controlled documents will be evaluated in accordance with the requirements of the Bases Control Program in Chapter 5.0 of the ITS, 10 CFR 50.59, or plant administrative processes. Therefore, no increase in the probability or consequences of an accident previously evaluated will be allowed. 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 require a physical alteration of the plant (no new or different type of equipment will be installed) or changes in parameters governing normal plant operation. The proposed change will not impose any different requirements and adequate control of the information will be maintained. Thus, this change does not create the possibility of a new or different kind of accident from any accident previously evaluated. 3. Does this change involve a significant reduction in a margin of safety? The proposed change will not reduce a margin of safety because it has no impact on any safety analysis assumptions. In addition, the requirements to be moved from the Technical Specifications to the Bases, FSAR, or other plant controlled documents are as they currently exist. Future changes to the requirements in the Bases, FSAR, or other plant controlled documents will be evaluated in accordance with the requirements of 10 CFR 50.59, the Bases Control Program in Chapter 5.0 of the ITS, or the applicable plant process and no reduction in a margin of safety will be allowed.

NSHC Number	NSHC Text
M Rev. A	In accordance with the criteria set forth in 10 CFR 50.92, PBNP has evaluated this proposed Technical Specifications change and determined it does not represent a significant hazards consideration. The following is provided in support of this conclusion. 1. Does the change involve a significant increase in the probability or consequences of an accident previously evaluated? The proposed change provides more restrictive requirements for operation of the facility. These more stringent requirements do not result in operation that will increase the probability of initiating an analyzed event and do not alter the assumptions relative to the mitigation of an accident or transient event. These more restrictive requirements continue to ensure process variables, structures, systems and components are maintained consistent with the safety analyses. Therefore, this change does not increase 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 require a physical alteration of the plant (no new or different type of equipment will be installed) or changes in parameters governing normal plant operation. The proposed change does impose different requirements. However, these changes are consistent with assumptions made in the safety analysis. Thus, this change does not create the possibility of a new or different kind of accident from any accident previously evaluated. 3. Does this change involve a significant reduction in a margin of safety? The imposition of more restrictive requirements either has no affect on or increases the margin of safety. Each change is providing additional restrictions to enhance plant safety. These changes are consistent with the safety analysis. Therefore, this change does not involve a reduction in a margin of safety.