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Nuclear Management Company, LLC
Point Beach Nuclear Plant
6610 Nuclear Road
Two Rivers, WI 54241

NPL 2001-0032

10 CFR 50.90

February 06, 2001

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

Ladies/Gentlemen:

DOCKETS 50-266 AND 50-301
SUPPLEMENT 9 TO APPLICATION FOR AMENDMENT TO FACILITY OPERATING LICENSE APPENDIX A: TECHNICAL SPECIFICATIONS IMPROVEMENT PROJECT
RESPONSE TO RAI ON ITS SECTIONS 3.3.1 AND 5.0
POINT BEACH NUCLEAR PLANT, UNITS 1 AND 2

On November 15, 1999, Wisconsin Electric Power Company (WE), then 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 letters dated November 17, 2000 and January 25, 2001, the NRC issued a Request for Additional Information (RAI) to Nuclear Management Company, LLC (NMC) on ITS Sections 3.3 and 5.0, respectively. Additionally, during a telephone conference with Point Beach personnel on December 14, 2000, NRC Staff requested additional information on beyond scope issues related to ITS 3.3.1.

Attachment 1 of this letter includes the NMC response to the Staff's questions related to ITS Section 5.0 and 3.3.1, 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

A-001

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. D". The balance of the ITS Section 3.3 RAI questions will be addressed in a future supplement.

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

D
RAI 3.4.1-1

The revision bar identifies the section that has been revised; the D in the triangle identifies revision D; and the RAI number identifies which RAI question the revision relates to. The old pages in the original submittal should be replaced with the new pages enclosed with this letter, following the instructions of attachment 2.

Additional changes to the conversion package for the subject ITS Sections have been identified as a result of ITS reviews by NMC 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.

NMC 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, NMC 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.

NMC 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.

Other supplements to the PBNP ITS submittal, in response to previous RAIs, are listed for reference:

- Supplement 2 dated June 15, 2000 (ITS sections 2.0, 3.1, 3.2, 3.5; letter NPL 2000-0260)
- Supplement 3 dated June 19, 2000 (ITS section 3.6; letter NPL 2000-0271)
- Supplement 4 dated July 28, 2000 (ITS section 3.8; letter NPL 2000-0341)
- Supplement 5 dated August 17, 2000 (ITS sections 3.4, 3.9; letter NPL 2000-0371)
- Supplement 6 dated September 14, 2000 (ITS section 5.5; letter NPL 2000-0411)
- Supplement 7 dated October 19, 2000 (ITS sections 3.6, 3.7.4, 3.7.5; letter NPL 2000-0465)
- Supplement 8 dated December 21, 2000 (ITS section 1.0; letter NPL 2000-0549)

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 NMC 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,

Mark Reddemann
Site Vice President

Subscribed to and sworn before me
on this 6 day of February, 2001

Notary Public, State of Wisconsin

My Commission expires on Oct. 17, 2004.

JG/jlk

Attachments

Enclosure

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

DOCKETS 50-266 AND 50-301
RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION
TECHNICAL SPECIFICATIONS IMPROVEMENT PROJECT SECTIONS 3.3.1 AND 5.0
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 November 17, 2000 and January 25, 2001, and a telephone request on December 14, 2000.

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

3.3.1-01 DOC LA1, M.2, JFD 28, 30, 31, 54
ITS 3.3.1, Table 3.3.1-1, Functions 1, 3, 4, 10, 12, 14
CTS 15.3.5, Table 15.3.5-2, Functions 1, 3, 4, 12, 14.b

The CTS specifies the Total Number of Channels, the Number of Channels to Trip, and the Minimum Operable Channels. The STS specifies only the Required Channels. For most functions the ITS sets the Required Channels equal to Total Number of Channels. For the Functions listed above the ITS uses the CTS Minimum Operable Channels instead of the Total Number of Channels as the Required Channels in the ITS. Comment: The STS format is to use the (CTS) total number of channels in the Required Channels column. Use of the CTS Minimum Operable Channels is inconsistent with the STS format. Use of Minimum Operable Channels for these functions while Total Number of Channels is used for all other functions also creates an internal inconsistency in the ITS.

Revise the ITS to use total number of channels. This will also require revision of the related ITS actions to be consistent with one channel inoperable and the STS. DOC LA.1, will also need to be modified to reflect that, in accordance with the STS, the Total Number of Channels is always pertinent to the ITS requirement.

Response:

For the functions listed above, the CTS does not provide required actions if one channel is inoperable, resulting in one less than the total number of channels being operable. The actions required by the CTS for these functions are related to the number of channels required by column 3, "Minimum Operable Channels." Despite this fact, the ITS has been revised to use the total number of channels for each of the above functions. This change also requires revision of the associated required actions to provide compensatory measures consistent with the level of degradation for one inoperable channel.

3.3.1-02 DOC L.2, L.5
ITS 3.3.1, Actions B, D, F ,G, H, K, L
CTS 15.3.5, Table 15.3.5-2, Footnote **

The CTS allows continued operation with one channel inoperable if the affected channel is placed in trip within one hour. The ITS actions adopt the STS requirement to place the channel in trip within 6 hours. Comment: The extended Completion Time is justified in DOC L.2 based upon the analysis contained in WCAP-10271-P-A, Supplement 2. This is consistent with the basis for the STS. The Safety Evaluation Reports for WCAP-10271 require that applicants for the proposed Technical Specification changes for individual plants must confirm the applicability of the generic analysis of the WCAP. The applicability of the WCAP-10271 analysis to Point Beach has not been discussed. Adopting the WCAP as the basis for Completion Times is a technical change that should be the subject of a separate technical evaluation. Note that the extended Completion Time justified by DOC L.5 appears to have been selected to be consistent with the other Completion Times discussed in DOC L.2, although this is not stated in L.5. Revise ITS Completion Times to be consistent with the CTS.

Response:

The ITS actions have been revised to be consistent with the requirements of the CTS.

3.3.1-03 Not used.

3.3.1-04 DOC A.8
ITS 3.3.1 Table 3.3.1-1, Functions 7.a, 9, 10.b - Applicable Modes, Note (d), (f)
CTS 15.3.5, Table 15.3.5-2, Functions 7, 8, 10.b - Permissible Bypass Conditions
CTS 15.2.3, Specification 2.A

The CTS specifies that the pressurizer low pressure, high pressurizer level, and reactor coolant flow for both loops functions be unblocked when Power range nuclear flux ≥ 9% (±1%) or Turbine load ≥ 10%. It is proposed that the Applicable Modes in the ITS be Mode 1 with Thermal Power > 10% RTP. Comment: This change should be classified L. The specific change is less restrictive and is not explicitly justified. In particular deletion of the specific measurements to be used and the change of the nominal unblock value from 9% to 10%, is not addressed. Provide specific DOC for this change.

Response:

The applicable Modes for the pressurizer low pressure, high pressurizer level and reactor coolant flow (two loop) function have been changed to Mode 1, above the P-7 interlock. The Required

Actions have also been revised to be consistent with the Mode of Applicability for these functions.

3.3.1-05 DOC A.8
ITS 3.3.1 Table 3.3.1-1, Function 9.a - Applicable Modes, Note (e)
CTS 15.3.5, Table 15.3.5-2, Function 10.a - Permissible Bypass Conditions
CTS 15.2.3 Specification 2.B

The CTS specifies that the Low Reactor Coolant System Low Flow in One Loop trip be unblocked at ≥ 50% of rated power. This is changed in the ITS to applicability in Mode 1 at greater than (but not equal to) 50% RTP. Comment: This change should be classified as L. The change to not require the function to be operable at 50% RTP has not been discussed. Provide specific DOC for this change.

Response:

The applicable Mode for the low reactor coolant system flow (single loop) function has been changed to Mode 1, above the P-8 interlock. The Required Actions have also been revised to be consistent with the Mode of Applicability for this trip function.

3.3.1-06 Not used.

3.3.1-07 DOC A.8
ITS 3.3.1 Table 3.3.1-1, Function 15 - Applicable Modes Note (i)CTS 15.3.5, Table 15.3.5-2, Function 10.b - Permissible Bypass Conditions

The CTS requires Operability of the Turbine Trip functions in all Modes. It does not explicitly specify that they may be blocked when power ≤ 50% RTP. The CTS does, however, provide for reducing power below 50% as the allowed action for 2 or more channels inoperable. The proposed ITS limits applicability for this function to Thermal Power > 50% RTP, or no circulating water pump breakers closed, or high condenser pressure. Comment: This change should be classified as L. The specific change to the tech specs has not been discussed. Under the current CTS, at least one channel is required to be operable in all Modes (presumably Modes 1 & 2). Operation with one operable channel is allowed if power ≤ 50% RTP. The proposed ITS allows the entire function to be inoperable if power ≤ 50% RTP. Furthermore, it allows the function to be inoperable under the additional conditions that no circulating water pump breakers are closed, and high condenser pressure. The ITS should retain the CTS requirement.

Response:

CTS 15.3.5-2, item 11a & 11b does allow operation below 50% of rated power, if one channel is inoperable. The CTS also allows unrestricted operation below 50% of rated power with no turbine trip channels operable. This implies the trip functions are only required to be operable at a Thermal Power ≥ 50% RTP. However, the ITS submitted does not reflect this. Therefore, the Mode of Applicability for ITS 3.3.1, Table 3.3.1-1, functions 15.a and 15.b are being revised to Mode 1, above the P-9 interlock. The Required Actions were also changed to be consistent with the Mode of Applicability for these functions.

3.3.1-08 DOC A.8
ITS 3.3.1 Table 3.3.1-1, Functions 11, 12 - Applicable Modes Note (d)
ITS 3.3.1 Required Actions G.2, H.2
CTS 15.3.5, Table 15.3.5-2, Functions 14.a, 14.b - Permissible Bypass Conditions
CTS 15.3.5, Table 15.3.5-2,
Operator Actions if Conditions of Column 3 Cannot be Met

The CTS does not specify that the 4 kV bus Undervoltage and Underfrequency functions may be blocked when power ≤ 10% RTP. Neither does the CTS provide for reducing power below 10% as the allowed action for 2 or more channels inoperable. The proposed ITS limits Applicability for this function to Thermal Power > 10% RTP. Comment: This change should be classified as L. The specific change to the tech specs has not been discussed. Under the current CTS, at least one channel is required to be operable in all Modes (presumably Modes 1 & 2). The proposed ITS allows the entire function to be inoperable if power ≤ 10% RTP. The Required Actions for Conditions G and J require reducing power below 10% rather than transition to Mode 3 which would be consistent with the CTS. The CTS Applicability and Required Actions should be retained in the ITS.

Response:

Per FSAR 14.1.8, the Undervoltage and Underfrequency Bus A01 and A02 reactor trip functions provide protection against core DNB during loss of coolant flow events that result from a loss of RCP's. Per PBNP design, these trips are automatically blocked below P-7 (Natural Circulation flow in the RCS would provide adequate core cooling under these conditions.) Therefore, the applicable Modes for these functions are Mode 1 above the P-7 interlock.

ITS LCO 3.3.1 has been changed to adopt the STS applicability for the Undervoltage and Underfrequency Bus A01 and A02 trips functions, to be consistent with when they are required to be operable. The Required Actions have also been revised to be consistent with the Mode of Applicability for these functions. Justification for these less restrictive changes is presented in DOC L.14.

3.3.1-09 DOC L.4
ITS 3.3.1 Table 3.3.1-1, Function16 - Applicable Modes
ITS 3.3.1 Required Action L.2
CTS 15.3.5, Table 15.3.5-2, Function 15

The CTS does not specify Permissible Bypass Conditions for the Safety Injection input to the RTS. The proposed ITS limits the Applicable Modes for this function to Modes 1 and 2. Comment: The justification for this change is not adequate. The justification provided for L.4 simply restates the proposed change. A technical explanation of why the changes is acceptable has not been provided. The No Significant Hazards Considerations states that the function is not required because the reactor is not critical. Certain elements of the RTS and the ESFAS SI function are, however, required to be operable in Modes 3, 4, and 5. The DOC and NSHC do not discuss why the interface between the RTS and ESFAS is not necessary under these circumstances. Improve the discussion provided in the DOC.

Response:

The RPS provides and processes signals which actuate equipment and systems important to safety. On a LOCA the system depressurizes rapidly to the low pressurizer pressure reactor trip setpoint. The protection system generates a reactor trip signal which results in core shutdown, reducing the heat generation in the core to decay levels required by 10CFR50, Appendix K. Further depressurization to the low pressurizer pressure SI setpoint generates a signal which actuates the SI system. When the SI system is actuated, a trip signal is also sent to RPS. Therefore, the reactor trip on SI actuation provides a backup trip to the low pressurizer pressure reactor trip. This backup reactor trip ensures the reactor is shutdown to meet the condition of acceptability for the LOCA. Therefore, this trip function must be OPERABLE in Mode 1 or 2, when the reactor is critical. In MODE 3, 4, 5 or 6 the reactor is not critical and the trip function does not need to be OPERABLE.

3.3.1-10 DOC A.8, M.21
ITS 3.3.1 Table 3.3.1-1, Functions 10.a, 10.b - Applicable Modes
ITS 3.3.1 Required Actions I.2, J.2
CTS 15.3.5, Table 15.3.5-2, Functions 16.a, 16.b

The CTS does not specify Permissible Bypass Conditions for the RCP Breaker Open Position functions. The proposed ITS limits the Applicable Modes for this function to Mode 1 above 50% RTP for a single loop trip and Mode 1 above 10% but below 50% RTP for a trip in two loops. Comment: This change should be classified as L. As written the CTS requires these functions to be operable under all conditions. This may be an administrative error in the CTS. Nevertheless, justification for the change should be provided. Additionally the statement of the Applicable Modes in the ITS requires the function to be operable below 50% RTP and the other function to be operable above 50% RTP, but does not require either function to be at 50% RTP.

If the Applicable Modes of the ITS are changed to be consistent with the above interpretation of the Applicability in the CTS then the ITS Required Actions I.2 and J.2 are not more restrictive because they allow unlimited operation at reduced power rather than requiring transition to Mode 3. Improve the discussion provided in the DOC and change one of the inequalities to include Mode 1 with Thermal Power = 50% RTP. Revise Required Actions for Conditions I and J as appropriate.

Response:

The applicable mode for the RCP Breaker Position (Single Loop) and RCP Breaker Position (Two Loop) function have been changed to MODE 1, above the P-8 setpoint, and MODE 1, above the P-7 interlock and below the P-8 setpoint, respectively. The Required Actions have also been revised to be consistent with the Mode of Applicability for these trip functions.

3.3.1-11 No DOC
ITS 3.3.1 Table 3.3.1-1, Functions 10.a, 10.b - Required Channels
CTS 15.3.5, Table 15.3.5-2, Functions 16.a, 16.b - Minimum Operable Channels

The CTS specifies the Minimum Operable Channels is 2 (presumably 2/RCP). The ITS proposes that the number of Required Channels be 1/RCP, consistent with the STS but different from the CTS. Comment: No DOC is referenced for this change. A change from 2 to 1 Required Channels should be classified as L. However, as discussed in comment 3.3.1-01 the CTS requirement should be retained as the number of Required Channels be the same as the total number of channels. Change Required Channels to be the total number of channels per RCP. Presumably this is 2/RCP.

Response:

The minimum number of RCP Breaker Position channels specified in CTS Table 15.3.5-2 refers to the number of channels between both RCP's, not per RCP. A RCP Breaker position channel consists of the RCP Breaker auxiliary contact and the associated RCP Loss of Power Trip Matrix relay. Therefore, these are a total of two channels (1 per RCP). Adopting the STS convention of specifying the number of channels "per RCP" does not result in a change in the required number of channels for the RCP Breaker Position trip functions. This change in convention is covered by DOC A.1.

3.3.1-12 JFD 6 (See beyond scope item 10)
ITS 3.3.1 Table 3.3.1-1, Function17.a - Applicable Modes, Note (c)
CTS 15.3.5, Table 15.3.5-2, Function 17.a, Applicability, Note (c)

The STS limits Applicability for the P-6 bypass function to conditions in which the P-6 interlock is not tripped, i.e., below the actual P-6 setpoint. The proposed ITS limits Applicability to when both Intermediate Range channels read less than 10^{-10} Amps. The proposed ITS also allows that the P-6 trip setpoint must be $\geq 10^{-10}$ Amps.

This comment is a placeholder for beyond scope item 10. It remains open pending technical branch disposition. In addition to technical branch comments, respond to the following.

The proposed ITS creates an incongruous situation in which the function is not required to be OPERABLE within the range of its trip setpoint. Change note (c) to the note used in the STS, "Below the P-6 (Intermediate Range Neutron Flux) interlocks."

Response:

The Applicability of the P-6 interlock has been changed to MODE 2, below the P-6 interlock. Below the P-6 setpoint the NI's Source Range Neutron Flux reactor trip function is automatically enabled. Therefore, the P-6 interlock will be required operable consistent with when it is required to perform its safety function.

3.3.1-13 JFD 10
ITS 3.3.1 CONDITION O (Table 3.3.1-1, Function17.a)
CTS 15.3.5, Table 15.3.5-2, Function 17.a, Conditions and Required Actions

Condition O is added to require verification within 1 hour that the P-6 interlock is in the required state, or opening RTBs within two hours if one or more channel(s) are inoperable when in Modes 3, 4, and 5 and rod withdrawal is possible. JFD 10 indicates that the P-6 interlock is credited for automatic action to mitigate uncontrolled rod withdrawal from subcritical conditions.
Comment: Manual verification of interlock state is not an appropriate measure to compensate for a loss of an automatic trip function (both channels inoperable). Furthermore, finding the interlock is not in the required state means that the Source Range trip function has been rendered inoperable by failure of the interlock. If this condition is reached via inoperability of the P-6 interlock, 2 hours is allowed to open RTBs via Required Action O.2. This is inconsistent with CONDITION F which indicates that RTBs must be opened immediately if the Source Range trip is rendered inoperable. Also the ITS Bases do not discuss the function identified in JFD 10. Change CONDITION O to apply to one channel inoperable. Change the Completion Time for Conditions O and F to be consistent. Revise the Bases to discuss the function described in JFD 10. A new CONDITION might be added to address both channels inoperable if so, the Required Actions and Completion times should be equivalent to the ACTIONS required for loss of the

Source Range function. [Note: JFD 10 may be in error when it describes P-6 as being required for protection against rod withdrawal since it is really describing a protective rather than a permissive function. The licensee change may have in mind the Source Range function which would be inoperable if the P-6 interlock failed. If this is the case, it is then not necessary that the P-6 LCO include the MODE 3, 4, and 5 applicability as the function of the Source Range will already be adequately addressed by Function 4 in Table 3.3.1-1.]

Response:

The requirement for the P-6 interlock to be operable in MODES 3, 4, and 5 with RTB's closed and Rod Control System capable of rod withdrawal has been deleted. Furthermore, requiring the interlock be verified in the required state for existing conditions with one or more channels inoperable is consistent with STS, as modified by approved TSTF-135. (Verifying the interlock status manually accomplishes the interlock's function.)

3.3.1-14 JFD 39
ITS 3.3.1 Table 3.3.1-1, Functions17.c, 17.3, Bases
CTS 15.3.5, Table 15.3.5-2, Functions 17.c, 17.e

The P-9 and Turbine Impulse Pressure interlocks are added to the ITS in LCO 3.3.1 and a discussion added to the bases. The P-7 interlock included in the STS is not included in the ITS. Comment: JFD 39 indicates that the Point Beach design does not include a P-7 interlock, but the Bases for the P-9 and Turbine Impulse Pressure interlocks state that they provides inputs to the P-7 interlock. Revise the Bases to be consistent with the plant specific design.

Response:

The Reactor Trip System Interlocks have been reorganized to indicate that the P-10 and Turbine Impulse Pressure interlocks are inputs to the P-7 interlock. The Bases have also been revised to be consistent with Point Beach design.

3.3.1-15 DOC M.24
ITS 3.3.1 SR 3.3.1.2, Note 1

A note has been added requiring that the NIS channel be adjusted if it differs from the heat balance calculation by more than 2%. Comment: A basis for adopting the STS criteria for recalibration has not been provided. Since the existing practice is not described, it is unknown whether the actual practice will be more or less restrictive than current practice regardless of the fact that including the value in the tech specs make the specs themselves more restrictive. Insert a value that is consistent with the assumptions of the plant specific setpoint analysis and justify the change based upon consistency with that analysis.

Response:

Point Beach current practice requires adjustment of NIS channel output, if the absolute difference between the calorimetric heat balance and the NIS channel output exceeds 2% RTP. This daily adjustment supports the accuracy assumptions in establishing the high and low flux trip setpoints. A calorimetric uncertainty of 2% is factored into the setpoint calculation along with other NI channel inaccuracies. Although consistent with current practice, this requirement is not in CTS. Therefore, adopting the ISTS SR 3.3.1.2 Note imposes additional requirements on unit operation and is more restrictive. DOC M.24 has been revised to include this discussion.

3.3.1-16 DOC M.6, M.4, M.7, M.9, A.8, A.7, A.21, A.22
ITS 3.3.1, Table 3.3.1-1, Surveillance Requirements
CTS 15.4.1, Table 15.4.1-1, Functions 1-9, 11, 12, 15, 16, 44, 45

The CTS requires surveillance under all plant conditions. The ITS adopts the STS philosophy of requiring surveillance only in the specified Applicable Modes for each function. Comment: These changes are classified as more restrictive, however, based upon a literal reading of the CTS they are indeed less restrictive. The DOCs do not discuss the reasons why these less restrictive changes are acceptable. Provide technical justification for the changes in the DOC.

Response:

Although CTS Table 15.4.1-1 states for many of the functions that the surveillance's are required in "ALL" plant conditions, CTS 15.4.0.1 states, "Surveillance requirements shall be met during all times that the system or component is required to be operable." Additionally, the CTS defines Operability as when a system, subsystem, train, component, or device is capable of performing its functions as analyzed in the safety analysis report. Therefore, adopting the STS philosophy requiring surveillance only when the respective Function is required to be OPERABLE, is <u>not</u> a less restrictive change. Point Beach interpretation of the CTS does not require surveillance's on functions when they are not required to be operable IAW the assumptions made in the safety analysis.

3.3.1-17 DOC M.27 JFD 6, JFD 16 (beyond scope item 10)
ITS 3.3.1 SR 3.3.1.8, Frequency
CTS 15.4.1, Table 15.4.1-1, Frequency "P"

The CTS requires functional testing of the Source Range prior to reactor criticality if not performed in the previous week. The STS requires surveillance after passing below P-6. The proposed ITS requires surveillance below 10^{-10} Amps for the source range instrumentation and prior to startup if not performed in the previous 92 days.

This comment is a placeholder for beyond scope item 10. It remains open pending technical branch disposition. In addition to technical branch comments, respond to the following.

The ITS has been, quite rightly, changed in the more restrictive direction to require surveillance of the Source Range function during shutdown. However, the STS was written to require surveillance once the interlock is invoked. Since the actual interlock setpoint may be greater than 10^{-10} Amps specifying the surveillance requirement based upon the current reading, rather than the interlock status allows a condition to occur where the interlock has been invoked but has not been confirmed operable. Consequently, the proposed change does not fully implement the intent of the STS nor the good intentions of WEPCO. In addition, the change to require testing within 92 days of criticality vs. within one week of criticality as required by the CTS has not been discussed in a DOC. Change the Frequency requirement to be based upon interlock status rather than current. Further justify changing the test interval prior to criticality from 7 to 92 days.

Response:

The ITS requirements for surveillance testing of the Source Range instrumentation has been changed from "below 10^{-10} amps" to "below P-6", consistent with the STS. Additionally, the change to require testing within 92 days of criticality vs. within one week of criticality is addressed in DOC L.9. Lastly, the ITS requirement for surveillance testing of the Intermediate Range instrumentation has been changed from "below 10% RTP" to "below P-10", consistent with the STS.

3.3.1-18 No DOC
ITS 3.3.1 SR 3.3.1.3
CTS 15.4.1, Table 15..4.1-1, Function 1, Note 4

Note 4 has not been incorporated into the ITS. Comment: Deletion of Note 4 has not been discussed. Include CTS note 4 into the ITS. It may be sufficient to add this information to the bases discussion for SR 3.3.1.3.

Response:

CTS 15.4.1, Table 15.4.1-1, Function 1, Note 4 has been moved to the Bases. These details are not necessary to adequately describe the actual regulatory requirement, and can therefore be moved to the Bases without an impact on safety. The Bases will be controlled by the Bases Control Process in Section 5 of the Proposed ITS. These changes to the submitted are discussed in DOC LA.4 and JFD 69.

3.3.1-19 DOC L.8
ITS 3.3.1 SR 3.3.1.3, Note 2
CTS 15.4.1, Table 15..4.1-1, Frequency "P"

The CTS requires comparison of Power Range axial flux difference with the incore detectors when at power. The proposed ITS requires this comparison below after Thermal Power is 50% RTP or greater. The CTS did not restrict the need to conduct the surveillance based on power level. The justification for the restriction is based upon the inaccuracy of the calorimetric at low power levels. Comment: The limitation of the surveillance requirement has not been justified. The DOC is not germane to the change as SR 3.3.1.3 deals with comparison to incore detectors, not to a calorimetric calculation. Provide an appropriate justification for the change.

Response:

Per CTS 15.4.0.4, the reactor shall not be placed in a condition where a system or component is required to be operable, if the specified surveillance's have not been performed satisfactorily within their specified frequencies. However, CTS 15.4.0.4 also states, if entry into a condition where the system or component is required to be operable is necessary in order to perform the specified surveillance, entry into the operating condition may be made provided prior testing or inspection provides reasonable assurance of operability and the surveillance is performed as soon as practicable following entry into the required operating condition.

Point Beach operating experience has shown that an accurate comparison of incore detector measurements to NIS axial flux difference cannot be made at lower reactor powers. Therefore, adoption of ITS SR 3.3.1.3, Note 2, is consistent with current operating practice and the requirements of CTS 15.4.0.4. However, specifying the surveillance is "required to be performed within 24 hours after THERMAL POWER is ≥ 50% RTP," is more restrictive than the CTS requirement of "as soon as practicable following entry into the required operating condition". This results in a change to the ITS submitted by deleting DOC L.8 and adding DOC M.30.

3.3.1-20 No DOC
ITS 3.3.1 SR 3.3.1.7
CTS 15.4.1, Table 15..4.1-1, Function 8, Note 17

CTS Note 17 is not carried over to the ITS. Comment: No DOC is provided for not including Note 17 in the ITS. Provide an appropriate justification for the change.

Response:

CTS Note 17 simply states all of the functions for which the Steam Generator Level instrumentation is used, and for RPS requires the logic for the low-low level trip to be tested.

The SG level reactor trip is tested as part of the ACTUATION LOGIC TEST surveillance requirement on the Automatic Trip Logic (SR 3.3.1.5).

3.3.1-21 DOC M.19
ITS 3.3.1 SR 3.3.1.14, Frequency
CTS 15.4.1, Table 15..4.1-1, Function 15, Frequency "M"

The CTS requires Monthly functional testing of the Turbine Trip functions. The ITS proposes that the testing only be required prior to exceeding 50% RTP whenever the unit has been in Mode 3, if not performed within previous 31 days. This has been justified as a more restrictive change. Comment: This is a less restrictive change which has not been justified. Modify the ITS to be consistent with the CTS (require TADOT every 31 days.)

Response:

CTS 15.4.1, Table 15.4.1-1, item 15, Reactor Trip signal from Turbine, requires a monthly test of the logic associated with the Turbine Autostop and Turbine Stop Valve functions. These testing requirements comport to ITS SR 3.3.1.5, Actuation Logic Test, at a frequency of 31 days ON A STAGGERED TEST BASIS. This less restrictive change is justified in DOC L.17.

The adoption of the TADOT surveillance requirement (SR 3.3.1.14) on each of these functions is consistent with NUREG-1431, and is discussed in DOC M.19.

3.3.1-22 DOC L.11, JFD 33
ITS 3.3.1 SR 3.3.1.5 Notes, SR 3.3.1.16
CTS 15.4.1, Table 15.4.1-1, Function 44

The CTS requires testing of RPS Actuation System Logic every 31 days on a Staggered Test Basis. The STS contains the same requirement. It is proposed that the ITS include notes to limit the Applicability of this surveillance to certain power levels for a number of functions, and to replace this surveillance with an 18 month surveillance for RCP Breaker Position and Reactor Coolant Flow Low in Two Loops functions (SR 3.3.1.16). Comment: The DOC and the JFD describe the change, but do not provide justification for the change. It is unlikely that the relaxation is needed as performance of the ACTUATION LOGIC TEST typically will not require operability of the associated measurement channels. Thus, testing the logic for these functions should be possible at all power levels. Furthermore, the use of power levels rather than interlock status in the notes would be an issue as discussed in comments 3.3.1-12 and 3.3.1-17. Delete the notes in SR 3.3.1.5 and delete SR 3.3.1.16.

Response:

Further evaluation of the Notes modifying SR 3.3.1.5 and SR 3.3.1.16 (SR 3.3.1.15) has resulted in the following changes:

SR 3.3.1.5, Note 1, only allows a delay in testing for the Source Range Neutron Flux instrumentation. This delay is necessary due to the deletion of the allowance contained in CTS 15.4.0.4 whereby entry into a condition where a system or component is required to be operable is necessary to perform the specified surveillance. Per CTS 15.4.0.4, this allowance is acceptable provided prior testing or inspection provides reasonable assurance of operability and the surveillance is performed as soon as practicable following entry into the required operating condition. The Completion Time of "as soon as practicable" is being modified in the ITS SR 3.3.1.5 Note to 'within 8 hours of entering the required operating condition.' Eight hours will provide a reasonable time in which to perform the specified surveillance.

SR 3.3.1.5, Note 2, excludes the RCP Breaker Position (Two Loops), Reactor Coolant Flow – Low (Two Loops) and Underfrequency Bus A01 and A02 trip functions and the P-6, P-7, P-8, P-9 and P-10 Interlocks from the surveillance requirement. These trip functions / interlocks will be tested via SR 3.3.1.15 at an 18 month interval. CTS Table 15.4.1-1, item #5, Reactor Coolant Flow, requires logic channel testing for loss of reactor coolant flow in both loops, i.e., RCP Breaker Position (Two Loops), Reactor Coolant Flow – Low (Two Loops) and Underfrequency Bus A01 and A02, at each refueling interval (18 months). CTS Table 15.4.1-1, item #45, Reactor Trip System Interlocks, requires testing of the interlocks at each refueling interval (18 months). Point Beach considers these CTS requirements as the equivalent STS requirement for performance of an Actuation Logic Test on the logic associated with these trip functions / interlocks. Therefore to maintain current licensing basis requirements, these trip functions / interlocks should be tested at an 18 month interval.

3.3.1-23 Not used.

3.3.1-24 No DOC
ITS 3.3.1 SR 3.3.1.4, Frequency
CTS 15.4.1, Table 15..4.1-2, Function 24, Frequency

The CTS requires Reactor Trip Breaker testing Monthly. The proposed ITS requires testing Monthly on a Staggered Test Basis. Comment: This change is not discussed in the DOC. Make the ITS consistent with the CTS.

Response:

The frequency of testing for the RTBs has been changed to 31 days on a Staggered Test Basis, consistent with the STS. DOC L.16 has been written to justify the change to the CTS.

3.3.1-25 No DOC, JFD 27, JFD 16
ITS 3.3.1 Table 3.3.1-1. Allowable Value
CTS 15.2.3

The limiting safety system settings (LSSS) in the CTS are expressed at trip setpoints (see CTS bases). The ITS proposes to use Allowable Values instead. Furthermore, the STS format presumes the existence of a plant specific setpoint analysis that sets the context for the form (i.e., allowable values, trip setpoints, or both) in which the LSSS are expressed in the ITS. In the absence of this analysis the expression of the LSSS in the ITS cannot be unambiguously used to determine instrument operability from measurements of component errors. The Point Beach Allowable Values do not appear to be derived from such an analysis as reference to setpoint analysis was deleted from the Bases, and numerical Allowable Values are not provided in the ITS for certain trip functions. Comment: This change is not discussed in a DOC and the plant-specific values do not appear to have been derived using a formal setpoint methodology. Provide justification for using Allowable Values instead of Trip Setpoints. Ensure that the values used are those that were calculated by the plant-specific setpoint analysis. Include a reference to the setpoint analysis in the Bases.

Response:

Per PBNP DG-I01, Instrument Setpoint Methodology, the Limiting Safety System Settings (LSSS) are equivalent to Allowable Values.

3.3.1-26 JFD 12
ITS 3.3.1 Table 3.3.1-1 Function 14. Allowable Value, and Surveillance Requirements, SR 3.3.1.15

The CTS provides no Trip Setpoint the Steam Generator Water Level Low function. The ITS also provides no Allowable Value for this function, contrary to the STS. Furthermore, the CTS requires calibration of the Steam Generator Water Level function, but the ITS specified surveillance is a TADOT instead of a COT and Calibration. Comment: The STS format requires that a Trip Setpoint and / or Allowable Value be provided for the function. Providing no setpoint is functionally equivalent to deleting the function which is unacceptable. Furthermore, the ITS specification of a TADOT is inappropriate for an analog measurement channel. Provide the proper Trip Setpoint and / or Allowable Value consistent with the resolution of comment

3.3.1-25. Replace the requirement for TADOT with a requirement for COT and Calibration (SR 3.3.1.7 and SR 3.3.1.11). Delete SR 3.3.1.15 which is not otherwise used.

Response:

The Steam Generator Water Level Low function is not included in the safety analysis, although a field setting has been established. This field setting was developed outside the scope of the setpoint methodology and can be found in documents provided by the NSSS supplier.

The field setting for the Steam Generator Water Level Low function has been included in Table 3.3.1-1.

3.3.1-27 JFD 16
ITS 3.3.1 Table 3.3.1-1 Functions 15.a, 17.c(2) Allowable Value, and Surveillance Requirements

No Allowable Value is provided for the Turbine Trip - Low Autostop Pressure and P-9 - Condenser High Pressure functions. Comment: The STS format requires that a TRIP SETPOINT and / or Allowable Value be provided for these functions. Since these functions measure an analog parameter, providing no setpoint is functionally equivalent to deleting the function which is unacceptable. Provide the proper Trip Setting and / or Allowable Value consistent with the resolution of comment 3.3.1-25.

Response:

The Turbine Trip –Low Autostop Oil Pressure function is not included in the safety analysis, although a field setting has been established. This field setting was developed outside the scope of the setpoint methodology and can be found in documents provided by the NSSS supplier.

The field setting for the Turbine Trip – Low Autostop Oil Pressure function has been included in Table 3.3.1-1.

The P-9 (Condenser High Pressure) function has been deleted from the submittal, consistent with the CTS.

3.3.1-28 DOC A30
CTS 15.3.1.F.3

The CTS requires confirmation that at least one source range detectors is on scale during approach to criticality. This has not been carried over to the ITS. Comment: The DOC cited as justification does not exist. Provide justification for the change.

Response:

CTS 15.3.1.F.3 is redundant to ITS 3.3.1, Table 3.3.1-1, item #4, and therefore, has not been retained as a separate item in ITS, as justified in DOC A.30.

3.3.1-29 No DOC
ITS 3.3.1, Actions, Note

The CTS does not explicitly allow separate condition entry. The ITS proposes that separate condition entry is allowed for each function. Comment: No DOC has been provided for the change. Provide justification for the change.

Response:

The CTS does not place restrictions on Condition entry. The Note allowing separate condition entry for each function has been adopted, because of the restrictions imposed by STS Specification 1.3. DOC A.32 has been written to provide an explanation for the adoption of the Note.

3.3.1-30 DOC L.1 (beyond scope item 7)
ITS 3.3.1, Action D
CTS 15.3.5, Table 15.3.5-2, New Footnote ##

The CTS does not allow taking an inoperable channel out of the tripped condition to allow surveillance testing of other channels. The ITS allows for this. The ITS modifies the STS provision that the inoperable channel may be placed in bypass.

This comment is a placeholder for beyond scope item 7. It remains open pending technical branch disposition. In addition to technical branch comments, respond to the following.

The provision to allow taking the inoperable channel out of the tripped condition is justified in DOC L.1 based upon the analysis contained in WCAP-10271-P-A, Supplement 2. This is consistent with the basis for the STS. The Safety Evaluation Reports for WCAP-10271 require that applicants for the proposed Technical Specification changes for individual plants must confirm the applicability of the generic analysis of the WCAP. The applicability of the WCAP-10271 analysis to Point Beach has not been discussed. Furthermore, the STS allowance is based upon a design which includes bypass provisions. The bypass function includes interlocks that prevent disabling more than one channel at a time. The basis for accepting the STS note allowing bypass in a design that lacks the STS assumed protective interlocks has not been discussed. This is a technical change that should be the subject of a separate technical evaluation. Delete the note allowing removal of inoperable channels from the tripped condition.

Response:

The ITS has been revised to be consistent with the requirements of the CTS.

Section 3.3 Beyond Scope Issues:

TR-1

Item 7: Clarify the functional difference between "bypass" and "taken out of the tripped condition" with justifications why the "taken out of the tripped condition" will meet the requirements of the "bypass condition. Provide schematic diagrams if required.

Response:

The Notes modifying the Required Actions that allow taking an inoperable channel out of the bypass condition to perform surveillances on other channels, are not being adopted in the ITS. PBNP has not confirmed the generic analysis of WCAP-10271, upon which this allowance is based.

TR-2

Item 13, 14, 15, and 18: Clarify why shutdown mode has been included with specific reasons for adding the three notes to item 18. (TR-2)

Response:

Further evaluation of the Notes modifying SR 3.3.1.5 and SR 3.3.1.16 (SR 3.3.1.15) has resulted in the following changes:

SR 3.3.1.5, Note 1, only allows a delay in testing for the Source Range Neutron Flux instrumentation. This delay is necessary due to the deletion of the allowance contained in CTS 15.4.0.4 whereby entry into a condition where a system or component is required to be operable is necessary to perform the specified surveillance. Per CTS 15.4.0.4, this allowance is acceptable provided prior testing or inspection provides reasonable assurance of operability and the surveillance is performed as soon as practicable following entry into the required operating condition. The Completion Time of "as soon as practicable" is being modified in the ITS SR 3.3.1.5 Note to 'within 8 hours of entering the required operating condition.' Eight hours will provide a reasonable time in which to perform the specified surveillance.

SR 3.3.1.5, Note 2, excludes the RCP Breaker Position (Two Loops), Reactor Coolant Flow – Low (Two Loops) and Underfrequency Bus A01 and A02 trip functions and the P-6, P-7, P-8, P-9 and P-10 Interlocks from the surveillance requirement. These trip functions / interlocks will be tested via SR 3.3.1.15 at an 18 month interval. CTS Table 15.4.1-1, item #5, Reactor Coolant Flow, requires logic channel testing for loss of reactor coolant flow in both loops, i.e., RCP Breaker Position (Two Loops), Reactor Coolant Flow – Low (Two Loops) and Underfrequency Bus A01 and A02, at each refueling interval (18 months). CTS Table 15.4.1-1, item #45, Reactor Trip System Interlocks, requires testing of the interlocks at each refueling interval (18 months). Point Beach considers these CTS requirements as the equivalent STS requirement for performance of an Actuation Logic Test on the logic associated with these trip functions / interlocks. Therefore to maintain current licensing basis requirements, these trip functions / interlocks should be tested at an 18 month interval.

TR-3

Items 20, 21: L12 relaxes Channel Functional test from monthly to 18 months on known reliability of the function and the multichannel redundancy available. Provide justification for this relaxation. M23 specifies that an inoperable channel must be restored in 48 hours. Provide justification for selection of this time period.

Response:

CTS Table 15.4.1-1, item 16, TEST, is a requirement to test the logic portion of the Reactor Trip Signal From SI function (Actuation Logic Test). Requiring a TADOT surveillance of this function is a new requirement. The new TADOT requirement adopts the STS frequency of 18 months. DOC M.32 describes this change.

The Required Actions for an inoperable RTB, RTBB or associated trip in MODES 3, 4 and 5 when the associated breaker is closed and the Rod Control System is capable of rod withdrawal, requires the RTB or RTBB to be restored to an operable status within 48 hours. This Completion Time is more restrictive than the CTS requirements, but is consistent with the STS. The discussion in DOC M.23 has been revised to reflect this justification.

Additional Corrections Required to ITS 3.3.1:

Additional corrections to the conversion package for ITS 3.3.1 have been identified as a result of ITS reviews by plant staff.

1. Per Errata #26, CTS 15.2.3.1.B.8.(b), Reactor Coolant Pump Motor Breaker Open – Low Voltage setpoint, was not dispositioned in the ITS submittal. This setpoint has been shown to be deleted in this supplement to the ITS submittal. DOC L.18 has been written to justify this deletion from the CTS.

2. Per Errata #90, DOC LA.2 has miscategorized the deletion of CTS Table 15.4.1-1, Note (19). DOC 1.19 has been written to replace DOC LA.2, as justification for the deletion of CTS Table 15.4.1-1, Note (19).

3. Per Errata #101, clarification of the description for the Manual Reactor Trip in the Bases is needed. As a result, the Bases description was modified to indicate that the Manual Reactor Trip only trips the Bypass Breaker via the Undervoltage mechanism.

4. Per Errata #110, the Actuation Logic Test surveillance requirements should be revised to reflect the CLB requirement to perform a test of the logic associated with the Reactor Trip System Interlocks each refueling interval (18 months). To address this issue, a note was added to SR 3.3.1.5 to exclude testing of the permissives from the 31 day (On a Staggered Test Basis) requirement. Additionally, a note was added to SR 3.3.1.15 to require performance of the surveillance on the permissives, thereby establishing a testing frequency requirement of 18 months.

5. Per Errata #126, the Bases discussion of the Manual Reactor Trip function should be revised to reflect PBNP design. To address this issue, the Bases have been modified to indicate the Manual Reactor Trip consists of four switches in two channels, with a channel being comprised of one switch in each train.

6. Per Errata #70, the changes made to the CTS by Amendments 193/198 should be incorporated into the ITS submittal. These changes have been made in this supplement to the ITS submittal.

7. Per Errata #145, the Power Range Neutron Flux – Low, Power Range Neutron Flux – High and Intermediate Range Neutron Flux trip functions should not be required to be operable below the minimum temperature for criticality, because of the decalibrating effects on the detectors. The requirements for these instruments to be operable in MODES 3, 4 and 5 with the RTBs closed and the Rod Control System capable of rod withdrawal have been deleted. This change makes the ITS submittal consistent with the CTS and the STS.

8. Per Errata #155, the Bases of ITS SR 3.3.1.7 and SR 3.3.1.9 state the frequencies are justified by the analysis contained in WCAP-10271-P-A, Supplement 2. However, the Safety Evaluation Reports for WCAP-10271 require that individual plants must confirm the applicability of the generic analysis of the WCAP. The applicability of the WCAP-10271 analysis to Point Beach has not been confirmed. Therefore, the statements in the Bases crediting the analysis of WCAP-10271 as justification for the frequencies of SR 3.3.1.7 and SR 3.3.1.9 have not been retained in ITS. The frequencies of SR 3.3.1.7 and SR 3.3.1.9 in the ITS are consistent with the requirements of the CTS.

ITS 5.1, Responsibility

no comments

ITS 5.2, Organization

5.2-01 JFD 6
ITS 5.2.1.c

Now that NMC is the license holder for Point Beach, the word "corporate" is not a potential point of confusion. Comment: Suggest using "specified Nuclear Management Corporation corporate officer" in place of STS' "specified corporate officer".

Response:

The standard terminology, "specified corporate officer", has been adopted; the omitted term "corporate" has been restored into the ITS. JFD 6 has been marked as not used. As discussed with the reviewer, adding the licensee holder name (NMC) is not needed for clarity.

ITS 5.3, Facility Staff Qualifications

5.3-01 JFD 1, DOCs A1 and M1

ITS omits the STS 5.3.1 sentence, "The staff not covered by [Regulatory Guide 1.8] shall meet or exceed the minimum qualifications of [Regulations, Regulatory Guides, or ANSI Standards acceptable to the staff.]" Comment: Confirm that the qualifications of all facility staff are covered by ANSI N18.1 - 1971, as supplemented by RG 1.8, Revision 1, September 1975, or adopt the STS sentence consistent with commitments of the current licensing basis.

Response:

The qualifications of all facility staff are covered by ANSI N18.1 - 1971, as supplemented by RG 1.8, Revision 1, September 1975. These qualification requirements are clarified in ITS 5.2 and ITS 5.3, consistent with the current licensing basis. Therefore, the ITS is acceptable as proposed.

ITS 5.4, Procedures

5.4-01 JFD 1

Comment: Adopt STS 5.4.1.b regarding emergency operating procedures consistent with the staff-accepted response to NUREG 0737 in the current licensing basis.

Response:

STS 5.4.1.b has been adopted as ITS 5.4.1.e. JFD 1 has been revised accordingly.

ITS 5.5, Programs and Manuals

5.5-01 DOC A3, A5
CTS 15.7.8.3
CTS 15.7.8.3.a, b, and c

ITS 5.5.1, ODCM, appears to omit some of the regulations and PBNP GDC, listed in CTS, for (1) the control of radioactive effluents, (2) the control of the release of and processing of waste materials, and (3) the assessment of radioactivity in the environs of PBNP. This omission seems to fit the LB-type DOC rather than an A-type DOC. Comment: Identify all such mis-categorizations in the admin controls section and provide suitable LB-type DOC(s).

Response:

The miscategorizations have been identified and reclassified as either LA-type changes or LB-type changes. Two new DOCs, LA.08 and LB.06 have been provided.

5.5-02 DOC LA7, A6
CTS 15.7.5.A
CTS 15.7.8.7.B.4
JFD 10 and 11
ITS 5.5.11, Explosive Gas Monitoring Program

(1) The CTS procedural and oxygen limit requirements for the on-service gas decay tank will be controlled by 50.59 only if these requirements are placed in the TRM or the FSAR; a plant procedure or the explosive gas monitoring program manual itself is not good enough, unless CTS 7.8.7.B.4 is retained in the program description in ITS 5.5.11. (2) JFD 10 does not fully explain why the omitted words from STS 5.5.12 are not applicable to PBNP ITS 5.5.11. These words, if applicable, ought to be adopted. (3) ITS 5.5.11 omits a hydrogen concentration limit because CTS 15.7.5 only has an oxygen limit. Since the actual oxygen limit is being placed

outside TS, why not also require a hydrogen limit? What is the hydrogen limit? Comment: Revise the submittal to address these issues.

Response:

(1) The requirements of CTS 7.8.7.B.4 and the oxygen limit requirements for the on-service gas decay tank are being maintained in the TRM. TRM 4.11, Explosive Gas Monitoring Program, will specify the oxygen limit and require that all changes regarding explosive gas must be made via the 10 CFR 50.59 process. DOC A.6 has been updated to reflect this. (2) JFD 10 has been revised to more fully explain why the omitted words from STS 5.5.12 are not applicable to PBNP ITS 5.5.11. (3) The current PBNP licensing basis regarding explosive gas concentration limits, as specified in CTS 15.7.5, only provides a limit on oxygen concentration to prevent occurrence of an explosive gas mixture from occurring. Although PBNP monitors both oxygen and hydrogen concentrations in the gas decay tank, there is no administrative or procedural limit on hydrogen concentration. By limiting oxygen concentration to less than or equal to 4% by volume, the gases in the gas decay tank cannot physically achieve an explosive mixture, irrespective of the hydrogen concentration.

5.5-03 CTS 15.3.13, Snubbers; split report
CTS 15.4.2.B.3, IST of snubbers
DOC LB5
ITS 5.5.7

(1) The removal of the reference to snubbers in CTS 15.4.2.B.3 ought to be included with the relocation of the snubber requirements of CTS 15.3.13. (2) JFD 4 explains that IST "components" only include pumps and valves to justify replacing "components" with "pumps and valves." What about snubbers? Comment: Revise submittal accordingly.

Response:

(1) Removal of the reference to snubbers in CTS 15.4.2.B.3 has been included with the relocation of the snubber requirements of CTS 15.3.13. Attachment 6, Application Of Selection Criteria, Appendix A, Justification For Specification Relocation, of the November 15, 1999 submittal has been revised to reflect this. (2) 10 CFR 50.55.a(f) provides the regulatory requirements for an **IST** Program. It specifies that ASME Code Class 1, 2, and 3 ***pumps and valves*** are the only components covered by an IST Program. 10 CFR 50.55.a(g) provides the regulatory requirements for an Inservice Inspection (**ISI**) Program. It specifies that ASME Code Class 1, 2, and 3 ***components*** are covered by the ISI Program. Therefore, the components that the IST Program applies to (i.e., pumps and valves) have been added for clarity. Snubbers will continue to be inspected and tested at Point Beach as required by 10 CFR 50.55.a(g). Additionally, CTS 15.4.13, Shock Suppressors (Snubbers), was deleted from the PBNP Technical Specifications via Amendments 191 (Unit 1) and 196 (Unit 2), issued by the NRC on December 6, 1999. The

accompanying NRC SER stated the following: Test requirements for safety-related shock suppressors (snubbers) are already included under the Point Beach ISI Program. The ISI testing requirements are more comprehensive and in general more conservative than the snubber testing requirements contained in the current TSs. The elimination of the snubber testing requirements is consistent with NUREG-1431. Having snubber testing requirements in the STS would be duplicative of the requirements of 10 CFR 50.55a, which requires ISI testing to be performed in accordance with ASME Code, Section XI, and the applicable addenda.

5.5-04 15.4.4.II
STS 5.5.6 Pre-stressed Concrete Containment Tendon Surveillance Program
JFD 5
DOC LB 2

Omission of tendon surveillance requirements in the form of the program description as in the STS is beyond scope because CTS contain such requirements. Reasons for the omission in DOC LB 2 are insufficient. Comment: Adopt the STS 5.5.6 programmatic requirement.

Response:

The STS 5.5.6 programmatic requirement has been adopted as ITS 5.5.17. Surveillance Requirement SR 3.6.1.2 had already been revised to conform to this program requirement in response to NRC RAI question 3.6.1-7. JFD 5 has been revised to address the renumbering of this specification. DOC LB.2 has been marked as not used and has been replaced by DOC LA.09. CTS 15.4.4 markup page 15.4.4-1 has been replaced by the revised page resulting from Amendments 196 (Unit 1) and 201 (Unit 2) issued by the NRC on June 27, 2000.

<u>ITS 5.5.6 Reporting Requirements</u>

5.6-01 STS/ITS 5.6.1
DOC A3
STS markup Insert 5.0-3

The STS as revised by TSTF-152, R0, indicates that the ITS should explicitly describe the "special maintenance," not just use the term "special maintenance" in the list of examples of work and job functions. Comment: Review the intent of the STS' language and revise the ITS appropriately.

Response:

The STS terminology "(describe maintenance)" has been restored after the term "special maintenance", to require a description of any special maintenance that may be subject to the reporting requirements. This is consistent with the STS terminology as approved for Ginna, Palisades, McGuire, and Braidwood.

5.6-02 not used

5.6-03 DOC A6
CTS 15.7.8.4.A.6

The referenced CTS is being deleted because it duplicates CTS 15.7.8.7, which according to DOC 5.5 LB1 is itself being deleted for duplication (with regulations). Thus, DOC A6 ought also to be an LB-type change. Comment: Revise the change classification of DOC A6.

Response:

This change has been reclassified as an LB-type change. DOC A.6 has been marked as not used and new DOC LB.4 has been written to document this change.

5.6-04 DOC A7
CTS 15.6.9.1.B.2.d

Deletion of the referenced CTS is not administrative. It is less restrictive. Comment: Replace DOC A7 with an L-type DOC.

Response:

DOC A.7 has been replaced with new DOC L.5; a corresponding CTS markup and NSHC has also been provided.

5.6-05 DOC LA1 for CTS 15.6.9.1.C.1
DOC LA2 for CTS 15.6.9.2.B

The DOCs fail to state both the new location and controlling regulatory requirements for the information being removed from TS. Procedures do not have sufficient change control requirements to ensure proper control of the information being removed. Comment: Correct these and all other LA-type DOCs in the entire submittal that have this deficiency.

Response:

DOCs LA.1 and LA.2 have been replaced by DOCs L.06 and L.07 respectively; corresponding CTS markups and NSHCs have also been provided. Other such LA-type DOCs have been addressed in the response to the corresponding RAIs.

5.6-06 DOC M3 & JFD 6 for COLR
DOC M4 & JFD 6 for PTLR
ITS 5.6.4 and 5.6.5

This comment is a placeholder for beyond scope items 97 (COLR) and 37a (PTLR). No response required pending approval of the proposed beyond-scope changes and the addition of these report requirements to the ITS.

Response:

ITS 5.6.5.b contained a bracketed placeholder regarding a listing of analytical methods that would be submitted to the NRC at a later date. That bracketed placeholder has been replaced by a listing of the specific analytical methods required by NUREG-1431. (see "Additional Corrections Required to ITS Section 5.0", below)

ITS 5.7 High Radiation Area

no comments

Additional Corrections Required to ITS Section 5.0:

Additional corrections to the conversion package for ITS Section 5.0 have been identified as a result of ITS reviews by plant staff.

CTS 15.4.4 was changed via Amendments 196 (Unit 1) and 201 (Unit 2), issued by the NRC on June 27, 2000. This change only consisted of a clarification to the requirement and does not affect the proposed ITS. The originally submitted marked up CTS page 15.4.4-1 has been replaced by the marked up amended page.

CTS 15.4.13, Shock Suppressors (Snubbers), was deleted via Amendments 191 (Unit 1) and 196 (Unit 2), issued by the NRC on December 6, 1999. Revised CTS pages are provided to update Attachment 6, Application Of Selection Criteria, Appendix A, Justification For Specification

Relocation, of the November 15, 1999 submittal. Additionally, Attachment 6, Application Of Selection Criteria, Appendix B, Relocated Technical Specification Pages, of the November 15, 1999 submittal has been annotated to reflect these Amendments. The specific pages are listed in Attachment 2 – Discard and Insertion Instructions.

ITS 5.5.13, Technical Specification Bases Control Program, has been changed to incorporate the rule change to 10 CFR 50.59, effective March 13, 2001. This change is consistent with TSTF-364. The term "unreviewed safety question" has been replaced with the appropriate requirement from the new 10 CFR 50.59.

ITS 5.6.5, PTLR, contained a bracketed placeholder in section b regarding a listing of analytical methods that would be submitted to the NRC at a later date. That bracketed placeholder has been replaced by a listing of the specific analytical methods required by NUREG-1431. A revision of Insert 5.0-11, identifying the NRC staff approval document by date, has been provided, along with a revised clean ITS page.

ITS 5.6.5, Monthly Operating Report, has been changed to make the reporting due date (no later than the 15th of each month) consistent with NUREG-1431.

ATTACHMENT 2
DISCARD AND INSERTION INSTRUCTIONS

VOLUME 1	
November 15, 1999 Submittal; ATTACHMENT 6, APPLICATION OF SELECTION CRITERIA, APPENDIX A, JUSTIFICATION FOR SPECIFICATION RELOCATION	
DISCARD	**INSERT**
Page 15 of 17 "13. SHOCK SUPPRESSORS (SNUBBERS)"	Page 15 of 17 "13. SHOCK SUPPRESSORS (SNUBBERS)"
November 15, 1999 Submittal; ATTACHMENT 6, APPLICATION OF SELECTION CRITERIA, APPENDIX B, RELOCATED TECHNICAL SPECIFICATION PAGES	
DISCARD	**INSERT**
CTS pages 15.4.13-1 through 15.4.13-3	CTS page 15.4.13-1 (December 6, 1999)

VOLUME 3	
SECTION 3.3.1	
DISCARD	**INSERT**
DOC pages 1 through 37 of 37	DOC pages 1 through 41 of 41
CTS markup pages 3, 7-10, 14, 15, 17-21, and 23 through 30 of 30	CTS markup pages 3, 7-10, 14, 15, 17-21, and 23 through 30 of 30
JFD pages 1 through 30 of 30	JFD pages 1 through 27 of 27
ISTS markup pages 3.3-1 through 3.3-9, 3.3-12 and 3.3-14 through 3.3-22	ISTS markup pages 3.3-1 through 3.3-9, 3.3-12 and 3.3-14 through 3.3-22
ISTS Inserts	ISTS Inserts (9 pages)
ISTS Bases markup pages B3.3.1-8, B3.3.1-10, B3.3.1-11, B3.3.1-13, B3.3.1-14, B3.3.1-18, B3.3.1-20 through B3.3.1-28, B3.3.1-30 through B3.3.1-35, B3.3.1-38, B3.3.1-39, B3.3.1-41 through B3.3.1-50, B3.3.1-52, B3.3.1-54, B3.3.1-55, B3.3.1-58, and B3.3.1-60	ISTS Bases markup pages B3.3.1-8, B3.3.1-10, B3.3.1-11, B3.3.1-13, B3.3.1-14, B3.3.1-18, B3.3.1-20 through B3.3.1-28, B3.3.1-30 through B3.3.1-35, B3.3.1-38, B3.3.1-39, B3.3.1-41 through B3.3.1-50, B3.3.1-52, B3.3.1-54, B3.3.1-55, B3.3.1-58, and B3.3.1-60
ISTS Bases Inserts	ISTS Bases Inserts (7 pages)

ATTACHMENT 2
DISCARD AND INSERTION INSTRUCTIONS

SECTION 3.3.1 (continued)	
DISCARD	**INSERT**
NSHC pages 1 through 17 of 17	NSHC pages 1 through 19 of 19
ITS pages 3.3.1-1 through 3.3.1-21	ITS pages 3.3.1-1 through 3.3.1-20
ITS Bases pages B 3.3.1-1 through B 3.3.1-49	ITS Bases pages B 3.3.1-1 through B 3.3.1-43
VOLUME 11	
SECTION 5.2	
DISCARD	**INSERT**
JFD page 2 of 2	JFD page 2 of 2
ISTS markup page 5.0-2	ISTS markup page 5.0-2
ITS page 5.2-1	ITS page 5.2-1
SECTION 5.4	
DISCARD	**INSERT**
JFD page 1 of 1	JFD page 1 of 1
ISTS markup page 5.0-6	ISTS markup page 5.0-6
ITS page 5.4-1	ITS page 5.4-1
SECTION 5.5	
DISCARD	**INSERT**
DOC pages 1 through 17 of 17	DOC pages 1 through 18 of 18
CTS markup pages 12 through 15, 32, and 33 of 41	CTS markup pages 12 through 15, 32, and 33 of 41
JFD pages 2, and 4 through 7 of 7	JFD pages 2, and 4 through 7 of 7
ISTS markup pages 5.0-10 and 5.0-16	ISTS markup pages 5.0-10 and 5.0-16
ITS pages 5.5-14 and 5.5-18	ITS pages 5.5-14 and 5.5-18

ATTACHMENT 2
DISCARD AND INSERTION INSTRUCTIONS

SECTION 5.6	
DISCARD	**INSERT**
DOC pages 1 through 10 of 10	DOC pages 1 through 10 of 10
CTS markup pages 2 through 4, and 11 of 16	CTS markup pages 2 through 4, and 11 of 16
JFD page 3 of 3	JFD page 3 of 3
ISTS markup page 5.0-20	ISTS markup page 5.0-20
ISTS Inserts 5.0-03 and 5.0-11	ISTS Inserts 5.0-03 and 5.0-11
NSHC pages 1 through 8 of 8	NSHC pages 1 through 11 of 11
ITS pages 5.6-1 and 5.6-4	ITS pages 5.6-1 and 5.6-4

ENCLOSURE

13. SHOCK SUPPRESSORS (SNUBBERS)

CTS LOCATION(S):

15.3.13
15.4.2.B.3
15.4.13 RAI 5.5-3

Discussion:

The Snubbers prevent unrestrained pipe motion under dynamic loads which may occur during a seismic event, a DBA or Transient. The restraining action of the Snubbers ensures that the initiating event failure does not propagate to other parts of the failed system or to other safety systems. Snubbers also allow normal thermal expansion of piping and nozzles to eliminate excessive thermal stresses during heatup or cooldown.

Comparison to Screening Criteria:

1. Snubbers are 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. Snubbers are not a process variable, design feature, or operating restriction that is an initial condition of a DBA or transient.

3. Snubbers are not part of a primary success path in the mitigation of a DBA or transient.

4. As discussed in Section 4.0 (Appendix A, page A-57) and summarized in Table 1 of WCAP-11618, snubbers in safety related systems were 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. The Point Beach PRA does not model snubbers; therefore, it provides no information to supplement the conclusions of the generic analysis. In addition, the snubber testing is required by 10 CFR 50.55a to be performed in accordance with ASME/ANSI OM Part 4, "Examination and Performance Testing of Nuclear Power Plant Dynamic Restraints."

Conclusion:

Since the screening criteria have not been satisfied, the Snubber LCO may be relocated to other plant controlled documents outside the Technical Specifications. Surveillance testing will be performed in accordance with O&M-4 which is referenced and required by 10 CFR 50.55. As such snubber testing is duplicative of 10 CFR 50.55 and need not be specified within the Technical Specifications. CTS 15.4.13 was deleted via Amendments 191/196, issued by the NRC on December 6, 1999.

15.4.13 SHOCK SUPPRESSORS (SNUBBERS)

DELETED

Description of Changes - NUREG-1431 Section 3.03.01

30-Jan-01

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:**
	15.02.03.01.A.01	LCO 3.03.01 T3.03.01-01 04-01
		LCO 3.03.01 T3.03.01-01 04-02
	15.02.03.01.A.02	LCO 3.03.01 T3.03.01-01 03
	15.02.03.01.A.03	LCO 3.03.01 T3.03.01-01 02B
	15.02.03.01.B.01	LCO 3.03.01 T3.03.01-01 02A
	15.02.03.01.B.02	LCO 3.03.01 T3.03.01-01 07B
	15.02.03.01.B.03	LCO 3.03.01 T3.03.01-01 05
	15.02.03.01.B.04	LCO 3.03.01 T3.03.01-01 05
		LCO 3.03.01 T3.03.01-01 06 NOTE 2
	15.02.03.01.B.05	LCO 3.03.01 T3.03.01-01 05 NOTE 1
		LCO 3.03.01 T3.03.01-01 06
	15.02.03.01.B.06	LCO 3.03.01 T3.03.01-01 11
	15.02.03.01.B.07	LCO 3.03.01 T3.03.01-01 09A
		LCO 3.03.01 T3.03.01-01 09B
	15.02.03.01.B.08.A	LCO 3.03.01 T3.03.01-01 12
	15.02.03.01.C.01	LCO 3.03.01 T3.03.01-01 08
	15.02.03.01.C.02	LCO 3.03.01 T3.03.01-01 13
	15.02.03.01.C.03	LCO 3.03.01 T3.03.01-01 14-01
	15.02.03.01.C.04	LCO 3.03.01 T3.03.01-01 15A
		LCO 3.03.01 T3.03.01-01 15B
	15.02.03.01.C.05	LCO 3.03.01 T3.03.01-01 16
	15.02.03.01.C.06	LCO 3.03.01 T3.03.01-01 01-01
	15.02.03.02.A.2	LCO 3.03.01 T3.03.01-01 17B-02
	15.02.03.02.B	LCO 3.03.01 T3.03.01-01 17C
	15.02.03.02.C	LCO 3.03.01 T3.03.01-01 17E
	15.02.03.02.D	LCO 3.03.01 T3.03.01-01 17A
	15.03.05 T 15.03.05-02	LCO 3.03.01 T3.03.01-01
	15.03.05 T 15.03.05-02 02.A	LCO 3.03.01 COND D
	15.03.05 T 15.03.05-02 02.A **	LCO 3.03.01 COND D RA D.1
	15.03.05 T 15.03.05-02 02.B	LCO 3.03.01 COND D

DOC Number	DOC Text	
	15.03.05 T 15.03.05-02 02.B **	LCO 3.03.01 COND D RA D.1
	15.03.05 T 15.03.05-02 05	LCO 3.03.01 COND D
	15.03.05 T 15.03.05-02 05 **	LCO 3.03.01 COND D RA D.1
	15.03.05 T 15.03.05-02 06	LCO 3.03.01 COND D
	15.03.05 T 15.03.05-02 06 **	LCO 3.03.01 COND D RA D.1
	15.03.05 T 15.03.05-02 07	LCO 3.03.01 COND K
	15.03.05 T 15.03.05-02 07 **	LCO 3.03.01 COND K RA K.1
	15.03.05 T 15.03.05-02 08	LCO 3.03.01 COND D
	15.03.05 T 15.03.05-02 08 **	LCO 3.03.01 COND D RA D.1
	15.03.05 T 15.03.05-02 09	LCO 3.03.01 COND K
	15.03.05 T 15.03.05-02 09 **	LCO 3.03.01 COND K RA K.1
	15.03.05 T 15.03.05-02 10.A **	LCO 3.03.01 COND L RA L.1
	15.03.05 T 15.03.05-02 10.B	LCO 3.03.01 COND K
	15.03.05 T 15.03.05-02 10.B **	LCO 3.03.01 COND K RA K.1
	15.03.05 T 15.03.05-02 11.A	LCO 3.03.01 COND O
	15.03.05 T 15.03.05-02 11.A **	LCO 3.03.01 COND O RA O.1
	15.03.05 T 15.03.05-02 11.B	LCO 3.03.01 COND O
	15.03.05 T 15.03.05-02 11.B **	LCO 3.03.01 COND O RA O.1
	15.03.05 T 15.03.05-02 13	LCO 3.03.01 COND D
	15.03.05 T 15.03.05-02 13 **	LCO 3.03.01 COND D RA D.1
	15.03.05 T 15.03.05-02 14.A **	LCO 3.03.01 COND K RA K.1
	15.03.05 T 15.03.05-02 15	LCO 3.03.01 COND P
	15.03.05 T 15.03.05-02 17 ****	LCO 3.03.01 COND Q RA Q.1 NOTE
	15.03.05 T 15.03.05-02 NOTE **	LCO 3.03.01 COND D RA D.1 LCO 3.03.01 COND K RA K.1 LCO 3.03.01 COND L RA L.1 LCO 3.03.01 COND O RA O.1
	15.03.05 T 15.03.05-02 NOTE ****	LCO 3.03.01 COND Q RA Q.1 NOTE
	15.03.05.B	LCO 3.03.01 COND A LCO 3.03.01 COND A RA A.1
	15.04.01 T 15.04.01-01 01.B	SR 3.03.01.08
	15.04.01 T 15.04.01-01 01.C (4)	LCO 3.03.01 T3.03.01-01 05 SR 3.03.01.03
	15.04.01 T 15.04.01-01 01.C (5)	LCO 3.03.01 T3.03.01-01 05 SR 3.03.01.03 NOTE 1
	15.04.01 T 15.04.01-01 04	LCO 3.03.01 T3.03.01-01 05 LCO 3.03.01 T3.03.01-01 06

DOC Number	DOC Text	
	15.04.01 T 15.04.01-01 04.A	LCO 3.03.01 T3.03.01-01 05 SR 3.03.01.07
	15.04.01 T 15.04.01-01 04.B	LCO 3.03.01 T3.03.01-01 06 SR 3.03.01.07
	15.04.01 T 15.04.01-01 05.A	LCO 3.03.01 T3.03.01-01 09A LCO 3.03.01 T3.03.01-01 09A SR 3.03.01.07
	15.04.01 T 15.04.01-01 05.B	LCO 3.03.01 T3.03.01-01 09B LCO 3.03.01 T3.03.01-01 21-01 SR 3.03.01.15 SR 3.03.01.15 NOTE
	15.04.01 T 15.04.01-01 06	LCO 3.03.01 T3.03.01-01 08 SR 3.03.01.07
	15.04.01 T 15.04.01-01 07	LCO 3.03.01 T3.03.01-01 07A LCO 3.03.01 T3.03.01-01 07A LCO 3.03.01 T3.03.01-01 07B LCO 3.03.01 T3.03.01-01 07B SR 3.03.01.07
	15.04.01 T 15.04.01-01 08	LCO 3.03.01 T3.03.01-01 13 LCO 3.03.01 T3.03.01-01 14-01 SR 3.03.01.07
	15.04.01 T 15.04.01-01 08 (17)	LCO 3.03.01 T3.03.01-01 13
	15.04.01 T 15.04.01-01 09	LCO 3.03.01 T3.03.01-01 14-02 SR 3.03.01.07
	15.04.01 T 15.04.01-01 11.B	LCO 3.03.01 T3.03.01-01 11 SR 3.03.01.09 SR 3.03.01.10
	15.04.01 T 15.04.01-01 12	LCO 3.03.01 T3.03.01-01 12 SR 3.03.01.10
	15.04.01 T 15.04.01-01 45.A	LCO 3.03.01 T3.03.01-01 17A SR 3.03.01.12
	15.04.01 T 15.04.01-01 45.A (24)	SR 3.03.01.11 NOTE
	15.04.01 T 15.04.01-01 45.B	LCO 3.03.01 T3.03.01-01 17C SR 3.03.01.12
	15.04.01 T 15.04.01-01 45.B (24)	SR 3.03.01.11 NOTE
	15.04.01 T 15.04.01-01 45.C	LCO 3.03.01 T3.03.01-01 17D SR 3.03.01.12
	15.04.01 T 15.04.01-01 45.C (24)	SR 3.03.01.11 NOTE

DOC Number	DOC Text	
	15.04.01 T 15.04.01-01 45.D	LCO 3.03.01 T3.03.01-01 17E
		SR 3.03.01.12
	15.04.01 T 15.04.01-01 45.D (24)	SR 3.03.01.11 NOTE
	15.04.01 T 15.04.01-01 45.E	LCO 3.03.01 T3.03.01-01 17B-02
		SR 3.03.01.12
	15.04.01 T 15.04.01-01 NOTE (17)	LCO 3.03.01 T3.03.01-01 13
	15.04.01 T 15.04.01-01 NOTE (23)	SR 3.03.01.05
	15.04.01 T 15.04.01-01 NOTE (24)	SR 3.03.01.11
	15.04.01 T 15.04.01-01 NOTE (5)	SR 3.03.01.03 NOTE 1
	15.04.01 T 15.04.01-02 24 (A)	LCO 3.03.01 T3.03.01-01 18-01
		LCO 3.03.01 T3.03.01-01 19-01
		SR 3.03.01.04
	15.04.01 T 15.04.01-02 25 (A)	LCO 3.03.01 T3.03.01-01 20-01
		LCO 3.03.01 T3.03.01-01 20-02
		SR 3.03.01.04
		SR 3.03.01.04 NOTE
A.02 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 the Point Beach ITS, consistent with the Standard Technical Specifications for Westinghouse Plants, NUREG-1431. The revised Bases are as shown in the Point Beach ITS Bases.	
	CTS:	**ITS:**
	BASES	B 3.03.01

DOC Number	DOC Text	
A.03 Rev. A	CTS Table 15.3.5-2, "Minimum Operable Channels" column is changed to ITS Table 3.3.1-1, "Required Channels" column. Per CTS LCO 15.3.5.c, if the number of operable channels for a particular subsystem is less than that required by the "Minimum Operable Channels" column of Table 15.3.5-2, operation shall be limited according to the requirements of the "Operator Action" column of the same Table. Furthermore, many of the items in Table 15.3.5-2 have a note associated with them in the "Minimum Operable Channels" column, that limits unit operation if the number of operable channels for that subsystem is one less than the "Total Number of Channels" column. Proposed ITS Table 3.3.1-1 combines these requirements by specifying "Required Channels" for each subsystem or trip function. ITS LCO 3.3.1, Condition A directs entry into the applicable Condition referenced in Table 3.3.1-1, when one or more Functions with one or more channels are inoperable. Therefore, instead of providing an operation limiting note applicable to some of the Functions, that refers to the total number of channels for that Function in another column, proposed ITS Table 3.3.1-1 "Required Channels" column provides the number of channels required for each Function to meet the OPERABILITY requirements for that Function, below which Required Actions are taken to mitigate the Conditions.	
	CTS:	**ITS:**
	15.03.05 T 15.03.05-02	LCO 3.03.01 T3.03.01-01
	15.03.05.C	LCO 3.03.01
A.04 Rev. A	CTS Table 15.3.5-2, "Permissible Bypass Conditions" column is changed to "Applicability" column in ITS Table 3.3.1-1. Table 15.3.5-2 provided a listing of conditions where each trip function was allowed to be bypassed. The ITS "Applicability" column specifies the MODES in which the instruments are required OPERABLE. The MODES specified for each Function are based on the safety analyses assumptions made for that Function, or the diverse protection that Function provides.	
	CTS:	**ITS:**
	15.03.05 T 15.03.05-02	LCO 3.03.01 T3.03.01-01
A.05 Rev. D	Not used.	
	CTS:	**ITS:**
	N/A	N/A
A.06 Rev. A	CTS Table 15.3.5-2, Note *, is being deleted. Note * exempts the Intermediate and Source Range Neutron Flux instruments from operator action in the event of one or more inoperable channels, when a block condition exists. This Note is not necessary in the ITS. ITS LCO 3.0.1 states LCOs shall be met during the MODES or other specified condition in the Applicability. When the Intermediate and Source Range Neutron Flux trip functions are blocked, they are not in applicable MODES, and therefore, are not required to be OPERABLE. Therefore no operator actions are required.	
	CTS:	**ITS:**
	15.03.05 T 15.03.05-02 NOTE *	N/A

DOC Number	DOC Text
A.07 Rev. D	The applicability of CTS Table 15.3.5-2, items #2.b, 5, 6, 8, 12, 13 and 15, have been indicated as MODES 1 and 2 in ITS Table 3.3.1-1, items #2.a, 5, 6, 7.b, 13, 14 and 16. This is consistent with the Operator Actions in CTS Table 15.3.5-2 for each of these Functions, in the event the Minimum Operable Channels requirement could not be met, e.g., be in hot shutdown in 8 hours. Therefore this change does not impose additional requirements nor relax any existing requirement, but rather implements the NUREG 1431 concept of MODES to these Functions.

CTS:	ITS:
15.03.05 T 15.03.05-02 02.B	LCO 3.03.01 T3.03.01-01 02A
15.03.05 T 15.03.05-02 05	LCO 3.03.01 T3.03.01-01 05
15.03.05 T 15.03.05-02 06	LCO 3.03.01 T3.03.01-01 06
15.03.05 T 15.03.05-02 08	LCO 3.03.01 T3.03.01-01 07B
15.03.05 T 15.03.05-02 12	LCO 3.03.01 T3.03.01-01 14-01
	LCO 3.03.01 T3.03.01-01 14-02
15.03.05 T 15.03.05-02 13	LCO 3.03.01 T3.03.01-01 13
15.03.05 T 15.03.05-02 15	LCO 3.03.01 T3.03.01-01 16
15.04.01 T 15.04.01-01 01.C	LCO 3.03.01 T3.03.01-01 05
15.04.01 T 15.04.01-01 07	LCO 3.03.01 T3.03.01-01 07A
	LCO 3.03.01 T3.03.01-01 07B
15.04.01 T 15.04.01-01 08	LCO 3.03.01 T3.03.01-01 13
	LCO 3.03.01 T3.03.01-01 14-01
15.04.01 T 15.04.01-01 16	LCO 3.03.01 T3.03.01-01 16

DOC Number	DOC Text	
A.08 Rev. D	CTS Table 15.3.5-2, items #2.a, 3, 11.a, 11.b, 16.a and 16.b have been revised by the addition of applicable MODES to reflect the conditions each function is required to be OPERABLE. Also, the applicabilities of CTS Table 15.4.1-1, items 3.b, 4.a, 4.b, 5.a, 15.a and 15.b have been changed from "ALL" to the specific MODE(S) under which each function is required to be operable. This change does not impose additional requirements nor relax existing requirements, but rather implements the NUREG 1431 concept of MODES to these functions.	
	CTS:	**ITS:**
	15.03.05 T 15.03.05-02 02.A	LCO 3.03.01 T3.03.01-01 02B
	15.03.05 T 15.03.05-02 03	LCO 3.03.01 T3.03.01-01 03
	15.03.05 T 15.03.05-02 11.A	LCO 3.03.01 T3.03.01-01 15A
	15.03.05 T 15.03.05-02 11.B	LCO 3.03.01 T3.03.01-01 15B
	15.03.05 T 15.03.05-02 16.A	LCO 3.03.01 T3.03.01-01 10A
	15.03.05 T 15.03.05-02 16.B	LCO 3.03.01 T3.03.01-01 10B
	15.04.01 T 15.04.01-01 03.B	LCO 3.03.01 T3.03.01-01 04-01
	15.04.01 T 15.04.01-01 04.A	LCO 3.03.01 T3.03.01-01 05
	15.04.01 T 15.04.01-01 04.B	LCO 3.03.01 T3.03.01-01 06
	15.04.01 T 15.04.01-01 05.A	LCO 3.03.01 T3.03.01-01 09A
	15.04.01 T 15.04.01-01 15.A	LCO 3.03.01 T3.03.01-01 15A
	15.04.01 T 15.04.01-01 15.B	LCO 3.03.01 T3.03.01-01 15B
A.09 Rev. A	The Applicability of CTS Table 15.3.5-2, item #17, Reactor Trip Breakers, has been indicated as MODES 1, 2 and MODES 3, 4, 5 with RTBs closed and the Rod Control System capable of rod withdrawal. This change does not impose additional requirements nor relax any existing requirement, but rather implements the NUREG 1431 concept of MODES to this function.	
	CTS:	**ITS:**
	15.03.05 T 15.03.05-02 17	LCO 3.03.01 T3.03.01-01 18-01
		LCO 3.03.01 T3.03.01-01 18-02
	15.04.01 T 15.04.01-02 24 (A)	LCO 3.03.01 T3.03.01-01 18-01
		LCO 3.03.01 T3.03.01-01 18-02
A.10 Rev. A	CTS Table 15.3.5-2, item #15, Safety Injection input to RPS, column 3, Minimum Operable Channels, refers to Table 15.3.5-3. ITS Table 3.3.1-1 requires 2 trains of SI input from ESFAS to be OPERABLE. This is more consistent with the portion of the SI circuit which provides the input to the Reactor Trip System. Inoperability of SI actuation channels is covered by the Conditions and Required Actions of ITS LCO 3.3.2.	
	CTS:	**ITS:**
	15.03.05 T 15.03.05-02 15	LCO 3.03.01 T3.03.01-01 16

DOC Number	DOC Text
A.11 Rev. A	CTS Table 15.4.1-1, Note (1) has been deleted. Note (1) establishes that during periods of refueling shutdown, various surveillances are not required to be performed, but must be performed prior to criticality, if not performed during the previous surveillance period. This Note is no longer required with the adoption of ITS SR 3.0.1 and SR 3.0.4. SR 3.0.1 states surveillance requirements shall be met during the MODES in the applicability for individual LCOs. SR 3.0.4 states entry into a MODE in the applicability of an LCO shall not be made unless the LCO's surveillances have been met within their specified Frequency. Therefore the concept of CTS Table 15.4.1-1, Note (1) has been retained in ITS and its deletion is administrative.

CTS:	**ITS:**
15.04.01 T 15.04.01-01 01.A (1)	LCO 3.03.01 T3.03.01-01 02A
15.04.01 T 15.04.01-01 01.B (1)	N/A
15.04.01 T 15.04.01-01 02.A	LCO 3.03.01 T3.03.01-01 03
15.04.01 T 15.04.01-01 02.A (1)	N/A
15.04.01 T 15.04.01-01 04.A (1)	N/A
15.04.01 T 15.04.01-01 04.B (1)	N/A
15.04.01 T 15.04.01-01 05 (1)	N/A
15.04.01 T 15.04.01-01 05.A (1)	N/A
15.04.01 T 15.04.01-01 06 (1)	N/A
15.04.01 T 15.04.01-01 07 (1)	N/A
15.04.01 T 15.04.01-01 08 (1)	N/A
15.04.01 T 15.04.01-01 09 (1)	N/A
15.04.01 T 15.04.01-01 11.B (1)	N/A
15.04.01 T 15.04.01-01 15.A (1)	N/A
15.04.01 T 15.04.01-01 15.B (1)	N/A
15.04.01 T 15.04.01-01 16 (1)	N/A
15.04.01 T 15.04.01-01 44 (1)	N/A
15.04.01 T 15.04.01-01 NOTE (1)	N/A

DOC Number	DOC Text
A.12 Rev. A	CTS Table 15.4.1-1, item 1, Nuclear Power Range, has been reorganized to group surveillance requirements together based on the high setting or low setting trip functions. Additionally, the daily heat balance is clarified to only apply to the Power Range Neutron Flux - High trip function in MODES 1 and 2.

CTS:	ITS:
15.04.01 T 15.04.01-01 01	LCO 3.03.01 T3.03.01-01 02A
15.04.01 T 15.04.01-01 01.A	LCO 3.03.01 T3.03.01-01 02A
	SR 3.03.01.01
	SR 3.03.01.02
15.04.01 T 15.04.01-01 01.B	LCO 3.03.01 T3.03.01-01 02B
	SR 3.03.01.07

DOC Number	DOC Text
A.13 Rev. A	CTS Table 15.4.1-1, Note (2), specifies that 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. This allowance is retained in ITS LCO 3.3.1, Power Range Neutron Flux - Low, SR 3.3.1.8. The Frequency requirement of SR 3.3.1.8 specifies the surveillance is required prior to reactor startup, when not performed within the previous 92 days. This change does not impose additional restrictions nor relax existing requirements and is therefore administrative.

CTS:	ITS:
15.04.01 T 15.04.01-01 01.B (2)	N/A
15.04.01 T 15.04.01-01 NOTE (2)	N/A

DOC Number	DOC Text
A.14 Rev. A	CTS Table 15.4.1-1, Note (20), is not being retained in ITS. CTS 15.3.10.E contains the requirements for hot channel factors, and these requirements must be met to confirm the hot channel factor limits are being satisfied. These facts are not influenced by the performance of a comparison of incore detector measurements to NIS axial flux difference. Note (20) does not establish or relax any requirement and this detail is not required in ITS to provide adequate protection of the public health and safety.

CTS:	ITS:
15.04.01 T 15.04.01-01 01.C (20)	N/A
15.04.01 T 15.04.01-01 NOTE (20)	N/A

DOC Number	DOC Text
A.15 Rev. A	CTS Table 15.4.1-1, comparison of incore detector measurements to NIS axial flux difference is changed to ITS SR 3.3.1.3. This surveillance verifies the f(delta I) input to the OT deltaT function. As such, this surveillance requirement has been reassigned to the OT deltaT function. Assigning this test to the OT deltaT function is an administrative change that conforms to NUREG 1431.

CTS:	ITS:
15.04.01 T 15.04.01-01 01.C	LCO 3.03.01 T3.03.01-01 05

DOC Number	DOC Text
A.16 Rev. D	CTS Table 15.4.1-1, item #2, Nuclear Intermediate Range, surveillance requirements have been reorganized. Additionally, it is not necessary to state the surveillances are required when the function is not blocked. With the implementation of MODES, this function is not applicable > P-10. Per proposed ITS SR 3.0.1, surveillance requirements shall be met during the MODES specified in the applicability for individual LCOs.

CTS:	ITS:
15.04.01 T 15.04.01-01 02.A	LCO 3.03.01 T3.03.01-01 03
15.04.01 T 15.04.01-01 02.B	N/A

DOC Number	DOC Text
A.17 Rev. D	CTS Table 15.4.1-1, item #3, Nuclear Source Range, surveillance requirements have been reorganized. Additionally, it is not necessary to state associated surveillances are required when the function is not blocked. With the implementation of MODES, this function is not applicable below P-6. Per proposed ITS SR 3.0.1, SRs shall be met during the MODES specified in the Applicability for individual LCOs.

CTS:	ITS:
15.04.01 T 15.04.01-01 03	LCO 3.03.01 T3.03.01-01 04-01
15.04.01 T 15.04.01-01 03.A	LCO 3.03.01 T3.03.01-01 04-01
15.04.01 T 15.04.01-01 03.B	LCO 3.03.01 T3.03.01-01 04-01 SR 3.03.01.01

DOC Number	DOC Text
A.18 Rev. A	CTS Table 15.4.1-1, Note (2), "Tests of 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," is attached to several surveillance requirements in CTS Table 15.4.1-1, where it will have no bearing on their performance. Therefore this allowance is being deleted from CTS Table 15.4.1-1, items #4.a, 4.b, 5.a, 6, 7 and 11.b, OT deltaT, OP deltaT, Reactor Coolant Flow, PZR Water Level, PZR pressure and 4KV Undervoltage Bus A01 &A02, respectively. This change will not result in a relaxation of the requirements, nor will it impose additional requirements, and is therefore administrative.

CTS:	ITS:
15.04.01 T 15.04.01-01 04.A (2)	N/A
15.04.01 T 15.04.01-01 04.B (2)	N/A
15.04.01 T 15.04.01-01 05.A (2)	N/A
15.04.01 T 15.04.01-01 06 (2)	N/A
15.04.01 T 15.04.01-01 07 (2)	N/A
15.04.01 T 15.04.01-01 11.B (2)	N/A

DOC Number	DOC Text	
A.19 Rev. A	CTS Table 15.4.1-1, item #5, Reactor Coolant Flow, surveillance requirements have been reorganized to accommodate the implementation of the NUREG 1431 MODES. This is necessary because the Single Loop and Two Loop functions are required to be OPERABLE in different conditions. This change will not result in a relaxation of the requirements, nor will it impose additional requirements, and is therefore administrative.	
	CTS:	**ITS:**
	15.04.01 T 15.04.01-01 05	LCO 3.03.01 T3.03.01-01 09A
	15.04.01 T 15.04.01-01 05.A	LCO 3.03.01 T3.03.01-01 09A
	15.04.01 T 15.04.01-01 05.B	LCO 3.03.01 T3.03.01-01 09B
A.20 Rev. A	CTS Table 15.4.1-1, Note (22) is not being retained in ITS. Note (22) establishes that during periods of cold and refueling shutdown, surveillances annotated with this Note are not required to be performed, but must be performed prior to criticality, if not performed during the previous surveillance period. This Note is no longer required with the adoption of ITS SR 3.0.1 and SR 3.0.4. SR 3.0.1 states SRs shall be met during the MODES in the applicability for individual LCOs. SR 3.0.4 states entry into a MODE in the applicability of an LCO shall not be made unless the LCOs surveillances have been met within their specified frequency. Therefore the concept of CTS Table 15.4.1-1, Note (22) has been retained in ITS and its deletion is administrative.	
	CTS:	**ITS:**
	15.04.01 T 15.04.01-01 09 (22)	N/A
	15.04.01 T 15.04.01-01 NOTE (22)	N/A
A.21 Rev. A	The applicability of CTS Table 15.4.1-1, item #44, Reactor Protection System Actuation System Logic, has been indicated as MODES 1, 2 and MODES 3, 4, 5 with RTBs closed and the Rod Control System capable of rod withdrawal. This change does not impose additional requirements nor relax any existing requirement, but rather implements the NUREG-1431 concept of MODES to this function and establishes when the associated surveillance requirement is required to be met.	
	CTS:	**ITS:**
	15.04.01 T 15.04.01-01 44	LCO 3.03.01 T3.03.01-01 21-01 LCO 3.03.01 T3.03.01-01 21-02

DOC Number	DOC Text	
A.22 Rev. A	The applicabilities of CTS Table 15.4.1-1, item #45, Reactor Trip System Interlocks, have been indicated consistent with the conditions under which they are required to allow the proper operation of the trip functions with which they interface. This change has not imposed additional requirements, nor has it resulted in the relaxation of current requirements. This change is administrative in that it implements the NUREG-1431 concept of MODES to these reactor trip interlocks.	
	CTS:	**ITS:**
	15.04.01 T 15.04.01-01 45.A	LCO 3.03.01 T3.03.01-01 17A
	15.04.01 T 15.04.01-01 45.B	LCO 3.03.01 T3.03.01-01 17C
	15.04.01 T 15.04.01-01 45.C	LCO 3.03.01 T3.03.01-01 17D
	15.04.01 T 15.04.01-01 45.D	LCO 3.03.01 T3.03.01-01 17E
	15.04.01 T 15.04.01-01 45.E	LCO 3.03.01 T3.03.01-01 17B-02
A.23 Rev. A	CTS Table 15.4.1-1, Note (24) has been deleted from the Channel Calibration surveillance requirement of the 1st Stage Turbine Impulse Pressure Interlock. Note (24) excludes neutron detectors from the Channel Calibration requirement. This exclusion is not necessary, as there are no neutron detectors associated with this interlock. This change does not affect the requirements of this reactor trip interlock and is administrative in nature.	
	CTS:	**ITS:**
	15.04.01 T 15.04.01-01 45.E (24)	N/A
A.24 Rev. A	CTS Table 15.4.1-1, Notations for Plant Conditions, have been deleted. The Applicability requirements for each function are provided in ITS using the NUREG-1431 concept of MODES, as defined in ITS Table 1.1-1. Therefore the terms used in CTS Table 15.4.1-1 are no longer necessary, and are administratively removed.	
	CTS:	**ITS:**
	15.04.01 T 15.04.01-01 ALL	N/A
	15.04.01 T 15.04.01-01 COLD S/D	N/A
	15.04.01 T 15.04.01-01 HOT S/D	N/A
	15.04.01 T 15.04.01-01 PWR - POWER OPER	N/A
	15.04.01 T 15.04.01-01 REF S/D	N/A

DOC Number	DOC Text	
A.25 Rev. A	CTS Table 15.4.1-2, Note (9) is not being retained in ITS. Note (9) establishes that during periods of cold and refueling shutdowns, surveillances annotated with the Note are not required to be performed, but must be performed prior to exceeding 200 F, if it has not been performed during the previous surveillance interval. This Note is no longer required with the adoption of ITS SR 3.0.1 and SR 3.0.4. SR 3.0.1 states surveillance requirements shall be met during the MODES in the applicability for individual LCOs. SR 3.0.4 states entry into a MODE in the applicability of an LCO shall not be made unless the LCOs surveillances have been met within the specified frequency. Therefore the concept of CTS Table 15.4.1-2, Note (9) has been retained in the ITS and its deletion is administrative.	
	CTS:	**ITS:**
	15.04.01 T 15.04.01-02 24 (A)(9)	N/A
		N/A
	15.04.01 T 15.04.01-02 NOTE (9)	N/A
A.26 Rev. A	CTS Table 15.4.1-2, item #25.b, verification of the Reactor Trip Bypass Breaker shunt trip function is not being retained in ITS. There is no automatic shunt trip associated with the Reactor Trip Bypass Breaker and therefore no need to require verification of its OPERABILITY.	
	CTS:	**ITS:**
	15.04.01 T 15.04.01-02 25 (B)	N/A
A.27 Rev. D	Not Used	
	CTS:	**ITS:**
	N/A	N/A
A.28 Rev. D	Not Used	
	CTS:	**ITS:**
	N/A	N/A
A.29 Rev. A	The Channel Calibration surveillance requirements for CTS Table 15.4.1-1, items #1, 2, 3, Power Range Neutron Flux - High, Power Range Neutron Flux - Low, Intermediate Range Neutron Flux and Source Range Neutron Flux trip instrumentation have been modified by the addition of a Note which excludes the neutron detectors from the calibration. Point Beach currently interprets the Channel Calibration requirement of Table 15.4.1-1, items 1, 2 and 3 to not include the neutron detectors. Therefore adding a Note to exclude the neutron detectors from the Channel Calibration surveillance requirement is an administrative change.	
	CTS:	**ITS:**
	15.04.01 T 15.04.01-01 01	SR 3.03.01.11 NOTE
	15.04.01 T 15.04.01-01 02	SR 3.03.01.11 NOTE
	15.04.01 T 15.04.01-01 03.B	SR 3.03.01.11 NOTE

DOC Number	DOC Text	
A.30 Rev. D	CTS 15.3.1.F.3 requires at least 1 cps, attributable to neutrons, to register on a narrow range source range nuclear instrument during an approach to criticality. The CTS Bases state this requirement ensures the source range instrumentation is functioning properly. ITS LCO 3.3.1, Table 3.3.1-1, item #4, requires the source range neutron flux instrumentation be operable in MODE 2 (below P-6) and in MODES 3, 4 and 5 (with the RTBs closed and the Rod Control System capable of rod withdrawal). To verify this operability, a Channel Check and COT are required to be met. Therefore the requirement of CTS 15.3.1.F.3 is unnecessary and redundant, and is therefore not retained in ITS.	
	CTS:	**ITS:**
	15.03.01.F.03	N/A
A.31 Rev. D	CTS 15.2.3.2.A states the "at power" reactor trips (low pressurizer pressure, high pressurizer level, and low reactor coolant flow for both loops) shall be unblocked when: (1) Power range nuclear flux greater than or equal to 9% (+/- 1%) of rated power, or (2) Turbine load greater than or equal to 10% of full load turbine pressure. The proposed P-7 setpoint of Power Range Neutron Flux (P-10) is greater than or equal to 8% RTP and < 10% RTP and Turbine Impulse Pressure < 10% turbine power ensures the "at power" reactor trips (listed above) will be unblocked as specified in CTS 15.2.3.2.A. Furthermore, specifying applicability of MODE 1, above the P-7 interlock, for the low pressurizer pressure and high pressurizer level, and MODE 1, above P-7 and below P-8 interlocks, for the reactor coolant flow (two loops) ensures their trip Functions will be available to perform their safety Functions as assumed in the accident analysis.	
	CTS:	**ITS:**
	15.02.03.02.A.1	LCO 3.03.01 T3.03.01-01 17B-01
	15.03.05 T 15.03.05-02 07	LCO 3.03.01 T3.03.01-01 07A
	15.03.05 T 15.03.05-02 09	LCO 3.03.01 T3.03.01-01 08
	15.03.05 T 15.03.05-02 10.B	LCO 3.03.01 T3.03.01-01 09B
	15.04.01 T 15.04.01-01 05.B	LCO 3.03.01 T3.03.01-01 09B
	15.04.01 T 15.04.01-01 06	LCO 3.03.01 T3.03.01-01 08
A.32 Rev. D	CTS 15.3.5 has been modified by the adoption of a Note allowing separate Condition entry for each Function. This Note is necessary because of the adoption of ITS Specification 1.3, which states, "Once a Condition as been entered, subsequent trains, subsystem, components, or variables expressed in the Condition discovered to be inoperable or not within limits, will not result in separate entry into the Condition, unless specifically stated." This restriction on Condition entry does not exist in the CTS, therefore, it is necessary to adopt the Note allowing separate Condition entry for each Function.	
	CTS:	**ITS:**
	15.03.05	LCO 3.03.01 COND NOTE

Description of Changes - NUREG-1431 Section 3.03.01

30-Jan-01

DOC Number	DOC Text	
L.01 Rev. D	Not used.	
	CTS:	**ITS:**
	N/A	N/A
L.02 Rev. D	Not used.	
	CTS:	**ITS:**
	N/A	N/A
L.03 Rev. A	CTS Table 15.3.5-2, item #15, Safety Injection, is modified by the addition of a Note that allows one train to be bypassed for up to 8 hours for surveillance testing, provided the other train is OPERABLE. Adopting this Note results in a relaxation of the LCO requirement, but is consistent with the allowance to bypass a RTB for 8 hours, if the other RTB is OPERABLE.	
	CTS:	**ITS:**
	NEW	LCO 3.03.01 COND P RA P.1 NOTE
L.04 Rev. D	CTS Table 15.3.5-2, Note *** has not been retained in ITS. This Note requires the unit to be in cold shutdown within 48 hours, if the minimum Conditions for SI input to the Reactor Trip System are not met within 24 hours after reaching hot shutdown. The RPS provides and processes signals which actuate equipment and systems important to safety. On a LOCA the system depressurizes rapidly to the low pressurizer pressure reactor trip setpoint. The protection system guarantees a reactor trip signal which results in core shutdown, reducing the heat generation in the core to decay levels required by 10CFR50, Appendix K. Further depressurization to the low pressurizer pressure SI setpoint generates a signal which actuates the SI system. When the SI system is actuated, a trip signal is also sent to RPS. Therefore, the reactor trip on SI actuation provides a backup trip to the low pressurizer pressure reactor trip. This backup reactor trip ensures the reactor is shutdown to meet the Condition of acceptability for the LOCA. Therefore, this trip Function must be OPERABLE in MODE 1 or 2, when the reactor is critical. In MODES 3, 4, 5 or 6 the reactor is not critical, and the trip Function does not need to be OPERABLE. Deleting this Note results in a relaxation of the requirements, but does not impact the safety of the plant.	
	CTS:	**ITS:**
	15.03.05 T 15.03.05-02 15 ***	N/A
	15.03.05 T 15.03.05-02 NOTE ***	N/A

DOC Number	DOC Text	
L.05 Rev. D	The Operator Actions of CTS Table 15.3.5-2, item 15, Safety Injection input from ESFAS, have been revised. If the Minimum Operable Channels requirement of CTS Table 15.3.5-2, column 3, cannot be met, the unit is required to be in hot shutdown in 8 hours. ITS LCO 3.3.1, Condition P requires the restoration of the inoperable train in 6 hours or be in MODE 3 in the next 6 hours. This results in a relaxation of the current requirements. This less restrictive change is acceptable considering that with one train operable, the safety function can be performed, and given the low probability of an event occurring in this interval.	
	CTS:	**ITS:**
	15.03.05 T 15.03.05-02 15	LCO 3.03.01 COND P LCO 3.03.01 COND P RA P.1 LCO 3.03.01 COND P RA P.2
L.06 Rev. A	CTS Table 15.4.1-1, item #1, Daily Heat Balance of the Nuclear Power Range instrumentation has been modified by a Note clarifying this surveillance is not required to be performed until 12 hours after THERMAL POWER is greater than or equal to 15% RTP. This is a relaxation of the requirement and is less restrictive. This is acceptable because at lower power levels calorimetric data is inaccurate. Therefore the change provides an allowance to delay performance of the SR until conditions necessary to perform the SR are established while ensuring the SR is performed at the earliest reasonable opportunity.	
	CTS:	**ITS:**
	NEW	SR 3.03.01.02 NOTE 2
L.07 Rev. A	CTS Table 15.4.1-1, monthly surveillance requirement to compare results of incore detector measurements to NIS axial flux difference has been changed to a frequency of 31 EFPD. This results in a relaxation of the CTS frequency, but is acceptable, based on unit operating experience, considering instrument reliability and operating history data for instrument drift. Additionally, the slow changes in neutron flux during the fuel cycle can be detected during this interval.	
	CTS:	**ITS:**
	15.04.01 T 15.04.01-01 01.C	SR 3.03.01.03
L.08 Rev. D	Not used.	
	CTS:	**ITS:**
	N/A	N/A

DOC Number	DOC Text	
L.09 Rev. A	CTS Table 15.4.1-1, Frequency P, Prior to reactor criticality, if not performed during the previous week," is revised into the proposed ITR SR 3.3.1.8, "Prior to reactor startup." This frequency is modified by a Note stating, "Only required when not performed within previous 92 days." This results in a relaxation of the current requirement, but is acceptable because the 92 days is consistent with the frequency of performance for this type of surveillance on similar instrumentation and will not result in an impact to the safety of the unit.	
	CTS:	**ITS:**
	15.04.01 T 15.04.01-01 02.B	SR 3.03.01.08
	15.04.01 T 15.04.01-01 03.B	SR 3.03.01.08
L.10 Rev. A	CTS Table 15.4.1-1, item #4, Reactor Coolant Temperature (OP deltaT and OT deltaT), Plant Conditions When Required, has been changed from "PWR, HOT S/D, COLD S/D", to "MODES 1, 2", in ITS Table 3.3.1-1, Functions 5 and 6. This results in a relaxation of the current requirements, but is acceptable because in MODES 3, 4, 5 and 6 the OT deltaT Function is not required to be OPERABLE due to insufficient heat production to be concerned about DNB. The OP deltaT Function is not required to be OPERABLE in MODES 3, 4, 5 and 6, because the reactor is not operating and there is insufficient heat production to be concerned about fuel overheating and fuel damage.	
	CTS:	**ITS:**
	15.04.01 T 15.04.01-01 04	LCO 3.03.01 T3.03.01-01 05
		LCO 3.03.01 T3.03.01-01 06

DOC Number	DOC Text	
L.11 Rev. D	The Actuation Logic Test requirement of CTS Table 15.4.1-1, item #44, has been modified by the addition of 2 notes. Note 1 allows a delay in the performance of this surveillance for the Source Range Neutron Flux Trip Function for up to 8 hours after power is reduced below P-6. Note 2 allows an exception to the performance of this surveillance for the RCP Breaker Position (Two Loops), Reactor Coolant Flow – Low (Two Loops) and Underfrequency Bus A01 and A02 Trip Functions and the P-6, P-7, P-8, P-9 and P-10 Interlocks. The addition of Note 1 is necessary due to the deletion of the allowance contained in CTS 15.4.0.4 whereby entry into a condition where a system or component is required to be operable is necessary to perform the specified surveillance. This allowance is acceptable provided prior testing or inspection provides reasonable assurance of operability and the surveillance is performed as soon as practicable following entry into the required operating condition. The Completion Time of "as soon as practicable" is being modified to 'within 8 hours of entering the required operating condition.' Eight hours will provide a reasonable time in which to perform the specified surveillance. (See Section 3.0, DOC M.3) Note 2 has been added to exclude the testing of the actuation logic associated with two loop loss of reactor coolant flow trips and reactor trip system interlocks. CTS Table 15.4.1-1, item #5, requires logic channel testing of the RCP Breaker Position (Two Loops), Reactor Coolant Flow – Low (Two Loops) and Underfrequency Bus A01 and A02 trip functions to be performed each refueling interval (18 months). CTS Table 15.4.1-1, item #45, requires logic testing of the reactor trip system interlocks to be performed each refueling interval (18 months). Furthermore, the design of this portion of the RPS actuation logic does not provide the capability to test it at power. Therefore, it is necessary to exclude this portion of the logic from the testing requirements of SR 3.3.1.5, and provide SR 3.3.1.15 to test this logic at a frequency of 18 months. **CTS:** NEW	 **ITS:** SR 3.03.01.05 NOTE 1 SR 3.03.01.05 NOTE 2 SR 3.03.01.15
L.12 Rev. D	Not used. **CTS:** N/A	 **ITS:** N/A

DOC Number	DOC Text
L.13 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.

CTS:	ITS:
15.04.01 T 15.04.01-01 01.A	SR 3.03.01.01
15.04.01 T 15.04.01-01 02.A	SR 3.03.01.01
15.04.01 T 15.04.01-01 03.B	SR 3.03.01.01
15.04.01 T 15.04.01-01 04	SR 3.03.01.01
15.04.01 T 15.04.01-01 06	SR 3.03.01.01
15.04.01 T 15.04.01-01 07	SR 3.03.01.01
15.04.01 T 15.04.01-01 08	SR 3.03.01.01
15.04.01 T 15.04.01-01 09	SR 3.03.01.01

DOC Number	DOC Text
L.14 Rev. D	CTS 15.3.5, Table 15.3.5-2, Function 14.a, Undervoltage Bus A01 and A02, and Function 14.b, Underfrequency Bus A01 and A02, each require one channel per bus to be operable. If either of these requirements cannot be met, operator action is required to place the unit in hot shutdown in 8 hours. With the absence of definitive plant conditions in the CTS under which each of these Functions is required to be operable, it is assumed that the above required actions place the unit in a condition whereby the functions are no longer required to be operable. The proposed applicability for these functions in ITS is MODE 1, above the P-7 interlock. This change is less restrictive, but is acceptable. The accident analyses state the Undervoltage Bus A01 and A02 and the Underfrequency Bus A01 and A02 reactor trip Functions provide protection against core DNB during loss of coolant flow events that result from a loss of RCPs. Per PBNP design, these trips are automatically blocked below P-7. Below the P-7 interlock, natural circulation in the RCS can provide adequate core cooling, such that a reactor trip is not required. Therefore the Undervoltage Bus A01 and A02 and the Underfrequency Bus A01 and A02 are only required to provide a safety function in MODE 1 above the P-7 interlock. This change also revises the default required actions for the Undervoltage Bus A01 and A02 trip function to require reducing Thermal Power to < P-7, consistent with when the trip function is required to be operable.

CTS:	**ITS:**
15.03.05 T 15.03.05-02 14.A	LCO 3.03.01 COND K
	LCO 3.03.01 COND K RA K.2
	LCO 3.03.01 T3.03.01-01 11
15.03.05 T 15.03.05-02 14.B	LCO 3.03.01 COND E
	LCO 3.03.01 COND E RA E.2
	LCO 3.03.01 T3.03.01-01 12
15.04.01 T 15.04.01-01 11.B	LCO 3.03.01 T3.03.01-01 11
15.04.01 T 15.04.01-01 12	LCO 3.03.01 T3.03.01-01 12

DOC Number	DOC Text
L.15 Rev. D	CTS Table 15.4.1-1, Function 15.A and 15.B, monthly functional test requirement is required to be performed prior to reactor criticality, if it has not been performed during the previous month. Proposed ITS SR 3.3.1.14 requires a TADOT to be performed on the Turbine Trip Function prior to exceeding the P-9 interlock whenever the unit has been in MODE 3, if not performed within previous 31 days. This change is less restrictive, but is acceptable, because the frequency is consistent with the applicability for these functions. The LCO requires three channels of Turbine Trip-Low Autostop Oil Pressure and two channels of Turbine Trip – Turbine Stop Valve Closure to be operable in MODE 1, above the P-9 interlock. Testing in MODE 1 prior to 50% RTP ensures these functions will be operable when required.

CTS:	**ITS:**
NEW	LCO 3.03.01 T3.03.01-01 15A
	LCO 3.03.01 T3.03.01-01 15B
	SR 3.03.01.14

DOC Number	DOC Text	
L.16 Rev. D	CTS 15.4.1, Table 15.4.1-2, item 24, Reactor Trip Breakers are required to be tested monthly. Proposed ITS 3.3.1, Table 3.3.1-1, Function 18, Reactor Trip Breakers, requires the RTBs to be tested every 31 days on a staggered test basis. This change results in a relaxation of the current requirements and is less restrictive. This change is acceptable based on operating experience and the reliability of the RTBs. This change makes the testing frequency of the RTBs consistent with STS.	
	CTS:	**ITS:**
	15.04.01 T 15.04.01-01 24	SR 3.03.01.04
L.17 Rev. D	CTS 15.4.1, Table 15.4.1-1, items 15.a, 15.b and 16 require a test of the logic associated with each of these Functions on a monthly basis. Proposed ITS 3.3.1, Table 3.3.1-1, Function 21, Automatic Trip Logic, requires an Actuation Logic Test on the logic associated with Functions 15.a, 15.b and 16 every 31 days on a staggered test basis. This change is consistent with NUREG-1431 and is acceptable based on industry experience, considering instrument reliability and operating history data.	
	CTS:	**ITS:**
	15.04.01 T 15.04.01-01 15.A	SR 3.03.01.05
	15.04.01 T 15.04.01-01 15.B	SR 3.03.01.05
	15.04.01 T 15.04.01-01 16	SR 3.03.01.05
L.18 Rev. D	CTS 15.2.3.1.B.8 (b) provides a setpoint for the low voltage trip of a RCP motor breaker. This setpoint will not be retained in ITS. This change is less restrictive, but is acceptable. The low voltage trip of an RCP Breaker does not cause a direct reactor trip, although an undervoltage condition on the RCP Buses (A01 and A02) does provide a reactor trip. However, the Bus A01 and A02 undervoltage reactor trip uses a different set of undervoltage relays than the RCP Breaker trip on low voltage. Furthermore, Point Beach accident analysis does not credit the indirect trip of the RCP Breakers on low voltage for the mitigation of a loss of flow event. Accordingly, this setpoint may be deleted from the technical specifications as it is not required to provide adequate protection of public health and safety.	
	CTS:	**ITS:**
	15.02.03.01.B.08.B	N/A
L.19 Rev. D	CTS Table 15.4.1-1, Note (19) is not being retained in ITS. This Note provides a description of a heat balance in relation to NIS. These details do not establish a regulatory requirement, but rather provide a description of a surveillance requirement, which is not required to provide adequate protection of public health and safety.	
	CTS:	**ITS:**
	15.04.01 T 15.04.01-01 01.A (19)	N/A
	15.04.01 T 15.04.01-01 NOTE (19)	N/A

DOC Number	DOC Text	
LA.01 Rev. A	The information contained in CTS Table 15.3.5-2, "Total No. of Channels" column and the "No. of Channels to Trip" column is deleted. This information provides details of design or process which are not directly pertinent to the actual requirement. Since these details are not necessary to adequately describe the actual regulatory requirement, they can be moved to license controlled documents without an impact on safety.	
	CTS:	**ITS:**
	15.03.05 T 15.03.05-02	N/A
LA.02 Rev. D	Not used.	
	CTS:	**ITS:**
	N/A	N/A
LA.03 Rev. A	CTS Table 15.4.1-1 contains an instrument surveillance requirement for Residual Heat Removal (RHR) Pump Flow. These instruments do not necessarily relate directly to OPERABILITY of the associated system or the ability to maintain the affected parameter within limits. In general, the Standard Technical Specifications, NUREG-1431, do not require "indication only instruments" to be OPERABLE to support OPERABILITY of a system or component. Control of the availability and necessary compensatory activities for indication instruments are addressed by plant procedures and policies. Therefore RHR Pump Flow instrument channel surveillances are not required to be in the ITS to provide adequate protection of the public health and safety, and are therefore moved to licensee controlled documents. This approach provides an effective level of regulatory control and provides a more appropriate change control process.	
	CTS:	**ITS:**
	15.04.01 T 15.04.01-01 39	TRM 3.03.01 T 3.03.01-01 03
LA.04 Rev. D	CTS 15.4.1, Table 15.4.1-1, Function 1, Note 4 states the requirement to compare results of the incore detector measurements to NIS axial flux difference is performed by means of the moveable incore detector system. These details are not necessary to adequately describe the actual regulatory requirement, and can therefore be moved to the Bases without an impact on safety. The Bases will be controlled by the Bases Control Process in Section 5 of the proposed ITS.	
	CTS:	**ITS:**
	15.04.01 T 15.04.01-01 NOTE (4)	B 3.03.01

DOC Number	DOC Text	
LA.05 Rev. D	The specific numerical values for the OP delta T and OT delta T setpoints in the CTS are being relocated to the Core Operating Limits Report (COLR). This is consistent with Approved TSTF-339, rev. 1, which relocated these values 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 proposed ITS retains the specific core safety limits in the form of the maximum fuel centerline temperature and the DNB limits. These are the limits that form the basis for the core safety limit curves and the specific values for parameters in the OP delta T and OT delta T setpoints. Thus, the technical specifications maintain the appropriate controls over the core safety limits making the relocation of the specific parameter values to the COLR, where they will be maintained in accordance with appropriate regulatory requirements with no reduction in the level of safety. This change is less restrictive, since the numerical values are being relocated out of the Technical Specifications and into the COLR, which is under licensee control.	
	CTS:	**ITS:**
	15.02.03.01.B.04	COLR
		LCO 3.03.01 T3.03.01-01 05
	15.02.03.01.B.05	COLR
		LCO 3.03.01 T3.03.01-01 06
M.01 Rev. A	The applicability of CTS Table 15.3.5-2, item #1, Manual Reactor Trip, has been changed to include MODES 3, 4 and 5 with the RTBs closed and the Rod Control System capable of rod withdrawal. In this condition, inadvertent control rod withdrawal is possible. This change represents an additional restriction on plant operation to assure reactor trip capability exists when control rods may be withdrawn.	
	CTS:	**ITS:**
	15.03.05 T 15.03.05-02 01	LCO 3.03.01 T3.03.01-01 01-02
	15.04.01 T 15.04.01-02 24 (B)	LCO 3.03.01 T3.03.01-01 01-01
		LCO 3.03.01 T3.03.01-01 01-02
	15.04.01 T 15.04.01-02 25 (C)	LCO 3.03.01 T3.03.01-01 01-01
		LCO 3.03.01 T3.03.01-01 01-02

DOC Number	DOC Text	
M.02 Rev. D	The Actions for an inoperable Manual Reactor Trip channel have been revised. Because CTS only requires one of the two channels to be operable when the channel becomes inoperable the unit is required to be in hot shutdown in 8 hours. ITS requires 2 channels of Manual Reactor Trip to be operable. The Required Actions for one inoperable channel in MODES 1 and 2 have been adopted from NUREG-1431, requiring restoration of the channel to an operable status in 48 hours or be in MODE 3 in 54 hours. In MODES 3,4 and 5 with the RTBs closed and the Rod Control System capable if rod withdrawal, an inoperable channel is required to be restored to an operable status in 48 hours or the RTBs shall be opened in 49 hours. If two channels of Manual Reactor Trip function are inoperable in any of the above conditions, LCO 3.0.3 shall be entered, requiring the unit to be in MODE 3 in 7 hours. Therefore, adopting the Required Actions of NUREG-1431 imposes additional requirements on unit operation and is more restrictive.	
	CTS:	**ITS:**
	15.03.05 T 15.03.05-02 01	LCO 3.03.03
	NEW	LCO 3.03.01 COND B
		LCO 3.03.01 COND B RA B.1
		LCO 3.03.01 COND B RA B.2
		LCO 3.03.01 COND C
		LCO 3.03.01 COND C RA C.1
		LCO 3.03.01 COND C RA C.2
M.03 Rev. A	CTS Table 15.4.1-1 has been modified by the adoption of RCP Breaker Position (Single Loop) and RCP Breaker Position (Two Loop) trip functions and the associated 18 month Trip Actuating Device Operational Test (TADOT) to verify their OPERABILITY. These functions anticipate the Reactor Coolant Flow - Low trip to avoid RCS heatup that would occur before the low flow trip actuates. These functions measure only the discrete position of the RCP breaker, using a position switch. The functions have no adjustable trip setpoint with which to associate an Allowable Value, and are therefore verified OPERABLE by the performance of a TADOT every 18 months. This change imposes additional requirements and is therefore more restrictive.	
	CTS:	**ITS:**
	NEW	LCO 3.03.01 T3.03.01-01 10A
		LCO 3.03.01 T3.03.01-01 10B
		SR 3.03.01.13
M.04 Rev. D	Not used.	
	CTS:	**ITS:**
	N/A	N/A

DOC Number	DOC Text	
M.05 Rev. D	The Operator Actions of CTS Table 15.3.5-2, for items #2.a and #2.b, Nuclear Power Range - Low and Nuclear Power Range - High in MODES 1 and 2 require an inoperable channel to be placed in the tripped condition within 1 hour or be in hot shutdown in 8 hours. Proposed ITS LCO 3.3.1, Condition D, requires an inoperable channel be placed in the tripped condition in 1 hour or be in MODE 3 in the next 6 hours. This results in additional restrictions on unit operation, but is a reasonable amount of time, based on operating experience, to place the unit in MODE 3 from full power in an orderly manner without challenging unit systems.	
	CTS:	**ITS:**
	15.03.05 T 15.03.05-02 02.A	LCO 3.03.01 COND D RA D.2
	15.03.05 T 15.03.05-02 02.B	LCO 3.03.01 COND D RA D.2
M.06 Rev. A	Not used.	
	CTS:	**ITS:**
	N/A	N/A
M.07 Rev. D	Not used.	
	CTS:	**ITS:**
	N/A	N/A
M.08 Rev. D	The actions for an inoperable Neutron Flux Intermediate Range trip channel have been revised. Because CTS only requires one of the two channels to be operable, when the required channel becomes inoperable, the unit is required to be in hot shutdown in 8 hours. However, per CTS 15.3.0.c, if the requirements of the LCO are met or are no longer applicable, I.e., < P-6 or > P-10, completion of the specified actions is not required. ITS requires 2 channels of Intermediate Range Neutron Flux to be operable. The Required Actions for one operable channel have been adopted from NUREG-1431, requiring THERMAL POWER either be reduced below P-6 or increased above P-10 in 24 hours. If two channels are inoperable, immediate action is required to suspend operations involving positive reactivity additions and reduce THERMAL POWER below P-6 in 2 hours. Therefore, adopting the Required Actions of NUREG-1431 imposes additional requirements on unit operation and is more restrictive.	
	CTS:	**ITS:**
	15.03.05 T 15.03.05-02 03	LCO 3.03.01 COND G
		LCO 3.03.01 COND G RA G.2
	NEW	LCO 3.03.01 COND F
		LCO 3.03.01 COND F RA F.1
		LCO 3.03.01 COND F RA F.2
		LCO 3.03.01 COND G RA G.1

DOC Number	DOC Text	
M.09 Rev. A	The applicability of CTS Table 15.3.5-2, item #4, Nuclear Flux Source Range, has been changed to include MODES 3, 4 and 5 with the RTBs closed and the Rod Control System capable of rod withdrawal. This change imposes additional requirements on plant operation and is therefore more restrictive. This change is necessary to ensure the availability of this function to mitigate an uncontrolled rod withdrawal event from subcritical conditions, as assumed in FSAR Section 14.1.1.	
	CTS:	**ITS:**
	15.03.05 T 15.03.05-02 04	LCO 3.03.01 T3.03.01-01 04-01
	NEW	LCO 3.03.01 T3.03.01-01 04-02
M.10 Rev. D	The actions for an inoperable Neutron Flux Source Range trip channel have been revised. Because CTS only requires one of the two channels to be operable when the required channel becomes inoperable the unit is required to be in hot shutdown in 8 hours. ITS requires 2 channels of Source Range Neutron Flux to be operable. The Required Actions for one inoperable channel in MODE 2, below P-6, have been adopted from NUREG-1431, requiring immediate suspension of operations involving positive reactivity additions. In MODES 3, 4 and 5 with the RTBs closed and the Rod Control System capable of rod withdrawal, an inoperable channel is required to be restored to an operable status in 48 hours or open the RTBs in 49 hours. If two channels of Source Range Neutron Flux are inoperable in any of the above conditions, immediate action is required to open the RTBs. Therefore, adopting the Required Actions of NUREG-1431 imposes additional requirements on unit operation and is more restrictive.	
	CTS:	**ITS:**
	15.03.05 T 15.03.05-02 04	LCO 3.03.01 COND I LCO 3.03.01 COND I RA I.1
	NEW	LCO 3.03.01 COND H LCO 3.03.01 COND H RA H.1 LCO 3.03.01 COND J LCO 3.03.01 COND J RA J.1 LCO 3.03.01 COND J RA J.2

DOC Number	DOC Text	
M.11 Rev. D	The Operator Actions of CTS Table 15.3.5-2, items #5, 6, 8 and 13 have been revised. If the Minimum Operable Channels requirement of CTS Table 15.3.5-2, column 3, cannot be met for any of the above functions, the unit is required to be in hot shutdown in 8 hours. Proposed ITS LCO 3.3.1, Condition D, requires placing the inoperable channel in the tripped condition in 1 hour or place the unit in MODE 3 in the next 6 hours. This results in placing the unit in a MODE where these functions are no longer required. The Completion Time of 6 hours is more restrictive than the current requirement, but is a reasonable amount of time to place the unit in MODE 3 from full power in an orderly manner without challenging unit systems.	
	CTS:	**ITS:**
	15.03.05 T 15.03.05-02 05	LCO 3.03.01 COND D RA D.2
	15.03.05 T 15.03.05-02 06	LCO 3.03.01 COND D RA D.2
	15.03.05 T 15.03.05-02 08	LCO 3.03.01 COND D RA D.2
	15.03.05 T 15.03.05-02 13	LCO 3.03.01 COND D RA D.2
M.12 Rev. D	CTS Table 15.3.5-2, item #7, Low Pressurizer Pressure, Operator Actions require the unit be in hot shutdown in 8 hours, if the conditions of column 3, Minimum Operable Channels, cannot be met for this function. The Low Pressurizer Pressure trip function protects against excessive boiling in the core and limits the pressure range in which the Overtemperature DT reactor trip is required to provide DNB protection. The Low Pressurizer Pressure trip function is blocked below 10% RTP, because of the excessive margin to DNB at lower power levels. Therefore this function is only required to be OPERABLE in MODE 1 above the P-7 interlock. ITS LCO 3.3.1, Condition K, requires an inoperable channel be placed in the tripped condition in 1 hour or reduce THERMAL POWER to < P-7 in the next 6 hours. This is more restrictive than the current requirement. CTS Table 15.3.5-2, item #7, requires the unit be in hot shutdown in 8 hours, but when THERMAL POWER is reduced to < 10% RTP, the actions can be discontinued per CTS 15.3.0.c. Therefore, the CTS allows additional time to reach this condition. The 6 hour completion time of ITS LCO 3.3.1, Required Action K.2, is a reasonable amount of time to reduce THERMAL POWER to < P-7 from full power in an orderly manner without challenging unit systems.	
	CTS:	**ITS:**
	15.03.05 T 15.03.05-02 07	LCO 3.03.01 COND K
		LCO 3.03.01 COND K RA K.2

DOC Number	DOC Text	
M.13 Rev. D	The Operator Actions of CTS Table 15.3.5-2, items #9 and 10.b require the unit to be in hot shutdown in 8 hours, if the Conditions of Column 3, Minimum Operable Channels, cannot be met for these functions. ITS LCO 3.3.1, Condition K, requires an inoperable channel be placed in the tripped condition in 1 hour or reduce THERMAL POWER to < P-7 in the next 6 hours. This results in placing the unit in a MODE where these Functions are no longer required. This is more restrictive than the current requirement. CTS Table 15.3.5-2 requires the unit be in hot shutdown in 8 hours, but when THERMAL POWER is reduced to < 10% RTP, the actions can be discontinued per CTS 15.3.0.c. Therefore the CTS allows additional time to reach this condition. The 6 hour completion time of ITS LCO 3.3.1, Required Action K.2, is a reasonable amount of time to reduce THERMAL POWER to < P-7 from full power in an orderly manner without challenging unit systems.	
	CTS:	**ITS:**
	15.03.05 T 15.03.05-02 09	LCO 3.03.01 COND K LCO 3.03.01 COND K RA K.2
	15.03.05 T 15.03.05-02 10.B	LCO 3.03.01 COND K LCO 3.03.01 COND K RA K.2
M.14 Rev. D	The Operator Actions of CTS 15.3.5-2, item #10.a, Low Reactor Coolant Flow, Single Loop, require the unit be in hot shutdown in 8 hours, if the Conditions of Column 3, Minimum Operable Channels, cannot be met for this function. ITS LCO 3.3.1, Condition L, requires an inoperable channel be placed in the tripped condition in 1 hour or reduce THERMAL POWER to < P-8 in the next 4 hours. This results in placing the unit in a MODE where this function is no longer required. This is more restrictive than the current requirement. CTS Table 15.3.5-2 requires the unit be in hot shutdown in 8 hours. Per CTS 15.2.3.2.B, the single loss of flow trip shall be unblocked greater than or equal to 50% of rated power. When THERMAL POWER is reduced to < 50% RTP, the actions can be discontinued per CTS 15.3.0.c. Therefore, the CTS allows additional time to reach this condition. The 4 hour completion time of ITS LCO 3.3.1, Required Action L.2, is a reasonable amount of time to reduce THERMAL POWER to < P-8 from full power in an orderly manner without challenging unit systems.	
	CTS:	**ITS:**
	15.03.05 T 15.03.05-02 10.A	LCO 3.03.01 COND L LCO 3.03.01 COND L RA L.2

DOC Number	DOC Text	
M.15 Rev. D	The actions for an inoperable Steam Flow/Feedwater Flow Mismatch channel have been revised. Because CTS only requires one of the two channels per loop to be operable, when the required channel becomes inoperable, the unit is required to be in hot shutdown in 8 hours. ITS requires 2 channels per loop of Steam Flow/Feedwater Flow Mismatch to be operable. The Required Actions for one inoperable channel requires placing the inoperable channel in trip in 6 hours or be in MODE 3 in 12 hours. If two channels of Steam Flow/Feedwater Flow Mismatch are inoperable, LCO 3.0.3 shall be entered, requiring the unit to be in MODE 3 in 7 hours. Therefore adopting the Required Actions of NUREG-1431 imposes additional requirements on unit operation and is more restrictive.	
	CTS:	**ITS:**
	15.03.05 T 15.03.05-02 12	LCO 3.03.03
	NEW	LCO 3.03.01 COND D
		LCO 3.03.01 COND D RA D.1
		LCO 3.03.01 COND D RA D.2
M.16 Rev. A	The frequency of CTS 15.4.1-2, items #24.b and 25.c have been changed from "Each refueling shutdown" to "18 months." The CTS does not define a specific frequency of performance for these surveillances, 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:**
	15.04.01 T 15.04.01-02 24 (B)	LCO 3.03.01 T3.03.01-01 01-01
		LCO 3.03.01 T3.03.01-01 01-02
		SR 3.03.01.13
	15.04.01 T 15.04.01-02 25 (C)	LCO 3.03.01 T3.03.01-01 01-01
		LCO 3.03.01 T3.03.01-01 01-02
		SR 3.03.01.13
M.17 Rev. D	Not used.	
	CTS:	**ITS:**
	N/A	N/A

DOC Number	DOC Text
M.18 Rev. D	The Operator Actions of CTS 15.4.1-1, item #14.b, Underfrequency Bus A01 and A02, require the unit to be in hot shutdown in 8 hours, if the Conditions of Column 3, Minimum Operable Channels cannot be met. Because Column 3 only requires 1 channel/each bus to be operable, no actions are required if one channel is inoperable such that one less than the total number of channels is still operable. In the ITS, 2 channels/each bus will be required to be operable. Therefore, it is necessary to require actions if one of the two required channels for a bus is inoperable. To address this, Condition E has been added to require one inoperable channel to be placed in trip within 6 hours, or reduce THERMAL POWER to < P-7 in 12 hours. The 6 hours to place an inoperable channel in trip is necessary, because it requires an Electrical Maintenance technician (who may not be on site during backshifts, weekends and holidays) to be dispatched to the location of the relay to place the channel in trip. Allowance of the 6 hours to reduce power to < P-7 takes into consideration the redundant capability provided by the remaining redundant OPERABLE channel, and the low probability of occurrence of an event during this period that may require the protection afforded by the Bus A01 and A02 Underfrequency trip function. This change imposes additional requirements on unit operation and is therefore more restrictive. **CTS:** 15.03.05 T 15.03.05-02 14.B **ITS:** LCO 3.03.01 COND E LCO 3.03.01 COND E RA E.1 LCO 3.03.01 COND E RA E.2
M.19 Rev. D	CTS Table 15.4.1-1, items #15.a and 15.b, Turbine Autostop and Turbine Stop Valve, have been revised by the addition of a TADOT surveillance requirement. This SR will be required to be performed prior to exceeding P-9, if not performed within the previous 31 days. Performance of this SR will ensure that the turbine trip function is operable prior to exceeding P-9. Adopting this SR imposes additional requirements on unit operation and is therefore more restrictive. **CTS:** NEW **ITS:** LCO 3.03.01 T3.03.01-01 15A LCO 3.03.01 T3.03.01-01 15B SR 3.03.01.14

DOC Number	DOC Text
M.20 Rev. D	The Operator Actions of CTS 15.3.5-2, item #16.a, RCP Breaker Open Position (>50% full power), require the unit be in hot shutdown in 8 hours, if the Conditions of Column 3, Minimum Operable Channels, cannot be met for this function. ITS LCO 3.3.1, Condition J requires restoration of the inoperable channel to an OPERABLE status within 1 hour, OR reduce THERMAL POWER to < P-8 in the next 4 hours. This results in placing the unit in a MODE where this function is no longer required. This is more restrictive than the current requirement. CTS Table 15.3.5-2 requires the unit be in hot shutdown in 8 hours, but when THERMAL POWER is reduced to < 50% RTP, the actions can be discontinued per CTS 15.3.0.c. Therefore the CTS allows additional time to reach this condition. The 4 hour completion time of ITS LCO 3.3.1, Required Action J.2, is a reasonable amount of time to reduce THERMAL POWER to < P-9 from full power. **CTS:** 15.03.05 T 15.03.05-02 16.A **ITS:** LCO 3.03.01 COND M LCO 3.03.01 COND M RA M.1 LCO 3.03.01 COND M RA M.2
M.21 Rev. D	The Operator Actions of CTS 15.3.5-2, item #16.b, RCP Breaker Open Position (10-50% full power), require the unit be in hot shutdown in 8 hours, if the Conditions of Column 3, Minimum Operable Channels, cannot be met for this function. ITS LCO 3.3.1, Condition K requires restoration of the inoperable channel to an OPERABLE status within 1 hour, OR reduce THERMAL POWER to < P-7 in the next 6 hours. This results in placing the unit in a MODE where this function is no longer required. This is more restrictive than the current requirement. CTS Table 15.3.5-2 requires the unit be in hot shutdown in 8 hours, but when THERMAL POWER is reduced to < 10% RTP, the actions can be discontinued per CTS 15.3.0.c. Therefore the CTS allows additional time to reach this condition. The 6 hour completion time of ITS LCO 3.3.1, Required Action K.2, is a reasonable amount of time to reduce THERMAL POWER to < P-7 from full power from full power in an orderly manner without challenging unit systems. **CTS:** 15.03.05 T 15.03.05-02 16.B **ITS:** LCO 3.03.01 COND N LCO 3.03.01 COND N RA N.1 LCO 3.03.01 COND N RA N.2

DOC Number	DOC Text
M.22 Rev. D	The Operator Actions of CTS Table 15.3.5-2, item #17, Reactor Trip Breakers, have been revised. If the Minimum Operable Channels requirement of CTS Table 15.3.5-2, column 3 cannot be met, the unit is required to be in hot shutdown in 8 hours. In MODES 1 and 2, ITS LCO 3.3.1, Condition Q requires the restoration of the inoperable RTB in 1 hour, OR be in MODE 3 in the next 6 hours. This Completion Time is more restrictive than the current requirement, but is a reasonable amount of time to place the unit in MODE 3 from full power in an orderly manner without challenging unit systems. Additionally, the Conditions and Required Actions for an inoperable RTB in MODES 3, 4 and 5 with the RTBs closed and the Rod Control System capable of rod withdrawal have been added. With the unit in this Condition, the RTBs are required to provide protection against an uncontrolled rod withdrawal accident. With one RTB inoperable, Condition T requires the restoration of the RTB in 48 hours or open the RTBs within the next 1 hour. This results in placing the unit in a MODE where the RTBs are no longer required.

CTS:	**ITS:**
15.03.05 T 15.03.05-02 17	LCO 3.03.01 COND Q
	LCO 3.03.01 COND Q RA Q.1
	LCO 3.03.01 COND Q RA Q.2
	LCO 3.03.01 COND T
NEW	LCO 3.03.01 COND T
	LCO 3.03.01 COND T RA T.1
	LCO 3.03.01 COND T RA T.2

DOC Number	DOC Text
M.23 Rev. D	CTS Table 15.3.5-2 has been modified by the addition of items 18, 19, 20, 21 and 22 and associated Conditions and Required Actions.

Item 18, Reactor Trip Breaker Undervoltage and Shunt Trip Mechanisms, are required to be OPERABLE for each RTB that is in service. They are not required to be OPERABLE for trip breakers that are open, racked out, incapable of supplying power to the Rod Control System, or declared inoperable under item 17. The OPERABILITY of both trip mechanisms on each breaker ensures that no single trip mechanism failure will prevent opening any breaker on a valid signal. Condition U applies to the RTB Undervoltage and Shunt Trip Mechanisms, or diverse trip features, in MODES 1 and 2. With one of the diverse trip features inoperable, it must be restored to an OPERABLE status within 48 hours or the unit must be placed in a MODE where the requirement does not apply. This is accomplished by placing the unit in MODE 3 within the next 6 hours. The Completion Time of 6 hours is a reasonable time, based on operating experience, to reach MODE 3 from full power in an orderly manner and without challenging unit systems. The Completion Time of 48 hours is reasonable considering that in this Condition there is one remaining diverse feature for the affected RTB, and one OPERABLE RTB capable of performing the safety function and given the low probability of an event occurring during this interval. Condition T applies to the RTB Undervoltage and Shunt Trip Mechanisms in MODES 3, 4, or 5 with the RTBs closed and the Rod Control System capable of rod withdrawal. With one trip mechanism inoperable, the inoperable trip mechanism must be restored to OPERABLE status within 48 hours. The Completion Time is reasonable considering that the remaining OPERABLE trip mechanism is adequate to perform the safety function, and given the low probability of an event occurring during this interval. If the trip mechanism cannot be restored to OPERABLE status within 48 hours, the unit must be placed in a MODE in which the requirement does not apply. This is accomplished by opening the RTBs within the next hour. The Completion Time of 1 hour provides sufficient time to accomplish this action in an orderly manner and takes into account the low probability of an event occurring in this interval. These Required Actions are consistent with the requirements of NUREG-1431.

Item 19, Reactor Trip System Interlocks, are provided to ensure reactor trips are in the correct configuration for the current unit status. They backup operator actions to ensure protection system Functions are not bypassed during unit conditions under which the safety analysis assumes the Functions are not bypassed. Therefore, the interlocks do not need to be OPERABLE when the associated reactor trip Functions are outside applicable MODES. Condition R applies to the P-6 and P-10 interlocks. With one or more channels inoperable for one-out-of-two or two-out-of-four coincidence logic, the associated interlock must be verified to be in its required state for the existing unit condition within 1 hour or the unit must be placed in MODE 3 within the next 6 hours. Verifying the interlock status manually accomplishes the interlock's Function. The Completion Time of 1 hour is based on operating experience and the minimum amount of time allowed for manual operator actions. The Completion Time of 6 hours is reasonable, based on operating experience, to reach MODE 3 from full power in an orderly manner and without challenging unit systems. The 1 hour and 6 hour Completion Times are equal to the time allowed by LCO 3.0.3 for shutdown actions in the event of a complete loss of RPS Function. Condition S applies to the P-7, P-8 and P-9 interlocks. With one or more channels inoperable for one-out-of-two or two-out-of-four coincidence logic, the associated interlock must be verified to be in its required state for the existing unit condition within 1 hour or the unit must be placed in MODE 2 within the next 6 hours. These actions are conservative for

DOC Number	DOC Text
	the case where power level is being raised. Verifying the interlock status manually accomplishes the interlock's Function. The Completion Time of 1 hour is based on operating experience and the minimum amount of time allowed for manual operator actions. The Completion Time of 6 hours is reasonable, based on operating experience, to reach MODE 2 from full power in an orderly manner and without challenging unit systems. Item 20, Reactor Trip Bypass Breaker and associated Undervoltage Trip Mechanism, are required to be OPERABLE when the Reactor Trip Bypass Breaker is racked in and closed. The bypass breaker and associated trip mechanism are not required to be OPERABLE when the Reactor Trip Bypass Breaker is open or racked out. Condition V applies to the Reactor Trip Bypass Breaker (RTBB) and associated Undervoltage Trip Mechanism in MODE 1 or 2, when the RTBB is racked in and closed. With the required RTBB inoperable, 1 hour is allowed to restore the RTBB to OPERABLE status or the unit must be placed in MODE 3 within the next 6 hours. The Completion Time of 6 hours is reasonable, based on operating experience, to reach MODE 3 from full power in an orderly manner and without challenging unit systems. The 1 hour and 6 hour completion times are equal to the time allowed by LCO 3.0.3 for shutdown action in the event of a complete loss of RPS Function. Placing the unit in MODE 3 removes the requirement for this particular Function. Condition W applies to the Reactor Trip Bypass Breaker (RTBB) and associated Undervoltage Trip Mechanism in MODES 3, 4, or 5, when an RTBB is racked in and closed and the Rod Control System is capable of rod withdrawal. With the required RTBB inoperable, 48 hours are allowed to restore the RTBB to OPERABLE status or the unit must be placed in a MODE in which the requirement does not apply. To achieve this status, the RTBs and RTBBs must be opened within the next 1 hour. The Completion Time of 1 hour provides sufficient time to accomplish the action in an orderly manner. With the RTBs and RTBBs open, this Function is no longer required. Item 21, Automatic Trip Logic, along with the requirements for the RTBs, ensure that means are provided to interrupt the power to allow the rods to fall into the reactor core. The reactor trip signals generated by the Automatic Trip Logic cause the RTBs and associated RTBBs to open and shutdown the reactor. Condition P applies to the RPS Automatic Trip Logic in MODES 1 and 2. With one train inoperable, 6 hours are allowed to restore the train to OPERABLE status or the unit must be placed in MODE 3 within the next 6 hours. The Completion Time of 6 hours to restore the train is reasonable considering that in this Condition, the remaining OPERABLE train is adequate to perform the safety function and given the low probability of an event during this interval. The Completion Time of 6 hours to place the unit in MODE 3 is reasonable, based on operating experience, to reach MODE 3 from full power in an orderly manner and without challenging unit systems. The Required Actions have been modified by a Note that allows bypassing one train for up to 8 hours for surveillance testing, provided the other train is OPERABLE. Condition X applies to the RPS Automatic Trip Logic in MODES 3, 4 or 5 with the RTBs closed and the Rod Control System capable of rod withdrawal. With one train inoperable, 48 hours are allowed to restore the train to an OPERBALE status. The Completion Time of 48 hours is reasonable considering that in this condition, the remaining OPERABLE train is adequate to perform the safety function, and given the low probability of an event occurring in this interval. If the RPS Automatic Trip Logic cannot be restored to OPERABLE status within 48 hours, the unit must be placed in a MODE where this Function is not required to be OPERABLE. To achieve this status, the RTBs must be opened within the next 1 hour. The additional hour provides sufficient time to accomplish the action in an orderly manner. With the RTBs open, the Automatic Trip Logic is no longer required.

DOC Number	DOC Text

Item 22, SG Water Level - Low in conjunction with the Steam Flow / Feedwater Flow Mismatch, ensures protection is provided against a loss of heat sink. This instrumentation is required to be operable in MODES 1 and 2, when the MFW System is in operation and the reactor is in operation. There are 2 SG level channels and 2 Steam Flow / Feedwater Flow Mismatch channels per SG; one Narrow Range channel sensing a low-level coincident with one Steam Flow / Feedwater Flow Mismatch channel sensing flow mismatch will actuate a reactor trip. The LCO requires one channel per SG of SG Water Level - Low coincident with Steam Flow / Feedwater Flow Mismatch. Condition D applies to the SG Water Level—Low coincident with Steam Flow / Feedwater Flow Mismatch in MODE 1 or 2. With one channel of the SG Water Level - Low or Steam Flow / Feedwater Flow Mismatch instrumentation inoperable, the inoperable channel is required to be placed in trip in 1 hour. If this cannot be accomplished within the allowed 1 hour Completion Time, the unit must be brought to a MODE in which the requirement does not apply. To achieve this status, the unit must be brought to at least MODE 3 within 6 additional hours. The 6 additional hours to reach MODE 3 is reasonable, based on operating experience, to reach MODE 3 from full power operation in an orderly manner and without challenging unit systems. With the unit in MODE 3, the trip Function is no longer required to be OPERABLE.

CTS Tables 15.4.1-1 and 15.4.1-2 contain surveillance requirements for the above functions, thereby establishing the requirements for their OPERABILITY. Adding these functions to CTS Table 15.3.5-2 clarifies the MODES under which each function is required to be OPERABLE and provides Required Actions to take in the event of inoperable channel(s). This change imposes additional requirements on unit operation and is therefore more restrictive.

CTS:	**ITS:**
15.04.01 T 15.04.01-01 05	SR 3.03.01.01
NEW	LCO 3.03.01 COND D
	LCO 3.03.01 COND D RA D.1
	LCO 3.03.01 COND D RA D.2
	LCO 3.03.01 COND P
	LCO 3.03.01 COND P RA P.1
	LCO 3.03.01 COND P RA P.2
	LCO 3.03.01 COND R
	LCO 3.03.01 COND R RA R.1
	LCO 3.03.01 COND R RA R.2
	LCO 3.03.01 COND S
	LCO 3.03.01 COND S RA S.1
	LCO 3.03.01 COND S RA S.2
	LCO 3.03.01 COND T
	LCO 3.03.01 COND T RA T.1
	LCO 3.03.01 COND T RA T.2
	LCO 3.03.01 COND U
	LCO 3.03.01 COND U RA U.1

DOC Number	DOC Text
NEW	LCO 3.03.01 COND U RA U.2
	LCO 3.03.01 COND V
	LCO 3.03.01 COND V RA V.1
	LCO 3.03.01 COND V RA V.2
	LCO 3.03.01 COND W
	LCO 3.03.01 COND W RA W.1
	LCO 3.03.01 COND W RA W.2
	LCO 3.03.01 COND X
	LCO 3.03.01 COND X RA X.1
	LCO 3.03.01 COND X RA X.2
	LCO 3.03.01 T3.03.01-01 14-01
	LCO 3.03.01 T3.03.01-01 14-02
	LCO 3.03.01 T3.03.01-01 17A
	LCO 3.03.01 T3.03.01-01 17B-01
	LCO 3.03.01 T3.03.01-01 17C
	LCO 3.03.01 T3.03.01-01 17D
	LCO 3.03.01 T3.03.01-01 17E
	LCO 3.03.01 T3.03.01-01 18-01
	LCO 3.03.01 T3.03.01-01 18-02
	LCO 3.03.01 T3.03.01-01 19-01
	LCO 3.03.01 T3.03.01-01 19-02
	LCO 3.03.01 T3.03.01-01 20-01
	LCO 3.03.01 T3.03.01-01 20-02
	LCO 3.03.01 T3.03.01-01 21-01
	LCO 3.03.01 T3.03.01-01 21-02
	LCO 3.03.01 T3.03.01-01 NOTE (a)
	LCO 3.03.01 T3.03.01-01 NOTE (b)
	LCO 3.03.01 T3.03.01-01 NOTE (C)
	LCO 3.03.01 T3.03.01-01 NOTE (d)
	LCO 3.03.01 T3.03.01-01 NOTE (e)
	LCO 3.03.01 T3.03.01-01 NOTE (f)
	LCO 3.03.01 T3.03.01-01 NOTE (g)
	LCO 3.03.01 T3.03.01-01 NOTE (h)
	LCO 3.03.01 T3.03.01-01 NOTE (i)
	LCO 3.03.01 T3.03.01-01 NOTE (k)
	LCO 3.03.01 T3.03.01-01 NOTE (l)
	LCO 3.03.01 T3.03.01-01 NOTE (n)

DOC Number	DOC Text	
M.24 Rev. D	CTS Table 15.4.1-1, item #1, Daily Heat Balance of the Nuclear Power Range instrumentation has been modified by a Note requiring the adjustment of the NIS channel if the absolute difference between the NIS channel output and the calorimetric is greater than 2% RTP. CTS Table 15.4.1-1, item #1, Daily Heat Balance of the Nuclear Power Range instrumentation has been modified by a Note requiring the adjustment of the NIS channel if the absolute difference between the NIS channel output and the calorimetric is greater than 2% RTP. This daily adjustment supports the accuracy assumptions in establishing the low and high flux trip setpoints. A calorimetric uncertainty of 2% is factored into the setpoint calculation along with other NI channel inaccuracies. This is consistent with the Point Beach current practice of adjusting NIS channel output, if the absolute difference between the calorimetric heat balance and the NIS channel output exceeds 2% RTP. This change imposes additional requirements on unit operation and is therefore more restrictive. This change imposes additional requirements on unit operation and is therefore more restrictive.	
	CTS:	**ITS:**
	15.04.01 T 15.04.01-01 01.A	LCO 3.03.01 T3.03.01-01 02A
	NEW	SR 3.03.01.02 NOTE 1
M.25 Rev. A	CTS Table 15.4.1-1, items #1, 2 and 3, Quarterly Channel Functional Test surveillance requirements are modified by a Note requiring the verification that the P-6 and P-10 interlocks are in their required state for existing unit conditions. This change imposes additional requirements on unit operation, but is necessary to ensure the operability of the trip functions.	
	CTS:	**ITS:**
	15.04.01 T 15.04.01-01 01.B	LCO 3.03.01 T3.03.01-01 02B SR 3.03.01.08 NOTE
	15.04.01 T 15.04.01-01 02.B	SR 3.03.01.08 NOTE
	15.04.01 T 15.04.01-01 03.B	LCO 3.03.01 T3.03.01-01 04-01 SR 3.03.01.08 NOTE
	NEW	SR 3.03.01.08 NOTE
M.26 Rev. A	The Channel Calibration surveillance requirements for CTS Table 15.4.1-1, items #11 and 12, 4KV Bus Undervoltage (A01 & A02) and 4KV Bus Underfrequency (A01 & A02), have been modified by the adoption of the NUREG-1431, SR 3.3.1.10, Note. This Note requires the verification that the time delays associated with these functions are adjusted to the prescribed values. This change imposes additional requirements on unit operation and is therefore more restrictive.	
	CTS:	**ITS:**
	NEW	SR 3.03.01.10 NOTE

DOC Number	DOC Text
M.27 Rev. D	CTS Table 15.4.1-1, Frequency P, "Prior to reactor criticality, if not performed during the previous week," has been revised, as it applies to proposed ITS SR 3.3.1.8. The frequency of SR 3.3.1.8 has the following additional requirements; four hours after reducing power below P-10 for power and intermediate range instrumentation, four hours after reducing power below P-6 for source range instrumentation, every 92 days thereafter. The frequency of every 92 days applies if the plant remains in the MODE of applicability after the initial performance prior to startup and four hours after reducing power < P-6 and < P-10. The MODE of applicability for this surveillance is < P-10 for power range low and intermediate range channels, and < P-6 for the source range channels. Once the unit is in MODE 3, this surveillance is no longer required. If power is to be maintained < P-10 or < P-6 for > 4 hours, then the testing required by this surveillance must be performed prior to the expiration of the 4 hour limit. This change imposes additional requirements on unit operation and is therefore more restrictive. **CTS:** 15.04.01 T 15.04.01-01 02.B **ITS:** LCO 3.03.01 T3.03.01-01 03
M.28 Rev. A	The applicability of CTS Table 15.4.1-1, item #3, Nuclear Source Range, has been changed to include MODES 3, 4 and 5 with the RTBs closed and the Rod Control System capable of rod withdrawal. This necessitates the addition of surveillance requirements to adequately verify the OPERABILITY of the function in these MODES. Therefore SR 3.3.1.1, SR 3.3.1.7 and SR 3.3.1.11 have been adopted for the Source Range Neutron Flux instrumentation in MODES 3, 4 and 5 with the RTBs closed and the Rod Control System capable of rod withdrawal. This change imposes additional requirements on unit operation and is therefore more restrictive. **CTS:** 15.04.01 T 15.04.01-01 03; NEW **ITS:** LCO 3.03.01 T3.03.01-01 04-01; SR 3.03.01.01; SR 3.03.01.07; SR 3.03.01.11; SR 3.03.01.11 NOTE
M.29 Rev. A	CTS Table 15.4.1-1, OT deltaT, has been revised by adopting ITS SR 3.3.1.6, Calibration of excore channels in agreement with incore detector measurements every 92 EFPD. This addition imposes additional requirements on unit operation and is therefore more restrictive. This surveillance is necessary to verify the f(delta I) input to the OT delta T function. A note modifying the SR states the surveillance is only required if reactor power is > 50% RTP and 24 hours is allowed for performing the first surveillance after reaching 50% RTP. The frequency of 92 EFPD is adequate. It is based on industry operating experience, considering instrument reliability and operating history data for instrument drift. **CTS:** NEW **ITS:** SR 3.03.01.06

DOC Number	DOC Text	
M.30 Rev. D	CTS Table 15.4.1-1, surveillance requirement to compare results of incore detector measurements to NIS axial flux difference has been modified by a Note clarifying this surveillance is not required to be performed until 24 hours after THERMAL POWER is greater than or equal to 50% RTP. Per CTS 15.4.0.4, the reactor shall not be placed in a condition where a system or component is required to be operable, if the specified surveillances have not been performed satisfactorily within their specified frequencies. However, CTS 15.4.0.4 also states if entry into a condition where the system or component is required to be operable is necessary in order to perform the specified surveillance, entry into the operating conditions may be made provided prior testing or inspection provides reasonable assurance of operability, and the surveillance is performed as soon as practicable following entry into the required operating condition. Point Beach operating experience has shown that an accurate comparison of incore detector measurements to NIS axial flux difference cannot be made at lower reactor powers. Therefore, adoption of ITS SR 3.3.1.3, Note 2 is consistent with current operating practice and the requirements of CTS 15.4.0.4. However, specifying the surveillance is "required to be performed within 24 hours after THERMAL POWER is greater than or equal to 50% RTP," is more restrictive than the CTS requirement of "as soon as practicable following entry into the required operating condition. **CTS:** 15.04.0.04	**ITS:** SR 3.03.01.03 NOTE 2
M.31 Rev. D	CTS 15.2.3 has been modified by the addition of field settings for SG Water Level - Low and Turbine Trip – Low Autostop Oil Pressure. These field settings were developed outside the scope of the setpoint methodology and can be found in documents provided by the NSSS supplier. Adopting these field settings does not imply that an analytical limit exists for these functions, or that these functions are necessary to prevent exceeding a safety limit. **CTS:** NEW	**ITS:** LCO 3.03.01 T3.03.01-01 14-01 LCO 3.03.01 T3.03.01-01 15A LCO 3.03.01 T3.03.01-01 NOTE (j) LCO 3.03.01 T3.03.01-01 NOTE (m)
M.32 Rev. D	CTS 15.4.1, Table 15.4.1-1, Function 16, Reactor Trip Signal from SI, has been modified by the addition of SR 3.3.1.13, TADOT, with a frequency of 18 months. Performance of this SR will ensure that the SI input to RPS for initiation of a reactor trip is operable. Adopting this SR imposes additional requirements on unit operation and is therefore more restrictive. This change is consistent with NUREG-1431. **CTS:** NEW	**ITS:** SR 3.03.01.13

DOC Number	DOC Text
M.33 Rev. D	CTS 15.3.5, Table 15.3.5-2, Functions 1, 3, 4, 12 and 14b, Manual Reactor Trip, Neutron Flux Intermediate Range, Neutron Flux Source Range, Steam Flow – Feedwater Flow Mismatch, and 4KV Bus Underfrequency, respectively, have been revised to require additional channels to be operable. In the proposed ITS LCO 3.3.1, Table 3.3.1-1, the Manual Reactor Trip Function will require 2 channels; the Neutron Flux Intermediate Range Function will require 2 channels; the Neutron Flux Source Range Function will require 2 channels; the Steam Flow – Feedwater Flow Mismatch Function will require 2 channels/loop; and the 4KV Bus Underfrequency Function will require 2 channels/bus. Increasing the required number of channels for each of these functions imposes additional requirements on unit operation and is therefore more restrictive. This change is consistent with NUREG-1431.

CTS:	ITS:
15.03.05 T 15.03.05-02 01	LCO 3.03.01 T3.03.01-01 01-01
15.03.05 T 15.03.05-02 03	LCO 3.03.01 T3.03.01-01 03
15.03.05 T 15.03.05-02 04	LCO 3.03.01 T3.03.01-01 04-01
15.03.05 T 15.03.05-02 12	LCO 3.03.01 T3.03.01-01 14-02
15.03.05 T 15.03.05-02 14.B	LCO 3.03.01 T3.03.01-01 12

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 Effluent Radiation Monitoring and Monitoring Programs can be relocated to licensee control.

The area radiation and process monitors are used to indicate when the radiation in the area or effluent stream has exceeded its allowable setpoint. There are no safety related automatic functions assumed in accident analyses that are performed by these instruments. The instruments are not used to mitigate a design basis accident or transient. The monitoring programs provide data on radiation levels, releases, and radioactive materials in exposure pathways. The land census program assures that changes in land use are identified and accounted for in the Environmental Monitoring Programs. The inter-laboratory comparison program assure independent checks are performed on the precision and accuracy of the instrumentation used to identify and quantify radioactive materials. In comparison to the screening criteria:

1. By established convention, this criterion refers to monitors to detect degradation that could lead to leakage from the RCS to the containment atmosphere. None of the radiation monitors nor programs on this list are used for that purpose.

2. The monitored parameters and programs are not process variables, design features, or operating restrictions that are initial conditions of a DBA or transient.

3. These monitors and programs are not devices that provide the primary automatic response to a DBA or transient.

4. Functions of radiation monitors like these have traditionally not been judged, by operating experience or by PSA, to represent significant risk to the public. The functions served by these monitors at Point Beach are similar to those at other plants. As discussed in Section 4.0 (Appendix A, page A-69) and summarized in Table 1 of WCAP-11618, the environmental monitoring and effluent monitoring instrumentation were deterministically found to be a non-significant risk contributor to core damage frequency and offsite releases.

In conclusion, since the screening criteria have not been satisfied, the Plant Radiation Monitoring LCOs and Surveillances for the instruments described in the LCO statement section above may be relocated to other plant controlled documents outside the Technical Specifications.

CTS:	**ITS:**
15.04.01 T 15.04.01-01 29	TRM 3.03.01 T 3.03.01-01 01
15.04.01 T 15.04.01-01 36-01	TRM 3.03.01 T 3.03.01-01 02A
15.04.01 T 15.04.01-01 36-02	TRM 3.03.01 T 3.03.01-01 02B
15.04.01 T 15.04.01-01 NOTE (7)	TSR 3.03.01.01

A.1

15.3.5 INSTRUMENTATION SYSTEM

Operational Safety Instrumentation

Applicability: Applies to plant instrumentation systems.

Objectives: To provide for automatic initiation of the Engineered Safety Features in the event that principal process variable limits are exceeded, and to delineate the conditions of the plant instrumentation and safety circuits necessary to ensure reactor safety.

Specification:

< See LCO 3.3.2 >

A. The Engineered Safety Features initiation instrumentation setting limits shall be as stated in Table 15.3.5-1.

B. For on-line testing or in the event of a sub-system instrumentation channel failure, plant operation at rated power shall be permitted to continue in accordance with Tables 15.3.5-2 through 15.3.5-4 < See LCO 3.3.2 >

C. In the event the number of channels of a particular sub-system in service falls below the limits given in the column entitled Minimum Operable Channels, operation shall be limited according to the requirement shown in Tables 15.3.5-2 through 15.3.5-4, Operator Action when minimum operable channels unavailable. < See LCO 3.3.2 > A.3

D. The post-accident monitoring instrumentation channels in Table 15.3.5-5 shall be operable. In the event the number of channels in a particular sub-system falls below the minimum number of operable channels given in Column 2, operation and subsequent operator action shall be in accordance with Column 3. This specification is not applicable in the cold or refueling shutdown conditions. < See LCO 3.3.3 >

Basis: Instrumentation has been provided to sense accident conditions and to initiate operation of the Engineered Safety Features(1)

A.2

Separate Condition entry is allowed for each function. A.32

D

RAI 3.3.1-29

A.1

TABLE 15.3.5-2
INSTRUMENT CONDITIONS FOR REACTOR TRIP

M.2: In MODE 1 or 2, restore inoperable channel within 48 hours, OR be in MODE 3 in 54 hours. In MODE 3, 4 or 5, with RTBs closed and Rod Control System capable of rod withdrawal, restore inoperable channel within 48 hours, OR open RTBs within 49 hours.

A.3 → REQUIRED

A.4 → APPLICABILITY

NO.	FUNCTIONAL UNIT	~~1~~ ~~TOTAL NO. OF CHANNELS~~	~~2~~ ~~NO. OF CHANNELS TO TRIP~~	3 MINIMUM OPERABLE CHANNELS	4 PERMISSIBLE BYPASS CONDITIONS	OPERATOR ACTION IF CONDITIONS OF COLUMN 3 CANNOT BE MET
1.	Manual	~~2~~	~~1~~	M.33 2 → 1	A.8 $1, 2, 3^{(a)}, 4^{(a)}, 5^{(a)}$ M.1	Be in hot shutdown in 8 hours
2.	Nuclear Power Range		LA.1			
a.	low setting	4	~~2~~	A.3 4 → $3^{\#**}$ See LCO 3.1.10	$1^{(b)}, 2$ 2 of 4 power range channels greater than 10% full power (low setting only)	Be in hot shutdown in 8 hours M.5 Be in MODE 3 in the next 6 hours.
b.	high setting	4	~~2~~	A.3 4 → $3^{\#**}$	1, 2 A.7	Be in hot shutdown in 8 hours
3.	Nuclear Flux Intermediate Range	~~2~~	~~1~~	M.33 2 → 1	$1^{(b)}, 2^{(c)}$ A.8 2 of 4 power range channels greater than 10% full power	A.6 Be in hot shutdown in 8 hours* Replace with Insert 15.3.5-1 M.8
4.	Nuclear Flux Source Range	~~2~~	~~1~~	M.33 2 → 1	$2^{(d)}, 3^{(a)}, 4^{(a)}, 5^{(a)}$ M.9 1 of 2 intermediate range channels greater than 10^{-10} amps.	Replace with Insert 15.3.5-2 M.10 Be in hot shutdown in 8 hours* A.6
5.	Overtemperature Delta T	4	~~2~~	A.3 4 → 3^{**}	1, 2 A.7	M.11 Be in MODE 3 in the next 6 hrs. Be in hot shutdown in 8 hours
6.	Overpower Delta T	4	~~2~~	4 → 3^{**}	1, 2 A.7	Be in hot shutdown in 8 hours
7.	Low Pressurizer Pressure	4	~~2~~	4 → 3^{**}	$1^{(e)}$ A.8	M.12 Reduce thermal power to < P-7 in next 6 hours. Be in hot shutdown in 8 hours
8.	Hi Pressurizer Pressure	~~3~~	~~2~~	3 → 2^{**}	1, 2 A.7	Be in hot shutdown in 8 hours

INSERT 15.3.5-1:
For one inoperable channel, either reduce power to < P-6 or increase power to > P-10 in 24 hours.

For two inoperable channels, immediately suspend operations involving positive reactivity additions and reduce power to < P-6 in 2 hours.

INSERT 15.3.5-2:
In MODE 2, below P-6, for one inoperable channel, immediately suspend operations involving a positive reactivity addition. In MODES 3, 4 and 5 with the RTBs closed and the Rod Control System capable of rod withdrawal, restore one inoperable channel to operable status in 48 hours or open RTBs in 49 hours. For two inoperable channels, immediately open RTBs.

A.1

TABLE 15.3.5-2(continued)

A.3 → REQUIRED   A.4 → APPLICABILITY

NO.	FUNCTIONAL UNIT	1 ~~TOTAL NO. OF CHANNELS~~	2 ~~NO. OF CHANNELS TO TRIP~~	3 REQUIRED MINIMUM OPERABLE CHANNELS	4 APPLICABILITY PERMISSIBLE BYPASS CONDITIONS	OPERATOR ACTION IF CONDITIONS OF COLUMN 3 CANNOT BE MET
9.	Hi Pressurizer Water Level	~~3~~	~~2~~	3 → ~~2~~**	$1^{(e)}$ ← A.31	Be in hot shutdown in 8 hours M.13 → Reduce thermal power to < P-7 in next 6 hours.
10.	Low Reactor Coolant System Flow	← LA.1 →		A.3		
a.	Low Flow in One Loop (>50% full power)	~~3/loop~~	~~2/loop~~	3 → ~~2~~/loop** (any loop)	$1^{(f)}$ ← A.8	M.14 → Reduce thermal power to < P-8 in 4 next hours. Be in hot shutdown in 8 hours
b.	Low Flow in Both Loops (10-50% full power)	~~3/loop~~	~~2/loop (both loops)~~	3 → ~~2~~/loop**	$1^{(g)}$ ← A.31	Be in hot shutdown in 8 hours M.13 → Reduce thermal power to < P-7 in next 6 hours.
11.	Turbine Trips			A.3		
a.	Turbine Autostop Oil Pressure	~~3~~	~~2~~	3 → ~~2~~**	$1^{(j)}$ ← A.8	Be <50% of rated power within 4 hours
b.	Turbine Stop Valve Position	~~2~~	~~2~~	2**	$1^{(j)}$	Be <50% of rated power within 4 hours
				M.33		
12.	Steam Flow-Feedwater Flow Mismatch	~~2/loop~~	~~1/loop~~	2 → ~~1~~/loop	1,2 ← A.7	Be in MODE 3 within next 6 hours. ← M.15 Be in hot shutdown in 8 hours
				A.3		
13.	Lo Lo Steam Generator Water Level (input to reactor trip)	~~3/loop~~	~~2/loop~~	3 → ~~2~~/loop**	1,2	Be in MODE 3 within next 6 hours. ← M.11 ~~Be in hot shutdown in 8 hours~~
				A.3		
14.	4KV Bus (A01 and A02)					
a.	Undervoltage (input to reactor trip)	~~2/each bus~~	~~1/each bus~~	2 → ~~1~~/each** bus	$1^{(e)}$ ← L.14	Reduce thermal power to < P-7 in next 6 hours. ← L.14 Be in hot shutdown in 8 hours
b.	Underfrequency	~~2/each bus~~	~~1/each bus~~	2 → ~~1~~/each bus M.33	$1^{(e)}$	Be in hot shutdown in 8 hours M.18 → Reduce thermal power to < P-7 within next 6 hours.

D

RAIs 3.3.1-1, 3.3.1-2, 3.3.1-4, 3.3.1-5, 3.3.1-7, 3.3.1-8, 3.3.1-30

A.1

TABLE 15.3.5-2(continued)

A.3 REQUIRED

A.4 APPLICABILITY

D RAI 3.3.1-1

NO.	FUNCTIONAL UNIT	~~1~~ ~~TOTAL NO. OF CHANNELS~~	~~2~~ ~~NO. OF CHANNELS TO TRIP~~	3 MINIMUM OPERABLE CHANNELS	4 PERMISSIBLE BYPASS CONDITIONS	OPERATOR ACTION IF CONDITIONS OF COLUMN 3 CANNOT BE MET
15.	Safety Injection	~~See Table 15.3.5-3~~	~~1~~	See Table 15.3.5-3 / 2 trains ### (A.10, L.3)	1,2 (A.7)	Be in hot shutdown in 8 hours ~~***~~ (L.4) / Restore one inoperable train to operable status within 6 hrs. OR be in MODE 3 in the next 6 hrs. (L.5)
16.	RCP Breaker Open Position	LA.1				Restore inoperable channel to operable status within 1 hour, OR reduce thermal power to < P-8 within the next 5 hours. (M.20) D RAI 3.3.1-10
a.	(>50% full power)	~~2~~	~~1~~	2	1[f] (A.8)	Be in hot shutdown in 8 hours
b.	(10 - 50% full power)	~~2~~	~~2~~	2	1[g] (A.8)	Be in hot shutdown in 8 hours (M.21)
17.	Reactor Trip Breakers	~~2~~	~~1~~	2 (M.23)	**** 1, 2, 3[(a)], 4[(a)], 5[(a)] (A.9)	Be in hot shutdown in 8 hours / Restore inoperable channel to operable status within 1 hour, OR reduce thermal power to < P-7 within the next 6 hours. D RAI 3.3.1-10

Add Reactor Trip Breaker Undervoltage and Shunt Trip Mechanisms, Reactor Trip System Interlocks, Reactor Bypass Breaker and associated Undervoltage Trip Mechanism, Automatic Trip Logic and SG Water Level - Low

\# One additional channel may be taken out of service for low power physics testing. See LCO 3.1.10

In MODE 1 or 2, restore inoperable RTB to operable status within 1 hour, OR be in MODE 3 in the next 6 hours. In MODE 3, 4 or 5, with RTBs closed and Rod Control System capable of rod withdrawal restore inoperable RTB to operable status in 48 hours, OR open RTBs within the next 1 hour. M.22

~~* When block condition exists, maintain normal operation.~~ A.6

** If a channel is determined to be inoperable, resulting in one less than the total number of channels being operable, power operation may continue if the following conditions are met:

1. The minimum number of operable channels is still satisfied.
2. The affected channel is placed in trip within 1 hour.

D RAI 3.3.1-30

~~*** If minimum conditions are not met within 24 hours after reaching hot shutdown, the unit shall be in cold shutdown within 48 hours of the event causing the unit shutdown.~~ L.4

**** When at power, one channel may be bypassed for up to 8 hours provided that the other channel is operable. When the plant is shutdown and rod withdrawal is possible, restore the inoperable channel to operable status within 48 hours or open the Reactor Trip Breakers within 1 hour.

D RAI 3.3.1-30

\#\#\# One train may be bypassed for up to 8 hours for surveillance testing provided the other train is OPERABLE. L.3

Function	Applicability	Required Channels	Conditions and Required Actions
18. Reactor Trip Breaker Undervoltage and Shunt Trip Mechanisms	1, 2	1 each per RTB	Restore one trip mechanism to an OPERABLE status within 48 hours, OR be in MODE 3 within the next 6 hours.
	3(a), 4(a), 5(a)	1 each per RTB	Restore one RTB or trip mechanism to an OPERABLE status within 48 hours, OR open RTBs within the next 1 hour.
19. Reactor Trip System Interlocks			
a. Intermediate Range Neutron Flux, P-6	2(c)	2	Verify interlock is in required status for existing unit conditions within 1 hour, OR be in MODE 3 within next 6 hours.
b. Low Power Reactor Trip Block, P-7			
(1) Power Range Neutron Flux	1	4	Verify interlock is in required status for existing unit conditions within 1 hour, OR be in MODE 2 within next 6 hours.
(2) Turbine Impulse Pressure	1	2	Verify interlock is in required status for existing unit conditions within 1 hour, OR be in MODE 2 within next 6 hours.
c. Power Range Neutron Flux, P-8	1	4	Verify interlock is in required status for existing unit conditions within 1 hour, OR be in MODE 2 within next 6 hours.
d. Power Range Neutron Flux, P-9	1(l)	4	Verify interlock is in required status for existing unit conditions within 1 hour, OR be in MODE 2 within next 6 hours.
e. Power Range Neutron Flux, P-10	1, 2	4	Verify interlock is in required status for existing unit conditions within 1 hour, OR be in MODE 3 within next 6 hours.

D

RAI 3.3.1-14, RAI 3.3.1-27, Errata #145

Function	Applicability	Required Channels	Conditions and Required Actions
20. Reactor Trip Bypass Breaker and associated Undervoltage Trip Mechnanism	1(k), 2(k)	1	Restore one RTB to OPERABLE status within 1 hour, OR be in MODE 3 within the next 6 hours.
	3(k), 4(k), 5(k)	1	Restore RTB to OPERABLE status within 48 hours, OR open RTBs and RTBBs within the next 1 hour.
21. Automatic Trip Logic	1, 2	2 trains###	Restore one train to OPERABLE status within 6 hours, OR be in MODE 3 within the next 6 hours.
	3(a), 4(a), 5(a)	2 trains###	Restore train to OPERABLE status within 48 hours, OR open RTBs within the next 1 hour.
22. SG Water Level-Low	1, 2	2 per SG	Restore required channel to OPERABLE status within 1 hour, OR be in MODE 3 within the next 6 hours.

D

RAI 3.3.1-1

Notes:

(a) With RTBs closed and Rod Control System capable of rod withdrawal.
(b) Below the P-10 (Power Range Neutron Flux) interlocks.
(c) Above the P-6 (Intermediate Range Neutron Flux) interlock.
(d) Below the P-6 (Intermediate Range Neutron Flux) interlock.
(e) With the RTBs open.
(f) Above the P-7 (Low Power Reactor Trip Block) interlock.
(g) Above the P-8 (Power Range Neutron Flux) interlock.
(h) Above the P-7 (Low Power Reactor Trip Block) and below the P-8 (Power Range Neutron Flux) interlocks.
(j) Above the P-9 interlock.
(k) When a RTBB is racked in and closed and Rod Control System is capable of rod withdrawal.
(l) With one of two circulating water pump breakers closed and vacuum $\geq 22''$ Hg

D

RAIs 3.3.1-4
RAI 3.3.1-5
RAI 3.3.1-7
RAI 3.3.1-8
RAI 3.3.1-10
RAI 3.3.1-12
RAI 3.3.1-27
Errata #145

A.1

TABLE 15.4.1-1

MINIMUM FREQUENCIES FOR CHECKS, CALIBRATIONS, AND TESTS OF INSTRUMENT CHANNELS

NO.	CHANNEL DESCRIPTION	CHECK	CALIBRATE	TEST	PLANT CONDITIONS WHEN REQUIRED
1.	Nuclear Power Range (A.12: High Setting; Low Setting) ~~-Heat Balance~~ ~~-Signal to delta T; bistable actions(rod stops, trips)~~ -Compare results of the incore detector measurements to NIS axial flux difference (A.15)	S(~~1~~) (L.13, A.11; A.12: SR 3.3.1.1; Insert SR 3.3.1.2 Note 2 (L.6)) - (Insert SR 3.3.1.2 Note 1 (M.24); SR 3.3.1.11) M(4,5,~~20~~) (L.7: 31 EFPD; A.14)	R D(~~1~~,19) (A.11, L.19; Insert SR 3.3.1.11 Note (A.29)) - (Insert SR 3.3.1.8 Note (M.25)) - (Insert SR 3.3.1.3 Note 2 (M.30))	- (SR 3.3.1.7, A.12) Errata #90 Q(~~1~~,2) (A.13, A.11) - RAI 3.3.1-19	ALL (1, 2, 3(a), 4(a), 5(a) — M.6) ALL (1, 2 — A.12) ALL (1(b), 2, 3(a), 4(a), 5(a) — M.4) PWR (1, 2 — A.7)
2.	Nuclear Intermediate Range ~~-when not blocked~~ ~~-logarithmic level; bistable action (rod stop, trips)~~ (A.16)	S(~~1~~) (L.13, A.11) - -	R (Insert SR 3.3.1.11 Note (A.29)) - -	- (Insert SR 3.3.1.8 Note (M.25)) P (M.27, L.9) -	ALL (1(b), 2, 3(a), 4(a), 5(a) — M.7) ALL ALL
3.	Nuclear Source Range (A.17) ~~-when not blocked~~ ~~-Bistable action (alarm and trips)~~	- (M.28: SR 3.3.1.1) S (A.17, L.13) -	R (A.17; M.28: SR 3.3.1.11) - - (Insert SR 3.3.1.11 Note (A.29))	- (M.28: SR 3.3.1.7) P (M.27, L.9; Insert SR 3.3.1.8 Note (M.25)) RAI 3.3 1-17	ALL (3(a), 4(a), 5(a) — M.9) ALL ALL (2(c) — A.8)
4.	Reactor Coolant Temperature -Overtemperature delta T -Overpower delta T	S (L.13) - -	R - (SR 3.3.1.6 — M.29) -	- Q(~~1~~,2) (A.18) Q(~~1~~,2)	PWR, HOT S/D, COLD S/D (1, 2 — L.10) ALL (1, 2 — A.8) ALL
5.	Reactor Coolant Flow -Analog and single loop loss-of-flow logic testing -Logic channel testing for reactor trip on loss of reactor coolant flow in both loops	S(~~1~~) (A.19, L.13, A.11) - -	R (A.19) - -	- (A.11) Q(~~1~~,2) R (A.18)	~~ALL~~ (A.19) ALL (1(f) — A.8) ALL (1(g) — A.31) RAI 3 3 1-4
6.	Pressurizer Water Level	S(~~1~~) (A.11)	R	Q(~~1~~,2) (A.11)	ALL (1(e) — A.31)
7.	Pressurizer Pressure	S(~~1~~) (L.13)	R	Q(~~1~~,2)	ALL (1, 2 — A.7)
8.	Steam Generator Level	S(~~1~~)	R	Q(~~1~~,17)	ALL

A.1

TABLE 15.4.1-1 (continued)

NO.	CHANNEL DESCRIPTION	CHECK	CALIBRATE	TEST	PLANT CONDITIONS WHEN REQUIRED
9.	Steam Generator Flow Mismatch	L.13 S(~~22~~) A.20	R	Q(~~1~~) A.11	ALL 1, 2 A.7
10.	Steam Generator Pressure	S(16)	R	Q(1)	ALL < See LCO 3.3.2 >
11.	4KV Bus Undervoltage (A01 & A02)				
	-AFW pump actuation	-	R	M(1)	ALL < See LCO 3.3.2 >
	-Reactor Protection actuation	-	R	M(~~1,2~~) A.11 A.18	ALL 1(e) L.14
12.	4KV Bus Underfrequency (A01 & A02)	M.26 Insert SR 3.3.1.10 Note			
	-to Reactor Coolant Pump trip	-	R	- A.18	ALL 1(e)
13.	Safeguards Bus Voltage				
	-Loss of 4KV	S	R	M	ALL
	-Degraded 4KV	S	R	M	ALL < See LCO 3.3.5 >
	-Loss of 480V	S	R	M	ALL
14.	120 Vac Instrument Buses	W(6)	-	-	ALL < See Section 3.8 >
15.	Reactor Trip Signal From Turbine		L.17 on a staggered test basis	A.11	
	-Turbine Autostop	-	-	M(~~1~~) SR 3.3.1.14 M.19	ALL 1(j) A.8
	-Turbine Stop Valve	-	-	M(~~1~~) SR 3.3.1.14 M.19	ALL
16.	Reactor Trip Signal From SI	-	-	M(~~1~~) M.32 SR 3.3.1.13	ALL 1, 2 A.7
17.	Feedwater Isolation on SI				
	-MFP Trip on Safety Injection	-	-	R	ALL
	-MFRV Shutting on Safety Injection	-	-	R	ALL < See LCO 3.3.2 >
18.	Accumulator Level and Pressure	S	R	-	ALL < See Section 3.5 >
19.	Analog Rod Position	S(8,22)	R	-	ALL < See Section 3.1 >
	-with step counters	S(22)	-	-	ALL
	-Monitoring by On-Line Computer	(18)	-	-	PWR, HOT S/D

D RAI 3.3.1-8

D RAI 3.3.1-21

A.1 TABLE 15.4.1-1 (continued)

NO.	CHANNEL DESCRIPTION	CHECK	CALIBRATE	TEST	PLANT CONDITIONS WHEN REQUIRED	
36.	Radiation Monitoring System					
	- RE-218 WDS Liquid Monitor	(7)	R(14)	Q	ALL	R.1
	- RE-223 Waste Distillate Overboard Monitor	(7)	R(14)	Q	ALL	
	- RE-231 A Steam Line Release Monitor	M(1)	R(14)	-	ALL	< See LCO 3.3.3 >
	- RE-231 B Steam Line Release Monitor	M(1)	R(14)	-	ALL	
	- RE-101 Control Room Monitor	S	R(14)	Q	ALL	< See LCO 3.3.7 >
	- RE-235 Control Room Noble Gas Monitor	S	R(14)	Q	ALL	
	- RE-215 Air Ejector Monitor	D(1)	R(14)	-	ALL	< See LCO 3.4.15 >
37.	Reactor Vessel Fluid Level System	M	R	-	ALL	< See LCO 3.3.3 >
38.	Refueling Water Storage Tank Level	-	R	-	ALL	< See LCO 3.3.3 >
39.	Residual Heat Removal Pump Flow	-	R	-	ALL	LA.3
40.	Safety Valve Position Indicator	M	R	-	ALL	< See LCO 3.3.3 >
41.	Subcooling Margin Monitor	M	R	-	ALL	< See LCO 3.3.3 >
42.	Deleted					
43.	Volume Control Tank Level	-	A	-	ALL	< See LCO 3.4.15 >
44.	Reactor Protection System and Emergency Safety Feature Actuation System Logic < See LCO 3.3.2 >	-	-	Insert SR 3.3.1.5 Notes (L.11) M(~~1~~,23) (A.11) SR 3.3.1.15 (L.11)	~~ALL~~ 1,2,3(a),4(a),5(a) (A.21)	D RAI 3.3 1-26
45.	Reactor Trip System Interlocks					
	-Intermediate Range Neutron Flux, P-6	-	R(24)	R	~~ALL~~ 2(c)	A.22
	-Power Range Neutron Flux, P-8	-	R(24)	R	~~ALL~~ 1	
	-Power Range Neutron Flux, P-9	-	R(24)	R	~~ALL~~ 1	
	-Power Range Neutron Flux, P-10	-	R(24)	R	~~ALL~~ 1,2	
	-1st Stage Turbine Impulse Pressure	-	R~~(24)~~ (A.23)	R	~~ALL~~ 1	
	Insert RCP Breaker Position (M.3)					

RCP Breaker Position Insert

46.	RCP Breaker Position		
	a. Single Loop	SR 3.3.1.13	1(f)
	b. Two loop	SR 3.3.1.13	1(g)

SR 3.3.1.5 Notes Insert

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

1. Not required to be performed for the Source Range Neutron Flux trip function until 8 hours after THERMAL POWER is below P-6.
2. Not required to be performed for the RCP Breaker Position (Two Loops), Reactor Coolant Flow - Low (Two Loops) and Underfrequency Bus A01 and A02 Trip Functions and the P-6, P-7, P-8, P-9 and P-10 Interlocks.

D

RAI 3.3.1-22
RAI TR-2
Errata #46

A.1

NOTATION USED IN TABLE 15.4.1-1

S- Each shift 12 hours L.13
D- Daily
W- Weekly
Q- Quarterly
M- Monthly
P- Prior to reactor criticality if not performed during the previous week. 92 days L. 9
AND four hours after reducing power below P-10 for power and intermediate range instrumentation AND four hours after reducing power below P-6 for source range instrumentation AND every 92 days thereafter. M.27
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.~~ A.24
~~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.~~

D RAI 3.3.1-17

NOTES USED IN TABLE 15.4.1-1

A.11

~~(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. A.13

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

(4) Perform by means of the moveable incore detector system. LA.4

D RAI 3.3.1-18

(5) Recalibrate if the absolute difference is ≥ 3 percent.

(6) Verification of proper breaker alignment and that the 120 Vac instrument buses are energized. < See Section 3.8 >

(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. R.1

(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. < See Section 3.1 >

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

NOTES USED IN TABLE 15.4.1-1 (continued)

(10) When used for the Overpressure Mitigating System, each PORV shall be demonstrated operable by: < See LCO 3.4.12 >

a. Performance of a channel functional test on the PORV actuation channel, but excluding valve operation, within 31 days prior to entering a condition in which the PORV is required operable and at least once per 31 days thereafter when the PORV is required operable.

(11) Performance of a channel functional test is required, excluding valve operation < See LCO 3.4.11 >

(12) Shiftly check is required when the reactor coolant system is not open to the atmosphere and the reactor coolant system temperature is less than the minimum temperature for the in-service pressure test as specified in TS Figure 15.3.1-1. < See LCO 3.4.12 >

(13) An AFW flow path to each steam generator shall be demonstrated operable, following each cold shutdown of greater than 30 days, prior to entering power operation by verifying AFW flow to each steam generator. < See LCO 3.7.5 >

(14) Calibration is to be a verification of response to a source. < See LCO 3.3.3 >

(15) Sample gas for calibration at 2% and 6%. < See LCO 3.3.3 >

< See LCO 3.4.3 >

(16) A check of one pressure channel per steam generator is required whenever the steam generator could be pressurized.

< See LCO 3.3.2 >

(17) Includes test of logic for reactor trip on low-low level, automatic actuation logic for auxiliary feedwater pumps, and test of logic for feedwater isolation on high steam generator level. < See LCO 3.3.2 >

(18) Rod positions must be logged at least once per hour, after a load change >10% or after >30 inches of control rod motion if the on-line computer is inoperable. < See Section 3.1 >

~~(19) The daily heat balance is a gain adjustment performed to match Nuclear Instrumentation System indicated power level with reactor thermal output.~~

~~(20) To confirm that hot channel factor limits are being satisfied, the requirements of TS 15.3.10.E must be met.~~ A.14 L.19 D Errata #90

(21) Check required only when the overpressure mitigation system is in operation. < See LCO 3.4.11 >

~~(22) Not required during period of cold and refueling shutdowns, but must be performed prior to reactor criticality if it has not been performed during the previous surveillance period.~~

A.20

(23) Each train tested at least every 62 days on a staggered basis .

(24) Neutron detectors excluded from calibration.

		Test	Frequency
< See Section 3.4 >			
21.	PORV Block Valves	a. Complete Valve Cycle	Quarterly (13)
		b. Open position check	Every 72 hours (14)
< See Section 5.0 >			
22.	Integrity of Post Accident Recovery Systems Outside Containment	Evaluate	Each refueling cycle
< See Section 3.6 >			
23.	Containment Purge Supply and Exhaust Isolation Valves	Verify valves are locked closed	Monthly (9)
24.	Reactor Trip Breakers	a. Verify independent operability of automatic shunt and undervoltage trip functions.	L.16 → on a staggered test basis Monthly ~~(9)~~ ← A.25 MODE 1, 2, 3(a), 4(a), 5(a) ← A.9
		b. Verify independent operability of manual trip to shunt and undervoltage trip functions.	M.16 Each refueling shutdown ← 18 months MODE 1, 2, 3(a), 4(a), 5(a) ← M.1
25.	Reactor Trip Bypass Breakers	a. Verify operability of the undervoltage trip function.	Prior to breaker use MODE 1(k), 2(k), 3(k), 4(k), 5(k) ← M.24
		~~b. Verify operability of the shunt trip functions.~~	~~Each refueling shutdown~~ ← A.26
		c. Verify operability of the manual trip to undervoltage trip functions.	M.16 Each refueling shutdown ← 18 months MODE 1, 2, 3(a), 4(a), 5(a) ← M.1
26.	120 VAC Vital Instr. Bus Power	Verify Energized(12)	Shiftly ← < See Section 3.8 >
27.	Power Operated Relief Valves (PORVs), PORV Solenoid Air Control Valves, and Air System Check ← < See Section 3.4 >	Operate(16)	Each shutdown(15)
28.	Atmospheric Steam Dumps	Complete valve cycle	Quarterly ← < See Section 3.7 >
29.	Deleted		

D
RAI 3.3.1-24

15.2.3 LIMITING SAFETY SYSTEM SETTINGS, PROTECTIVE INSTRUMENTATION

Applicability:

Applies to trip settings for instruments monitoring reactor power and reactor coolant pressure, temperature, flow, pressurizer level, and permissives related to reactor protection.

Objective:
To provide for automatic protective action in the event that the principal process variables approach a safety limit.

Specification:

1. Protective instrumentation for reactor trip settings shall be as follows:

   A. Startup protection

Table 3.3.1-1, #4 (1) High flux, source range - within span of source range instrumentation.

Table 3.3.1-1, #3 (2) High flux, intermediate range - ≤40% of rated power.

Table 3.3.1-1, #2.b (3) High flux, power range (low setpoint) - ≤25% of rated power.

   B. Core limit protection

Table 3.3.1-1, #2.a (1) High flux, power range (high setpoint) - ≤108% of rated power.

Table 3.3.1-1, #7.b (2) High pressurizer pressure ≤2385 psig for operation at 2250 psia primary system pressure
≤2210 psig for operation at 2000 psia primary system pressure and cores not containing 422V+ fuel assemblies

D
Errata #70

D
Errata #70

Table 3.3.1-1, #7.a

(3) Low pressurizer pressure - ≥1905 psig for operation at 2250 psia primary system pressure
≥1800 psig for operation at 2000 psia primary system pressure and cores not containing 422V+ fuel assemblies

D
Errata #70

Table 3.3.1-1, #5

(4) Overtemperature

$$\Delta T \left(\frac{1}{1+\tau_3 S}\right) \leq \Delta T_o \left(K_1 - K_2\left(T\left(\frac{1}{1+\tau_4 S}\right) - T'\right)\left(\frac{1+\tau_1 S}{1+\tau_2 S}\right) + K_3(P - P') - f(\Delta I)\right)$$

where (values are applicable to operation at both 2000 psia and 2250 psia unless otherwise indicated)

LA.5 → [*]

ΔT_o	=	indicated ΔT at rated power, °F	
T	=	average temperature, °F	
T'	≤	569.0°F	(for cores containing 422V+ fuel assemblies)
T'	≤	572.9°F	(for cores not containing 422V+ fuel assemblies)
P	=	pressurizer pressure, psig	
P'	=	2235 psig	(for 2250 psia operation)
P'	=	1985 psig	(for 2000 psia operation and cores not containing 422V+ fuel assemblies)
K_1	≤	1.16	(for 2250 psia operation and cores containing 422V+ fuel assemblies)
K_1	≤	1.19	(for 2250 psia operation and cores not containing 422V+ fuel assemblies)
K_1	≤	1.14	(for 2000 psia operation and cores not containing 422V+ fuel assemblies)
K_2	=	0.0149	(for 2250 psia operation and cores containing 422V+ fuel assemblies)
K_2	=	0.025	(for 2250 psia operation and cores not containing 422V+ fuel assemblies)
K_2	=	0.022	(for 2000 psia operation and cores not containing 422V+ fuel assemblies)
K_3	=	0.00072	(for 2250 psia operation and cores containing 422V+ fuel assemblies)
K_3	=	0.0013	(for 2250 psia operation and cores not containing 422V+ fuel assemblies)
K_3	=	0.001	(for 2000 psia operation and cores not containing 422V+ fuel assemblies)
τ_1	=	25 sec	
τ_2	=	3 sec	
τ_3	=	2 sec	for Rosemont or equivalent RTD
	=	0 sec	for Sostman or equivalent RTD
τ_4	=	2 sec	for Rosemont or equivalent RTD
	=	0 sec	for Sostman or equivalent RTD

and $f(\Delta I)$ is an even function of the indicated difference between top and bottom detectors of the power-range nuclear ion chambers; with gains to be selected based on measured instrument response during plant startup tests, where q_t and q_b are the percent power in the top and bottom halves of the core respectively, and $q_t + q_b$ is total core power in percent of rated power, such that:

[*] ← LA.5

(a) for $q_t - q_b$ within −17 , +5 percent, $f(\Delta I) = 0$ for cores not containing 422V+ fuel assemblies;
for $q_t - q_b$ within −12, +5 percent, $f(\Delta I) = 0$ for cores containing 422V+ fuel assemblies.

(b) for each percent that the magnitude of q_t - q_b exceeds +5 percent, the ΔT trip setpoint shall be automatically reduced by an equivalent of 2.0 ~~percent of rated power for cores not containing 422V+ fuel assemblies~~ and reduced by an equivalent of 2.12 ~~percent of rated power for cores containing 422V+ fuel assemblies.~~

(c) for cores not containing 422V+ fuel assemblies, for each percent that the magnitude of q_t - q_b exceeds −17 percent, the ΔT trip setpoint shall be automatically reduced by an equivalent of 2.0 ~~percent of rated power; for cores containing 422V+ fuel assemblies, for each percent that the magnitude of q_t - q_b exceeds −12 percent, the ΔT trip setpoint shall be automatically reduced by an equivalent of 2.0 percent of rated power.~~

The values denoted with [*] are specified in the COLR. LA.5

[*]

LA.5

(5) Overpower

$$\Delta T\left(\frac{1}{1+\tau_3 S}\right) \le \Delta T_o\left[K_4 - K_5\left(\frac{\tau_5 S}{\tau_5 S+1}\right)\left(\frac{1}{1+\tau_4 S}\right)T - K_6\left[T\left(\frac{1}{1+\tau_4 S}\right) - T'\right]\right]$$

Table 3.3.1-1, #6

where (values are applicable to operation at both 2000 psia and 2250 psia)

ΔT_o	=		indicated ΔT at rated power, °F
T	=		average temperature, °F
T'	≤	569.0°F	(for cores containing 422V+ fuel assemblies)
T'	≤	572.9°F	(for cores not containing 422V+ fuel assemblies)
K_4	≤	1.10	of rated power (for cores containing 422V+ fuel assemblies)
K_4	≤	1.09	of rated power (for cores not containing 422V+ fuel assemblies)
K_5	=	0.0262	for increasing T
	=	0.0	for decreasing T
K_6	=	0.00103	for T ≥ T' (for cores containing 422V+ fuel assemblies)
K_6	=	0.00123	for T ≥ T' (for cores not containing 422V+ fuel assemblies)
	=	0.0	for T < T'
τ_5	=	10 sec	
τ_3	=	2 sec	for Rosemont or equivalent RTD
		0 sec	for Sostman or equivalent RTD
τ_4	=	2 sec	for Rosemont or equivalent RTD
		0 sec	for Sostman or equivalent RTD

LA.5

The values denoted with [*] are specified in the COLR.

(6) Undervoltage - ≥3120V Table 3.3.1-1, #11

(7) Indicated reactor coolant flow per loop ≥90 percent of normal indicated loop flow Table 3.3.1-1, #9.a, 9.b

(8) Reactor coolant pump motor breaker open Table 3.3.1-1, #12

(a) Low frequency set point ≥55.0 HZ

~~(b) Low voltage set point ≥3120V~~ L.18

D

Errata #26
Errata #70

C. Other reactor trips:

Errata #70

Table 3.3.1-1, #8

(1) High pressurizer water level - ≤95% of span

(2) Low-low steam generator water level - ≥20% of narrow range instrument span

Table 3.3.1-1, #13

Errata #70

(3) Steam-Feedwater Flow Mismatch Trip - ≤1.0 x 10^6 lb/hr

Table 3.3.1-1, #14

(4) Turbine Trip (Not a protection circuit)

Table 3.3.1-1, #15.a, 15.b

(5) Safety Injection Signal

Table 3.3.1-1, #16

(6) Manual Trip

Table 3.3.1-1, #1

(7) Steam Generator Water Level –Low field setting: ≥ 30% of narrow range instrument span (nominal)

M.31

RAI 3.3.1-26

Autostop Oil Pressure - Low field setting: ≥ 45 psig (nominal)

M.31

RAI 3.3.1-27

RAI 3.3.1-26

2. Protective instrumentation settings for reactor trip interlocks shall be as follows:

A. The "at power" reactor trips (low pressurizer pressure, high pressurizer level, and low reactor coolant flow for both loops) shall be unblocked when:

Table 3.3.1-1, #17.b.1

Table 3.3.1-1, #17.b.2

(1) Power range nuclear flux ~~≥9% (±1%)~~ ≥ 10% of rated power, or

(2) Turbine Load ≥10% of full load turbine pressure .

A.31

D

RAI 3.3.1-4
RAI 3.3.1-14

B. The single loss of flow trip shall be unblocked when the power range nuclear flux ≥50% of rated power.

Table 3.3.1-1, #17.c

C. The power range high flux level low range trip, and intermediate range high flux level trip shall be unblocked when power is ≤9% (±1%) of rated power.

Table 3.3.1-1, #17.e

D. The source range high flux reactor trip shall be unblocked when the intermediate range flux is $\leq 10^{-10}$ amperes.

A.2

D

Errata #70

Basis

The source range high flux reactor trip prevents a startup accident from subcritical conditions from proceeding into the power range. Any setpoint within its range would prevent an excursion from proceeding to the point at which significant thermal power is generated.[1]

The high flux low power reactor trip provides redundant protection in the power range for a power excursion beginning from low power. This trip insures that a more restrictive trip point is used for this case than for an excursion beginning from near full power.[1]

The overpower nuclear flux reactor trip protects the reactor core against reactivity excursions which are too rapid to be protected by temperature and pressure circuitry. The prescribed setpoint, with allowance for errors, is consistent with the trip point assumed in the accident analysis.[3]

The overpower ΔT reactor trip prevents power density anywhere in the core from exceeding 118% of design power density, and includes corrections for change in density and heat capacity of water with temperature, and dynamic compensation for piping delays from the core to the loop temperature detectors. The specified setpoints meet this requirement and include allowance for instrument errors.[2]

The overtemperature ΔT reactor trip provides core protection against DNB for all combinations of pressure, power, coolant temperature, and axial power distribution, provided only that (1) the transient is slow with respect to piping transit delays from the core to the temperature detectors (about 4 seconds)[5], and (2) pressure is within the range between the high and low pressure reactor trips. With normal axial power distribution, the reactor trip limit, with allowance for errors[2], is always below the core safety limit as shown on Figures 15.2.1-1 and 15.2.1-2. If axial peaks are greater than design, as indicated by the difference between top and bottom power range nuclear detectors, the reactor trip limit is automatically reduced[6][7].

The overpower, overtemperature and pressurizer pressure system setpoints for OFA and Upgraded OFA fuel include the effect of reduced system pressure operation (including the effects of fuel densification). The setpoints for 422V+ fuel do not include the effect of reduced system pressure operation; therefore, cores containing 422V+ fuel must be operated at 2250 psia. The setpoints will not exceed the core safety limits as shown in Figures 15.2.1-1 (for OFA and Upgraded OFA fuel only cores) and 15.2.1-2 (for cores containing 422V+ fuel).

D
Errata #70

The overpower limit criteria is that core power be prevented from reaching a value at which fuel pellet centerline melting would occur. The reactor is prevented from reaching the overpower limit condition by action of the nuclear overpower and overpower ΔT trips.

The high and low pressure reactor trips limit the pressure range in which reactor operation is permitted. The high pressurizer pressure reactor trip setting is lower than the set pressure for the safety valves (2485 psig) such that the reactor is tripped before the safety valves actuate. The low pressurizer pressure reactor trip trips the reactor in the unlikely event of a loss-of-coolant accident[(4)].

The low flow reactor trip protects the core against DNB in the event of either a decreasing actual measured flow in the loops or a sudden loss of power to one or both reactor coolant pumps. The setpoint specified is consistent with the value used in the accident analysis[(8)]. The low loop flow signal is caused by a condition of less than 90 percent flow as measured by the loop flow instrumentation. The loss of power signal is caused by the reactor coolant pump breaker opening as actuated by either high current, low supply voltage or low electrical frequency, or by a manual control switch. The significant feature of the breaker trip is the frequency setpoint, 55.0 HZ, which assures a trip signal before the pump inertia is reduced to an unacceptable value. The high pressurizer water level reactor trip protects the pressurizer safety valves against water relief. The specified setpoint allows adequate operating instrument error[(2)] and transient overshoot in level before the reactor trips.

The low-low steam generator water level reactor trip protects against loss of feedwater flow accidents. The specified setpoint assures that there will be sufficient water inventory in the steam generators at the time of trip to allow for starting delays for the auxiliary feedwater system.[(9)]

Numerous reactor trips are blocked at low power where they are not required for protection and would otherwise interfere with normal plant operations. The prescribed setpoint above which these trips are unblocked assures their availability in the power range where needed. Specifications 15.2.3.2.A(1) and 15.2.3.2.C have $\pm 1\%$ tolerance to allow for a 2% deadband of the P10 bistable which is used to set the limit of both items. The difference between the nominal and maximum allowed value (or minimum allowed value) is to account for "as measured" rack drift effects. Sustained power operation is not be permitted with only one reactor coolant pump. If a pump is lost while operating below 50 percent power, an orderly shutdown is allowed. The power-to-flow ratio will be maintained equal to or less than unity, which ensures that the minimum DNB ratio increases at lower flow because the maximum enthalpy rise does not increase above the maximum enthalpy rise which occurs during full power and full flow operation.

A.2

References

(1) FSAR 14.1.1	(4) FSAR 14.3.1	(7) FSAR 3.2.1
(2) FSAR 14.0	(5) FSAR 14.1.2	(8) FSAR 14.1.8
(3) FSAR 14.2.6	(6) FSAR 7.2, 7.7	(9) FSAR 14.1.10 and 14.1.11

D

Errata #70

Justification For Deviations - NUREG-1431 Section 3.03.01

30-Jan-01

JFD Number	JFD Text	
01 Rev. D	Not used.	
	ITS:	**NUREG:**
	N/A	N/A
02 Rev. D	Not used.	
	ITS:	**NUREG:**
	N/A	N/A
03 Rev. A	LCO 3.3.1 Bases discussion of the Turbine Trip-Turbine Stop Valve Closure LSSS is replaced with a statement explaining that no analytical value exists in the current licensing basis for this function in the mitigation of analyzed accidents.	
	ITS:	**NUREG:**
	B 3.03.01	B 3.03.01
04 Rev. A	ITS LCO 3.3.1, Condition C and Required Action C.1 references to "trains" have been deleted. Trip functions involving "trains" of instrumentation, that previously referred to Condition C, have been revised to refer to other Conditions. Condition C provides Required Actions exclusively for a loss of a required Manual Reactor Trip channel in MODES 3, 4 and 5 with the RTBs closed and the Rod Control System capable of rod withdrawal. Therefore the restoration of "trains" of instrumentation no longer applies to this Condition. Additionally, TSTF-135 changes to Condition C have not been incorporated into the ITS. TSTF-135 provides an alternative action to opening the RTBs, by relocating the specific method to the Bases. TSTF-135 states this change is necessary to eliminate undesirable secondary effects of opening RTBs related to the P-4 interlock. Point Beach design does not include a P-4 interlock that would be tripped when the RTBs are open, leading to the isolation of normal feedwater. Therefore not specifying the method of precluding rod withdrawal is not justified.	
	ITS:	**NUREG:**
	LCO 3.03.01 COND C	LCO 3.03.01 COND C
	LCO 3.03.01 COND C RA C.1	LCO 3.03.01 COND C RA C.1
05 Rev. D	Not used.	
	ITS:	**NUREG:**
	N/A	N/A

Justification For Deviations - NUREG-1431 Section 3.03.01

30-Jan-01

JFD Number	JFD Text	
06 Rev. D	Not used.	
	ITS:	**NUREG:**
	N/A	N/A
07 Rev. A	Required Actions D.1.2, D.2.1 and D.2.2 and the Note which states, "Only required to be performed when the power range neutron flux input to QPTR is inoperable", are duplicates of SR 3.2.4.1 and SR 3.2.4.2. As such, these Required Actions are unnecessary and inconsistent with the rest of the NUREG format. Deleting these Required Actions causes Condition E to be duplicative of Condition D. Functions in Table 3.3.1-1 which previously referred to Condition E, now refer to Condition D. Additionally, "Power Range Neutron Flux-High" has been deleted from the entry Condition to reflect use of this Condition for inoperable channels associated with several Trip Functions, as identified in Table 3.3.1-1.	
	ITS:	**NUREG:**
	B 3.03.01	B 3.03.01
	LCO 3.03.01 COND D RA D.1	LCO 3.03.01 COND D RA D.1.1
	LCO 3.03.01 COND D RA D.2	LCO 3.03.01 COND D RA D.3
	LCO 3.03.01 T3.03.01-01 02B	LCO 3.03.01 T3.03.01-01 02B
	LCO 3.03.01 T3.03.01-01 05	LCO 3.03.01 T3.03.01-01 06
	LCO 3.03.01 T3.03.01-01 06	LCO 3.03.01 T3.03.01-01 07
	LCO 3.03.01 T3.03.01-01 07B	LCO 3.03.01 T3.03.01-01 08B
	LCO 3.03.01 T3.03.01-01 13	LCO 3.03.01 T3.03.01-01 14
	LCO 3.03.01 T3.03.01-01 14-01	LCO 3.03.01 T3.03.01-01 15-01
	LCO 3.03.01 T3.03.01-01 14-02	LCO 3.03.01 T3.03.01-01 15-02
	N/A	LCO 3.03.01 COND D RA D.1.2 LCO 3.03.01 COND D RA D.2.1 LCO 3.03.01 COND D RA D.2.2 LCO 3.03.01 COND D RA D.2.2 NOTE
08 Rev. A	SR 3.3.1.10 is modified by a Note requiring the surveillance to include verification that the time "constants" are adjusted to prescribed values. "Constants" has been changed to "delays" to reflect Point Beach nomenclature used for these devices.	
	ITS:	**NUREG:**
	B 3.03.01	B 3.03.01
	SR 3.03.01.10 NOTE	SR 3.03.01.10 NOTE

JFD Number	JFD Text	
09 Rev. D	Not used.	
	ITS:	**NUREG:**
	N/A	N/A
10 Rev. D	Not used.	
	ITS:	**NUREG:**
	N/A	N/A
11 Rev. D	Not used.	
	ITS:	**NUREG:**
	N/A	N/A
12 Rev. D	Not used.	
	ITS:	**NUREG:**
	N/A	N/A
13 Rev. D	Condition K addresses the loss of one or more RCP Breaker Position - Single Loop trip channels in MODE 1 above P-8. Point Beach currently requires both channels be operable. Therefore the Condition is modified to provide Required Actions and Completion Times commensurate with a loss of function. A Completion Time of one hour is given to restore the channel(s) to operable status or reduce power to < P-8 in 5 hours, thereby placing the unit in a MODE where the function is no longer required to be operable. Additionally, the NUREG Note allowing an inoperable channel to be bypassed for surveillance testing of other channels, is deleted, because both channels are required to be OPERABLE and extended operation with one channel in bypass would not be permitted.	
	ITS:	**NUREG:**
	B 3.03.01	B 3.03.01
	LCO 3.03.01 COND M RA M.1	LCO 3.03.01 COND O RA O.1
	N/A	LCO 3.03.01 COND O RA O.1 NOTE

JFD Number	JFD Text	
14 Rev. D	Condition N and associated Required Actions are added to address the RCP Breaker Position - Two Loops function. A loss of one channel in this function requires restoration of the function within one hour. If restoration cannot be completed in one hour, THERMAL POWER is required to be reduced to < P-7 in the next 6 hours. This places the unit in a MODE where the LCO is no longer applicable. The Completion Time of 6 hours is reasonable, based on operating experience, to reduce THERMAL POWER to below P-7 from full power in an orderly manner without challenging unit systems.	
	ITS:	**NUREG:**
	B 3.03.01	B 3.03.01
	LCO 3.03.01 COND N	N/A
	LCO 3.03.01 COND N RA N.1	N/A
	LCO 3.03.01 COND N RA N.2	N/A
	LCO 3.03.01 T3.03.01-01 10B	LCO 3.03.01 T3.03.01-01 11B
15 Rev. A	ITS LCO 3.3.1, Table 3.3.1-1, "Applicable Modes or Other Specified Conditions" column has been relabeled "Applicable Modes", to alleviate confusion with the usage of "Conditions" in another column in the Table.	
	ITS:	**NUREG:**
	LCO 3.03.01 T3.03.01-01	LCO 3.03.01 T3.03.01-01

JFD Number	JFD Text	
16 Rev. A	The brackets have been removed and the proper plant specific information has been provided. Additionally, text added to the Bases description of SR 3.3.1.8 and SR 3.3.1.13, via TSTF-205, has not been incorporated into the ITS. Point Beach design necessitates COT testing that is inconsistent with the verbiage added by TSTF-205.	
	ITS:	**NUREG:**
	B 3.03.01	B 3.03.01
	LCO 3.03.01 COND P RA P.1 NOTE	LCO 3.03.01 COND Q RA Q.1 NOTE
	LCO 3.03.01 T3.03.01-01 02A	LCO 3.03.01 T3.03.01-01 02A
	LCO 3.03.01 T3.03.01-01 02B	LCO 3.03.01 T3.03.01-01 02B
	LCO 3.03.01 T3.03.01-01 03	LCO 3.03.01 T3.03.01-01 04-01
	LCO 3.03.01 T3.03.01-01 04-01	LCO 3.03.01 T3.03.01-01 05-01
	LCO 3.03.01 T3.03.01-01 04-02	LCO 3.03.01 T3.03.01-01 05-02
	LCO 3.03.01 T3.03.01-01 05	LCO 3.03.01 T3.03.01-01 06
	LCO 3.03.01 T3.03.01-01 06	LCO 3.03.01 T3.03.01-01 07
	LCO 3.03.01 T3.03.01-01 07A	LCO 3.03.01 T3.03.01-01 08A
	LCO 3.03.01 T3.03.01-01 07B	LCO 3.03.01 T3.03.01-01 08B
	LCO 3.03.01 T3.03.01-01 08	LCO 3.03.01 T3.03.01-01 09
	LCO 3.03.01 T3.03.01-01 09A	LCO 3.03.01 T3.03.01-01 10A
	LCO 3.03.01 T3.03.01-01 09B	LCO 3.03.01 T3.03.01-01 10B
	LCO 3.03.01 T3.03.01-01 11	LCO 3.03.01 T3.03.01-01 12
	LCO 3.03.01 T3.03.01-01 12	LCO 3.03.01 T3.03.01-01 13
	LCO 3.03.01 T3.03.01-01 13	LCO 3.03.01 T3.03.01-01 14
	LCO 3.03.01 T3.03.01-01 14-01	LCO 3.03.01 T3.03.01-01 15-01
	LCO 3.03.01 T3.03.01-01 14-02	LCO 3.03.01 T3.03.01-01 15-02
	LCO 3.03.01 T3.03.01-01 15A	LCO 3.03.01 T3.03.01-01 16A
	LCO 3.03.01 T3.03.01-01 15B	LCO 3.03.01 T3.03.01-01 16B
	LCO 3.03.01 T3.03.01-01 17A	LCO 3.03.01 T3.03.01-01 18A
	LCO 3.03.01 T3.03.01-01 17B-02	LCO 3.03.01 T3.03.01-01 18F
	LCO 3.03.01 T3.03.01-01 17C	LCO 3.03.01 T3.03.01-01 18C
	LCO 3.03.01 T3.03.01-01 17E	LCO 3.03.01 T3.03.01-01 18E

JFD Number	JFD Text	
	LCO 3.03.01 T3.03.01-01 NOTE (h)	N/A
	LCO 3.03.01 T3.03.01-01 NOTE (i)	N/A
	SR 3.03.01.02 NOTE 2	SR 3.03.01.02 NOTE 2
	SR 3.03.01.06	SR 3.03.01.06
	SR 3.03.01.07	SR 3.03.01.07
	SR 3.03.01.08 NOTE	SR 3.03.01.08 NOTE
	SR 3.03.01.09	SR 3.03.01.09
	SR 3.03.01.10	SR 3.03.01.10
	SR 3.03.01.11	SR 3.03.01.11
	SR 3.03.01.13	SR 3.03.01.14
17 Rev. D	The Notes modifying the Required Actions of ITS LCO 3.3.1, Condition Q have been revised to allow an RTB to be bypassed for up to 8 hours, provided the other train is OPERABLE. This is consistent with the Point Beach current licensing basis.	
	ITS:	**NUREG:**
	B 3.03.01	B 3.03.01
	LCO 3.03.01 COND Q RA Q.1 NOTE	LCO 3.03.01 COND R RA R.1 NOTE 1
	N/A	LCO 3.03.01 COND R RA R.1 NOTE 2
18 Rev. A	LCO 3.3.1 Bases discussions regarding applicability of trip functions in MODES 3, 4 and 5 when the RTBs are closed and the CRD System is capable of rod withdrawal, have been modified to "... the Rod Control System is capable of rod withdrawal" to be consistent with ITS LCO 3.3.1, Table 3.3.1-1, Note (a),which modifies MODES 3. 4 and 5.	
	ITS:	**NUREG:**
	B 3.03.01	B 3.03.01
19 Rev. D	Not used.	
	ITS:	**NUREG:**
	N/A	N/A

JFD Number	JFD Text	
20 Rev. D	Condition T is added to provide Required Actions for an inoperable RTB or undervoltage / shunt trip mechanism in MODES 3, 4 and 5 with the RTBs closed and the Rod Control System capable of rod withdrawal. NUREG 1431 Condition C previously addressed the above inoperabilities and provided actions for a loss of a Manual Reactor Trip channel, but was revised to exclusively address the loss of a Manual Reactor Trip channel.	
	ITS:	**NUREG:**
	B 3.03.01	B 3.03.01
	LCO 3.03.01 COND T	LCO 3.03.01 COND C
	LCO 3.03.01 COND T RA T.1	LCO 3.03.01 COND C RA C.1
	LCO 3.03.01 COND T RA T.2	LCO 3.03.01 COND C RA C.2
	LCO 3.03.01 T3.03.01-01 18-02	LCO 3.03.01 T3.03.01-01 19-02
	LCO 3.03.01 T3.03.01-01 19-02	LCO 3.03.01 T3.03.01-01 20-02
21 Rev. D	SR 3.31.5 has been modified by the addition of two Notes. Note 1 allows a delay in the performance of the surveillance for the Source Range Neutron Flux trip function for up to 8 hours after power is reduced below P-6. Note 2 allows an exception to the performance of this surveillance for the RCP Breaker Position (Two Loops), Reactor Coolant Flow – Low (Two Loops) and Underfrequency Bus A01 and A02 Trip Functions and the P-6, P-7, P-8, P-9 and P-10 Interlocks. The addition of Note 1 is necessary due to the adoption of SR 3.0.4, which unlike CTS 15.4.0.4, does not allow entry into a condition where a system or component is required to be operable, when such entry is necessary to perform the specified surveillance. Per the Bases of SR 3.0.4, this allowance should be specified in the frequency as 'not due until the specific conditions needed are met,' or alternately, in the form of a note as 'not required to be performed until a particular event, condition, or time has been reached.' Note 2 has been provided to exclude the testing of actuation logic associated with functions tested at an 18 month frequency via SR 3.3.1.15, consistent with CLB.	
	ITS:	**NUREG:**
	SR 3.03.01.05 NOTE 1	N/A
	SR 3.03.01.05 NOTE 2	N/A
22 Rev. A	LCO 3.3.1 Bases discussions regarding trip function signals preventing automatic or manual rod withdrawal have not been retained in ITS. Rod Stops do not provide a safety function at Point Beach and do not need to be discussed as part of the RPS Bases.	
	ITS:	**NUREG:**
	B 3.03.01	B 3.03.01

Justification For Deviations - NUREG-1431 Section 3.03.01

30-Jan-01

JFD Number	JFD Text	
23 Rev. A	NUREG 1431, SR 3.3.1.8 frequency is modified by inserting "range" between "intermediate" and "instrumentation", to more clearly stipulate the requirement.	
	ITS:	**NUREG:**
	SR 3.03.01.08	SR 3.03.01.08
24 Rev. A	The ITS definition of TADOT has been modified to not include verification of the setpoint. Therefore it is no longer necessary to exclude this verification from SR 3.3.1.9, SR 3.3.1.14 and SR 3.3.1.15, resulting in deletion of the Note from each of these SRs. Additionally, text added to the Bases description of the above SRs, via TSTF-205, has not been incorporated into the ITS. Point Beach design of RPS necessitates TADOT testing which is inconsistent with the verbiage added by TSTF-205.	
	ITS:	**NUREG:**
	B 3.03.01	B 3.03.01
	N/A	SR 3.03.01.09 NOTE SR 3.03.01.14 NOTE SR 3.03.01.15 NOTE

JFD Number	JFD Text
25 Rev. A	Point Beach current licensing basis does not include the performance of Reactor Protection System component Response Time Testing. Therefore, NUREG 1431, SR 3.3.1.16 is not being retained in the ITS. Additionally, text added to the Bases description of SR 3.3.1.16, via TSTF-111, has not been incorporated.

ITS:	**NUREG:**
B 3.03.01	B 3.03.01
LCO 3.03.01 T3.03.01-01 02A	LCO 3.03.01 T3.03.01-01 02A
LCO 3.03.01 T3.03.01-01 02B	LCO 3.03.01 T3.03.01-01 02B
LCO 3.03.01 T3.03.01-01 04-01	LCO 3.03.01 T3.03.01-01 05-01
LCO 3.03.01 T3.03.01-01 04-02	LCO 3.03.01 T3.03.01-01 05-02
LCO 3.03.01 T3.03.01-01 05	LCO 3.03.01 T3.03.01-01 06
LCO 3.03.01 T3.03.01-01 06	LCO 3.03.01 T3.03.01-01 07
LCO 3.03.01 T3.03.01-01 07A	LCO 3.03.01 T3.03.01-01 08A
LCO 3.03.01 T3.03.01-01 07B	LCO 3.03.01 T3.03.01-01 08B
LCO 3.03.01 T3.03.01-01 09A	LCO 3.03.01 T3.03.01-01 10A
LCO 3.03.01 T3.03.01-01 09B	LCO 3.03.01 T3.03.01-01 10B
LCO 3.03.01 T3.03.01-01 11	LCO 3.03.01 T3.03.01-01 12
LCO 3.03.01 T3.03.01-01 12	LCO 3.03.01 T3.03.01-01 13
LCO 3.03.01 T3.03.01-01 13	LCO 3.03.01 T3.03.01-01 14
LCO 3.03.01 T3.03.01-01 14-01	LCO 3.03.01 T3.03.01-01 15-01
LCO 3.03.01 T3.03.01-01 14-02	LCO 3.03.01 T3.03.01-01 15-02
N/A	SR 3.03.01.16 SR 3.03.01.16 NOTE

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30-Jan-01

JFD Number	JFD Text	
26 Rev. A	Reviewer's Note (a) in Table 3.3.1-1 is not being retained in ITS, resulting in the re-lettering of all subsequent Notes.	
	ITS:	**NUREG:**
	LCO 3.03.01 T3.03.01-01 01-02	LCO 3.03.01 T3.03.01-01 01-02
	LCO 3.03.01 T3.03.01-01 02B	LCO 3.03.01 T3.03.01-01 02B
	LCO 3.03.01 T3.03.01-01 03	LCO 3.03.01 T3.03.01-01 04-01
	LCO 3.03.01 T3.03.01-01 04-01	LCO 3.03.01 T3.03.01-01 05-01
	LCO 3.03.01 T3.03.01-01 04-02	LCO 3.03.01 T3.03.01-01 05-02
	LCO 3.03.01 T3.03.01-01 07A	LCO 3.03.01 T3.03.01-01 08A
	LCO 3.03.01 T3.03.01-01 08	LCO 3.03.01 T3.03.01-01 09
	LCO 3.03.01 T3.03.01-01 09A	LCO 3.03.01 T3.03.01-01 10A
	LCO 3.03.01 T3.03.01-01 10A	LCO 3.03.01 T3.03.01-01 11A
	LCO 3.03.01 T3.03.01-01 10B	LCO 3.03.01 T3.03.01-01 11B
	LCO 3.03.01 T3.03.01-01 11	LCO 3.03.01 T3.03.01-01 12
	LCO 3.03.01 T3.03.01-01 12	LCO 3.03.01 T3.03.01-01 13
	LCO 3.03.01 T3.03.01-01 15A	LCO 3.03.01 T3.03.01-01 16A
	LCO 3.03.01 T3.03.01-01 15B	LCO 3.03.01 T3.03.01-01 16B
	LCO 3.03.01 T3.03.01-01 17A	LCO 3.03.01 T3.03.01-01 18A
	LCO 3.03.01 T3.03.01-01 18-02	LCO 3.03.01 T3.03.01-01 19-02
	LCO 3.03.01 T3.03.01-01 19-02	LCO 3.03.01 T3.03.01-01 20-02
	LCO 3.03.01 T3.03.01-01 NOTE (a)	LCO 3.03.01 T3.03.01-01 NOTE (b)
	LCO 3.03.01 T3.03.01-01 NOTE (b)	LCO 3.03.01 T3.03.01-01 NOTE (C)
	LCO 3.03.01 T3.03.01-01 NOTE (C)	LCO 3.03.01 T3.03.01-01 NOTE (d)
	LCO 3.03.01 T3.03.01-01 NOTE (d)	LCO 3.03.01 T3.03.01-01 NOTE (e)
	LCO 3.03.01 T3.03.01-01 NOTE (e)	LCO 3.03.01 T3.03.01-01 NOTE (g)
	LCO 3.03.01 T3.03.01-01 NOTE (f)	LCO 3.03.01 T3.03.01-01 NOTE (h)
	LCO 3.03.01 T3.03.01-01 NOTE (g)	LCO 3.03.01 T3.03.01-01 NOTE (i)
	N/A	LCO 3.03.01 T3.03.01-01 NOTE (a)

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30-Jan-01

JFD Number	JFD Text	
27 Rev. A	The "Trip Setpoint" column in Table 3.3.1-1 is being eliminated. The setpoint methodology at Point Beach uses Allowable Values derived from the analytical limits contained in the safety analysis.	
	ITS:	**NUREG:**
	B 3.03.01	B 3.03.01
	LCO 3.03.01 T3.03.01-01	LCO 3.03.01 T3.03.01-01
28 Rev. D	The Bases have been revised to reflect Point Beach Nuclear Plant design. Point Beach Manual Reactor Trip design utilizes four switches in two channels. Each channel is comprised of two switches (one in each train).	
	ITS:	**NUREG:**
	B 3.03.01	B 3.03.01
	LCO 3.03.01 T3.03.01-01 01-01	LCO 3.03.01 T3.03.01-01 01-01
	LCO 3.03.01 T3.03.01-01 01-02	LCO 3.03.01 T3.03.01-01 01-02

JFD Number	JFD Text	
29 Rev. A	The Power Range Neutron Flux Rate function is not being retained in ITS. Point Beach current licensing basis does not include this function as a RPS trip. Deletion of this function results in the re-numbering of subsequent functions.	
	ITS:	**NUREG:**
	B 3.03.01	B 3.03.01
	LCO 3.03.01 T3.03.01-01 03	LCO 3.03.01 T3.03.01-01 04-01
	LCO 3.03.01 T3.03.01-01 04-01	LCO 3.03.01 T3.03.01-01 05-01
	LCO 3.03.01 T3.03.01-01 04-02	LCO 3.03.01 T3.03.01-01 05-02
	LCO 3.03.01 T3.03.01-01 05	LCO 3.03.01 T3.03.01-01 06
	LCO 3.03.01 T3.03.01-01 06	LCO 3.03.01 T3.03.01-01 07
	LCO 3.03.01 T3.03.01-01 07A	LCO 3.03.01 T3.03.01-01 08A
	LCO 3.03.01 T3.03.01-01 07B	LCO 3.03.01 T3.03.01-01 08B
	LCO 3.03.01 T3.03.01-01 08	LCO 3.03.01 T3.03.01-01 09
	LCO 3.03.01 T3.03.01-01 09A	LCO 3.03.01 T3.03.01-01 10A
	LCO 3.03.01 T3.03.01-01 09B	LCO 3.03.01 T3.03.01-01 10B
	LCO 3.03.01 T3.03.01-01 10A	LCO 3.03.01 T3.03.01-01 11A
	LCO 3.03.01 T3.03.01-01 10B	LCO 3.03.01 T3.03.01-01 11B
	LCO 3.03.01 T3.03.01-01 11	LCO 3.03.01 T3.03.01-01 12
	LCO 3.03.01 T3.03.01-01 12	LCO 3.03.01 T3.03.01-01 13
	LCO 3.03.01 T3.03.01-01 13	LCO 3.03.01 T3.03.01-01 14
	LCO 3.03.01 T3.03.01-01 14-01	LCO 3.03.01 T3.03.01-01 15-01
	LCO 3.03.01 T3.03.01-01 14-02	LCO 3.03.01 T3.03.01-01 15-02
	LCO 3.03.01 T3.03.01-01 15A	LCO 3.03.01 T3.03.01-01 16A
	LCO 3.03.01 T3.03.01-01 15B	LCO 3.03.01 T3.03.01-01 16B
	LCO 3.03.01 T3.03.01-01 16	LCO 3.03.01 T3.03.01-01 17
	LCO 3.03.01 T3.03.01-01 17A	LCO 3.03.01 T3.03.01-01 18A
	LCO 3.03.01 T3.03.01-01 17C	LCO 3.03.01 T3.03.01-01 18C
	LCO 3.03.01 T3.03.01-01 17D	LCO 3.03.01 T3.03.01-01 18D
	LCO 3.03.01 T3.03.01-01 17E	LCO 3.03.01 T3.03.01-01 18E
	LCO 3.03.01 T3.03.01-01 18-01	LCO 3.03.01 T3.03.01-01 19-01

JFD Number	JFD Text	
	LCO 3.03.01 T3.03.01-01 18-02	LCO 3.03.01 T3.03.01-01 19-02
	LCO 3.03.01 T3.03.01-01 19-01	LCO 3.03.01 T3.03.01-01 20-01
	LCO 3.03.01 T3.03.01-01 19-02	LCO 3.03.01 T3.03.01-01 20-02
	N/A	LCO 3.03.01 T3.03.01-01 03A LCO 3.03.01 T3.03.01-01 03B
30 Rev. D	Not used.	
	ITS:	**NUREG:**
	N/A	N/A
31 Rev. D	Not used.	
	ITS:	**NUREG:**
	N/A	N/A
32 Rev. D	The Channel Calibration surveillance requirement for Pressurizer Pressure - Low, Pressurizer Pressure - High, Pressurizer Water Level - High, Reactor Coolant Flow - Low (Single Loop), Reactor Coolant Flow - Low (Two Loop), SG Water Level - Low Low, Steam Flow / Feed Flow Mismatch, SG Water Level - Low and the Turbine Impulse Pressure Reactor Protection System Interlock, have been changed. These functions do not have time constants associated with them at Point Beach. Therefore the surveillance requirement for each of the above functions has been changed from SR 3.3.1.10 to SR 3.3.1.11.	
	ITS:	**NUREG:**
	LCO 3.03.01 T3.03.01-01 07A	LCO 3.03.01 T3.03.01-01 08A
	LCO 3.03.01 T3.03.01-01 07B	LCO 3.03.01 T3.03.01-01 08B
	LCO 3.03.01 T3.03.01-01 08	LCO 3.03.01 T3.03.01-01 09
	LCO 3.03.01 T3.03.01-01 09A	LCO 3.03.01 T3.03.01-01 10A
	LCO 3.03.01 T3.03.01-01 09B	LCO 3.03.01 T3.03.01-01 10B
	LCO 3.03.01 T3.03.01-01 13	LCO 3.03.01 T3.03.01-01 14
	LCO 3.03.01 T3.03.01-01 14-01	LCO 3.03.01 T3.03.01-01 15-01
	LCO 3.03.01 T3.03.01-01 14-02	LCO 3.03.01 T3.03.01-01 15-02
	LCO 3.03.01 T3.03.01-01 17B-02	LCO 3.03.01 T3.03.01-01 18F

JFD Number	JFD Text	
33 Rev. D	LCO 3.3.1 has been modified by the addition of SR 3.3.1.15, ACTUATION LOGIC TEST. This surveillance is performed every 18 months on the automatic actuation logic associated with the RCP Breaker Position (Two Loops), Reactor Coolant Flow-Low (Two Loops) and Underfrequency Bus A01 and A02 Trip Functions and the P-6, P-7, P-8, P-9 and P-10 Interlocks. This is consistent with the current licensing basis of CTS Table 15.4.1-1, items 5 and 45, which require testing of each of the above trip functions / interlocks each refueling interval (18 months).	
	ITS:	**NUREG:**
	B 3.03.01	B 3.03.01
	LCO 3.03.01 T3.03.01-01 21-01	LCO 3.03.01 T3.03.01-01 21-01
	SR 3.03.01.15	N/A
	SR 3.03.01.15 NOTE	N/A
34 Rev. A	The "Undervoltage RCPs" and "Underfrequency RCPs" functions have been renamed "Undervoltage Bus A01 and A02" and "Underfrequency Bus A01 and A02", to reflect the nomenclature currently used at Point Beach.	
	ITS:	**NUREG:**
	B 3.03.01	B 3.03.01
	LCO 3.03.01 T3.03.01-01 11	LCO 3.03.01 T3.03.01-01 12
	LCO 3.03.01 T3.03.01-01 12	LCO 3.03.01 T3.03.01-01 13
35 Rev. A	The NUREG 1431, Undervoltage Bus A01 and A02 trip function TADOT surveillance requirement is not being retained in ITS. Point Beach currently performs a Channel Calibration once per refueling outage. This surveillance includes the Channel Functional Test by definition. The proposed ITS will require the performance of a Channel Calibration once per 18 months. This surveillance requirement will encompass the requirements of a TADOT and separate performance of the TADOT is not required.	
	ITS:	**NUREG:**
	LCO 3.03.01 T3.03.01-01 12	LCO 3.03.01 T3.03.01-01 13
36 Rev. A	The "Low Fluid Oil Pressure Turbine Trip" has been changed to "Low Autostop Oil Pressure Turbine Trip", to reflect the nomenclature currently used at Point Beach to describe this RPS trip function.	
	ITS:	**NUREG:**
	B 3.03.01	B 3.03.01
	LCO 3.03.01 T3.03.01-01 15A	LCO 3.03.01 T3.03.01-01 16A

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JFD Number	JFD Text	
37 Rev. A	Point Beach design provides for two Turbine Stop Valves. Therefore, only two Turbine Stop Valve Closure trip function channels are require to be operable, instead of four, as listed in NUREG 1431, Table 3.3.1-1, function 16.b.	
	ITS:	**NUREG:**
	B 3.03.01	B 3.03.01
	LCO 3.03.01 T3.03.01-01 15B	LCO 3.03.01 T3.03.01-01 16B
38 Rev. A	The Channel Calibration surveillance requirement for the Low Autostop Oil Pressure and Turbine Stop Valve Closure trip functions are not being retained in ITS. Point Beach current licensing basis does not require the performance of a Channel Calibration on either of these trip functions, because there are no analytical values, required range or accuracy associated with these trip functions. Operability of these functions can be adequately verified by the performance of a TADOT prior to reactor startup.	
	ITS:	**NUREG:**
	LCO 3.03.01 T3.03.01-01 15A	LCO 3.03.01 T3.03.01-01 16A
	LCO 3.03.01 T3.03.01-01 15B	LCO 3.03.01 T3.03.01-01 16B
39 Rev. D	The P-7 function of ITS Table 3.3.1-1 has been modified to reflect the Point Beach design. Turbine Impulse Pressure is a direct input to P-7. For completeness and accuracy with Point Beach design, the NI Power Range inputs are included in P-7. Since P-7 includes process instrumentation inputs, it is not limited to a logic only function as described in NUREG-1431.	
	ITS:	**NUREG:**
	B 3.03.01	B 3.03.01
	LCO 3.03.01 T3.03.01-01 17B-01	LCO 3.03.01 T3.03.01-01 18B N/A
	LCO 3.03.01 T3.03.01-01 17B-02	LCO 3.03.01 T3.03.01-01 18F

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30-Jan-01

JFD Number	JFD Text	
40 Rev. D	The Power Range Neutron Flux, P-9, interlock nomenclature is being changed to reflect the requirement to have the condenser available. Point Beach design incorporates signals from the Power Range Neutron Flux, Condenser Pressure, and Circulating Water Pump status. When power is above 50% RTP, Condenser pressure is high, or the minimum number of required CW pumps are not operating, a turbine trip will result in a reactor trip to prevent exceeding the steam dump system capacity on the ensuing load rejection. Designating the P-9 interlock applicable in MODE 1, with one of two circulating water pump breakers closed and vacuum greater than or equal to 22 "Hg, is consistent with the approved Ginna ITS submittal.	
	ITS:	**NUREG:**
	B 3.03.01	B 3.03.01
	LCO 3.03.01 T3.03.01-01 17D	LCO 3.03.01 T3.03.01-01 18D
	LCO 3.03.01 T3.03.01-01 NOTE (l)	N/A
41 Rev. A	The NUREG 1431 Turbine Impulse Pressure, P-13, interlock nomenclature is being modified by deleting "P-13" from the description. Point Beach does not use this designation in describing this interlock.	
	ITS:	**NUREG:**
	B 3.03.01	B 3.03.01
	LCO 3.03.01 T3.03.01-01 17B-02	LCO 3.03.01 T3.03.01-01 18F

JFD Number	JFD Text
42 Rev. A	NUREG 1431, Table 3.3.1-1, Note (k) is being replaced with another Note to reflect the addition of a separate function covering the operability requirements for the Reactor Trip Bypass Breaker and associated UV Trip Mechanism. Point Beach current licensing basis does not require Reactor Trip Breaker "trains" consisting of a main breaker and/or bypass breaker. Requiring an operable RTB "train" in this manner implies unlimited operation on the bypass breaker. Point Beach only allows the RTB to be bypassed for up to 8 hours when at power, and up to 48 hours when the unit is shutdown and rod withdrawal is possible. Additionally, it was necessary to separate the Bypass Breaker from the RTB UV and Shunt Trip Mechanism requirement, because the Bypass Breakers do not have a diverse Shunt trip mechanism.

ITS:	NUREG:
B 3.03.01	B 3.03.01
LCO 3.03.01 COND Q	LCO 3.03.01 COND R
LCO 3.03.01 COND Q RA Q.1	LCO 3.03.01 COND R RA R.1
LCO 3.03.01 COND Q RA Q.1 NOTE	LCO 3.03.01 COND R RA R.1 NOTE 1
LCO 3.03.01 T3.03.01-01 18-01	LCO 3.03.01 T3.03.01-01 19-01
LCO 3.03.01 T3.03.01-01 18-02	LCO 3.03.01 T3.03.01-01 19-02
LCO 3.03.01 T3.03.01-01 20-01	N/A
LCO 3.03.01 T3.03.01-01 20-02	N/A
LCO 3.03.01 T3.03.01-01 NOTE (n)	N/A
N/A	LCO 3.03.01 T3.03.01-01 NOTE (k)

JFD Number	JFD Text
43 Rev. D	Point Beach current licensing basis does not require RTB "trains" consisting of a main breaker and/or bypass breaker, therefore a separate function covering the operability requirements for the Reactor Trip Bypass Breaker (RTBB) and associated UV Trip Mechanism has been added to the ITS. Condition V has been added to provide Required Actions for an inoperable RTBB in MODES 1 and 2, when the bypass breaker is racked in and closed and the Rod Control System is capable of rod withdrawal. With the required RTBB inoperable, 1 hour is allowed to restore the RTBB to OPERABLE status or the unit must be placed in MODE 3 within the next 6 hours. The Completion Time of 6 hours is reasonable, based on operating experience, to reach MODE 3 from full power in an orderly manner and without challenging unit systems. The 1 hour and 6 hour completion times are equal to the time allowed by LCO 3.0.3 for shutdown action in the event of a complete loss of RPS Function. Placing the unit in MODE 3 removes the requirement for this particular Function. Condition W applies to the RTBB and associated Undervoltage Trip Mechanism in MODES 3, 4, or 5, when an RTBB is racked in and closed and the Rod Control System is capable of rod withdrawal. With the required RTBB inoperable, 48 hours are allowed to restore the RTBB to OPERABLE status or the unit must be placed in a MODE in which the requirement does not apply. To achieve this status, the RTBs and RTBBs must be opened within the next 1 hour. The Completion Time of 1 hour provides sufficient time to accomplish the action in an orderly manner. With the RTBs and RTBBs open, this Function is no longer required.

ITS:	NUREG:
B 3.03.01	B 3.03.01
LCO 3.03.01 COND V	LCO 3.03.01 COND R
LCO 3.03.01 COND V RA V.1	LCO 3.03.01 COND R RA R.1
LCO 3.03.01 COND V RA V.2	LCO 3.03.01 COND R RA R.2
LCO 3.03.01 COND W	N/A
LCO 3.03.01 COND W RA W.1	N/A
LCO 3.03.01 COND W RA W.2	N/A

JFD Number	JFD Text
44 Rev. D	Condition X has been added to provide Required Actions for inoperable Automatic Trip Logic train(s) in MODES 3, 4 and 5 with the RTBs closed and the Rod Control System capable of rod withdrawal. NUREG 1431 Condition C previously provided actions for an inoperable Automatic Trip Logic train, RTB train, or Manual Reactor Trip channel, but was revised to exclusively address the loss of a Manual Reactor Trip channel. With one train inoperable, 48 hours are allowed per Required Action X.1 to restore the train to an OPERABLE status. The Completion Time of 48 hours is reasonable considering that in this condition, the remaining OPERABLE train is adequate to perform the safety function, and given the low probability of an event occurring in this interval. If the Automatic Trip Logic cannot be restored to OPERABLE status within 48 hours, Required Action X.2 directs the RTBs to be opened within the next hour, to place the unit in a MODE where this Function is not required to be OPERABLE. The additional hour provides sufficient time to accomplish the action in an orderly manner.

ITS:	NUREG:
B 3.03.01	B 3.03.01
LCO 3.03.01 COND X	N/A
LCO 3.03.01 COND X RA X.1	N/A
LCO 3.03.01 COND X RA X.2	N/A
LCO 3.03.01 T3.03.01-01 21-02	LCO 3.03.01 T3.03.01-01 21-02

JFD Number	JFD Text
45 Rev. A	The Overtemperature delta T and Overpower delta T Allowable Value Notes have been modified by incorporating Point Beach specific information related to the calculation of the Allowable Values.

ITS:	NUREG:
LCO 3.03.01 T3.03.01-01 05 NOTE 1	LCO 3.03.01 T3.03.01-01 06 NOTE 1
LCO 3.03.01 T3.03.01-01 06 NOTE 2	LCO 3.03.01 T3.03.01-01 07 NOTE 2

JFD Number	JFD Text
46 Rev. A	LCO 3.3.1 Bases Background discussion is modified to reflect Point Beach Reactor Protection System design. The Point Beach RPS does not use a Solid State Protection System, but rather uses a logic system of relays and conductors.

ITS:	NUREG:
B 3.03.01	B 3.03.01

JFD Number	JFD Text	
47 Rev. A	LCO 3.3.1 Bases Applicable Safety Analyses, LCO and Applicability discussion of the Power Range Neutron Flux instrumentation is modified to reflect Point Beach design. The NIS Power Range detectors do not provide an input to the SG Water Level Control System.	
	ITS:	**NUREG:**
	B 3.03.01	B 3.03.01
48 Rev. A	LCO 3.3.1 Bases Applicable Safety Analyses, LCO and Applicability discussion of the Source Range Neutron Flux trip function is modified to reflect Point Beach design. The Source Range function does not provide an input to the Boron Dilution Protection System (BDPS). Point Beach does not have a BDPS, and the Source Range Neutron Flux function does not provide protection for a boron dilution event.	
	ITS:	**NUREG:**
	B 3.03.01	B 3.03.01
49 Rev. A	NUREG 1431, LCO 3.9.2, "Unborated Water Source Isolation Valves", is not applicable. A boron dilution event has been analyzed for Point Beach, as described in FSAR Section 14.1.4. Therefore NUREG 1431, LCO 3.9.2 has been deleted, resulting in the re-numbering of subsequent LCOs.	
	ITS:	**NUREG:**
	B 3.03.01	B 3.03.01
50 Rev. A	LCO 3.3.1 Bases Applicable Safety Analyses, LCO and Applicability discussion of the Underfrequency Bus A01 and A02 trip function is modified to reflect Point Beach design. An underfrequency condition on both RCP buses will not directly trip the reactor, but will trip both RCP breakers, when operating above 10% RTP. When both RCP breakers are tripped open, a reactor trip signal will be generated.	
	ITS:	**NUREG:**
	B 3.03.01	B 3.03.01
51 Rev. A	LCO 3.3.1 Bases Applicable Safety Analyses, LCO and Applicability discussion of the Intermediate Range Neutron Flux, P-6, Interlock has been modified to reflect Point Beach design. The P-6 interlock at Point Beach does not provide a backup block signal to the source range flux doubling circuit. Therefore this discussion has been deleted.	
	ITS:	**NUREG:**
	B 3.03.01	B 3.03.01

JFD Number	JFD Text	
52 Rev. A	LCO 3.3.1 Bases discussions of Overtemperature delta T and Overpower delta T trip functions have been modified. At Point Beach these trip functions are calculated for each channel, not each loop. Furthermore, the Overtemperature delta T and Overpower delta T trip functions each actuate via a 2-out-of-4 logic configuration, with 2 channels from each loop, and do not require signals from both loops to initiate a reactor trip.	
	ITS:	**NUREG:**
	B 3.03.01	B 3.03.01
53 Rev. A	SR 3.3.1.4 Bases have been modified to reflect the Point Beach Reactor Trip Bypass Breaker (RTBB) design. The bypass breakers do not have a diverse shunt trip mechanism; however, the undervoltage trip shall be included as part of the TADOT on the RTBB. The verification of the RTBB undervoltage trip shall be performed as a part of SR 3.3.1.4, instead of SR 3.3.1.14, as indicated in the Bases of NUREG-1431. Additionally, text added to the Bases description of SR 3.3.1.4, via TSTF-205, has not been incorporated into the ITS. Point Beach design of RPS necessitates TADOT testing which is inconsistent with the verbiage added by TSTF-205.	
	ITS:	**NUREG:**
	B 3.03.01	B 3.03.01
54 Rev. D	Not used.	
	ITS:	**NUREG:**
	N/A	N/A

Justification For Deviations - NUREG-1431 Section 3.03.01

30-Jan-01

JFD Number	JFD Text
55 Rev. A	The Channel Calibration surveillance requirement for Overtemperature delta T and Overpower delta T trip functions have been changed. A Note modifying NUREG-1431, SR 3.3.1.12 requires the verification of the RCS RTD bypass loop flow rate as a part of the Channel Calibration. Point Beach Overtemperature delta T and Overpower delta T trip functions do not require verification of the RCS RTD bypass loop flow rate. Therefore NUREG-1431, SR 3.3.1.12 has been deleted and the Channel Calibration requirements for Overtemperature delta T and Overpower delta T trip functions refer to ITS SR 3.3.1.11. Implementation of this change also results in the re-numbering of subsequent surveillance requirements. Additionally, text added to the Bases description of SR 3.3.1.12, via TSTF-19, has not been incorporated into the ITS.

ITS:	NUREG:
B 3.03.01	B 3.03.01
LCO 3.03.01 T3.03.01-01 01-01	LCO 3.03.01 T3.03.01-01 01-01
LCO 3.03.01 T3.03.01-01 01-02	LCO 3.03.01 T3.03.01-01 01-02
LCO 3.03.01 T3.03.01-01 05	LCO 3.03.01 T3.03.01-01 06
LCO 3.03.01 T3.03.01-01 06	LCO 3.03.01 T3.03.01-01 07
LCO 3.03.01 T3.03.01-01 10A	LCO 3.03.01 T3.03.01-01 11A
LCO 3.03.01 T3.03.01-01 10B	LCO 3.03.01 T3.03.01-01 11B
LCO 3.03.01 T3.03.01-01 15A	LCO 3.03.01 T3.03.01-01 16A
LCO 3.03.01 T3.03.01-01 15B	LCO 3.03.01 T3.03.01-01 16B
LCO 3.03.01 T3.03.01-01 16	LCO 3.03.01 T3.03.01-01 17
LCO 3.03.01 T3.03.01-01 17A	LCO 3.03.01 T3.03.01-01 18A
LCO 3.03.01 T3.03.01-01 17B-02	LCO 3.03.01 T3.03.01-01 18F
LCO 3.03.01 T3.03.01-01 17C	LCO 3.03.01 T3.03.01-01 18C
LCO 3.03.01 T3.03.01-01 17E	LCO 3.03.01 T3.03.01-01 18E
N/A	SR 3.03.01.12 SR 3.03.01.12 NOTE
SR 3.03.01.12	SR 3.03.01.13
SR 3.03.01.13	SR 3.03.01.14

JFD Number	JFD Text	
56 Rev. D	Not used.	
	ITS:	**NUREG:**
	N/A	N/A
57 Rev. A	LCO 3.3.1 Bases discussion of the logic configuration when one RPS channel is also used as a control system input has been modified to reflect that four channels in a two-out-of-four logic configuration are "generally" required to account for the possibility of the shared channel failing in a manner that creates a transient requiring RPS action. At Point Beach, the Pressurizer Pressure-High trip function does not require a fourth channel, but rather relies on a two-out-of-three logic. Although a pressurizer pressure channel failure that produces a low pressure signal could turn on the pressurizer backup heaters, the resulting pressure increase is easily handled by the PORVs without the requirement for a reactor trip.	
	ITS:	**NUREG:**
	B 3.03.01	B 3.03.01
58 Rev. A	LCO 3.3.1 Bases discussion of the Pressurizer Pressure-High reactor trip function has been modified to reflect Point Beach operation at 2250 psia and at 2000 psia. The NUREG-1431 statement that the LSSS is above the PORV setting holds true for operation at 2250 psia. For operation at 2000 psia, the Pressurizer Pressure-High LSSS is below the PORV setting. However, the PORVs are not relied on to avoid an unnecessary reactor trip. A 50% load rejection with steam dump from operation at 2000 psia results in a peak pressure less than the Pressurizer Pressure-High LSSS and less than the PORV actuation setpoint.	
	ITS:	**NUREG:**
	B 3.03.01	B 3.03.01
59 Rev. A	LCO 3.3.1 Bases discussion of the Undervoltage Bus A01 and A02 reactor trip function has been modified by deleting the sentence stating this function uses the same relays as the ESFAS Undervoltage RCP start of AFW. At Point Beach these functions do not share the same relays.	
	ITS:	**NUREG:**
	B 3.03.01	B 3.03.01
60 Rev. A	"Reactor Trip System (RTS)" has been changed to "Reactor Protection System (RPS)", to reflect the nomenclature currently used at Point Beach.	
	ITS:	**NUREG:**
	B 3.03.01	B 3.03.01
	LCO 3.03.01	LCO 3.03.01

JFD Number	JFD Text	
61 Rev. A	LCO 3.3.1 Bases descriptions regarding the P-7 interlock have been revised. Point Beach design of the P-7 interlock uses inputs from the P-10 interlock and the Turbine Impulse Pressure Interlock. As such, there is no "setpoint" for the P-7 interlock. Therefore all instances of "P-7 setpoint" have been changed to "P-7 interlock".	
	ITS:	**NUREG:**
	B 3.03.01	B 3.03.01
62 Rev. A	LCO 3.3.1 Bases discussions of the RCP Breaker Position trip functions have been modified. A description of a RCP Breaker Position channel has been added to aid in the verification of OPERABILITY of each function.	
	ITS:	**NUREG:**
	B 3.03.01	B 3.03.01
63 Rev. A	LCO 3.3.1 Bases discussion of the SG Water Level-Low, Coincident with Steam Flow/Feedwater Flow Mismatch has been modified to reflect Point Beach design. This trip function does not provide an input to AFW actuation. Therefore this statement has been deleted.	
	ITS:	**NUREG:**
	B 3.03.01	B 3.03.01
64 Rev. A	LCO 3.3.1 Bases discussion of the Power Range Neutron Flux, P-8, interlock has been modified. The statement that P-8 automatically enables the Reactor Coolant Flow-Low (Single Loop) and RCP Breaker Position (Single Loop) reactor trips "on low flow in one or more RCS loops on increasing power." is confusing. The RCP Breaker Position trip functions do not receive input from RCS loop flow. Therefore the phrase "on low flow in one or more RCS loops" has been deleted.	
	ITS:	**NUREG:**
	B 3.03.01	B 3.03.01
65 Rev. A	LCO 3.3.1 Bases discussion of Power Range Neutron Flux, P-10, interlock has been modified. Point Beach design of the P-10 interlock does not interface with the rod stop circuitry. Therefore statements to this regard have been deleted.	
	ITS:	**NUREG:**
	B 3.03.01	B 3.03.01
66 Rev. A	LCO 3.3.1 Bases discussion of SR 3.3.1.4 has been modified. The phrase "SR 3.3.1.4 to" has been added to clarify that the Note indicating that the test must be performed on the bypass breaker prior to placing it in service, is modifying SR 3.3.1.4.	
	ITS:	**NUREG:**
	B 3.03.01	B 3.03.01

JFD Number	JFD Text	
67 Rev. A	LCO 3.3.1 Bases discussion of SR 3.3.1.7 has been modified to reflect Point Beach setpoint methodology. The "as found" and "as left" values obtained during the performance of a COT are verified to be within limits. These values are not reviewed for consistency with the assumptions of WCAP-10271-P-A, Supplement 2. Additionally, text added to the Bases description of SR 3.3.1.7, via TSTF-205, has not been incorporated into the ITS. Point Beach design of RPS necessitates COT testing which is inconsistent with the verbiage added by TSTF-205.	
	ITS:	**NUREG:**
	B 3.03.01	B 3.03.01
68 Rev. A	ITS LCO 3.3.1 Bases discussion of undervoltage and shunt trip "mechanisms" has been changed to undervoltage and shunt trip "circuits" to reflect Point Beach design for these devices.	
	ITS:	**NUREG:**
	B 3.03.01	B 3.03.01
69 Rev. D	ITS LCO 3.3.1 Bases discussion of SR 3.3.1.3 has been modified to indicate the SR is performed by means of a movable incore detector system. This addition to the Bases retains information previously contained in the CTS.	
	ITS:	**NUREG:**
	B 3.03.01	B 3.03.01
70 Rev. D	The Allowable Values associated with the SG Water Level – Low and Turbine Trip – Low Autostop Oil Pressure reactor trips have been replaced with field settings. These field settings were developed outside of the setpoint methodology and have been provided by the NSSS supplier. No analytical limit or Allowable Value has been established for any of these functions as they are not credited in the safety analysis for the mitigation of any accident. SG Water Level – Low is an anticipatory trip for the SG Water Level – Low Low trip for the mitigation of a Loss of Normal Feedwater event. Reactor trip on Turbine trip is an anticipatory trip to other reactor trips that would be challenged by a load rejection event (OTdeltaT, Pressurizer Pressure – High, and SG Water Level – Low Low). Therefore, the field setting for each of these functions being provided in Table 3.3.1-1does not imply that an analytical limit exists for them, or that they are necessary to prevent exceeding a safety limit.	
	ITS:	**NUREG:**
	B 3.03.01	B 3.03.01
	LCO 3.03.01 T3.03.01-01 14-01	LCO 3.03.01 T3.03.01-01 15-01
	LCO 3.03.01 T3.03.01-01 15A	LCO 3.03.01 T3.03.01-01 16A
	LCO 3.03.01 T3.03.01-01 NOTE (j)	N/A
	LCO 3.03.01 T3.03.01-01 NOTE (m)	N/A

JFD Number	JFD Text	
71 Rev. D	The Notes modifying NUREG-1431, Conditions D, E, M, N and P have not been retained in ITS. The provision to allow taking the inoperable channel out of the tripped condition for 4 hours for surveillance testing of other channels is based upon the analysis contained in WCAP-10271-P-A, Supplement 2. The SERs for WCAP-10271 required individual plants to confirm the applicability of the generic analysis of the WCAP. Point Beach Nuclear Plant has not confirmed the applicability of the generic analysis of WCAP-10271, and therefore will not adopt these notes.	
	ITS:	**NUREG:**
	N/A	LCO 3.03.01 COND D RA D.1.1 NOTE
		LCO 3.03.01 COND E RA E.1 NOTE
		LCO 3.03.01 COND M RA M.1 NOTE
		LCO 3.03.01 COND N RA N.1 NOTE
		LCO 3.03.01 COND P RA P.1 NOTE
72 Rev. D	The time allowed to place an inoperable channel in the tripped condition has been changed from 6 hours to 1 hour. The 6 hour completion time of NUREG-1431, Required Actions D.1.1, E.1, M.1, N.1 and P.1 are based upon the analysis contained in WCAP-10271-P-A, Supplement 2. The SERs for WCAP-10271 require individual plants to confirm the applicability of the generic analysis of the WCAP. Point Beach Nuclear Plant has not confirmed the applicability of the generic analysis of WCAP-10271 and therefore, will retain the Completion Time requirements of the current licensing basis. This change also results in the revision of the Completion Times associated with successive Required Actions, such that the assumptions for completion of these required actions remain valid.	
	ITS:	**NUREG:**
	LCO 3.03.01 COND D RA D.1	LCO 3.03.01 COND D RA D.1.1
	LCO 3.03.01 COND D RA D.2	LCO 3.03.01 COND D RA D.3
	LCO 3.03.01 COND K RA K.1	LCO 3.03.01 COND M RA M.1
	LCO 3.03.01 COND K RA K.2	LCO 3.03.01 COND M RA M.2
	LCO 3.03.01 COND L RA L.1	LCO 3.03.01 COND N RA N.1
	LCO 3.03.01 COND L RA L.2	LCO 3.03.01 COND N RA N.2
	LCO 3.03.01 COND O RA O.1	LCO 3.03.01 COND P RA P.1
	LCO 3.03.01 COND O RA O.2	LCO 3.03.01 COND P RA P.2

JFD Number	JFD Text	
73 Rev. D	Point Beach CLB doesn't require action for an inoperable Underfrequency Bus A01 and A02 trip channel that results in one less than the total number of channels being operable. With the adoption of the NUREG-1431 requirement for two channels per bus, it is necessary to provide required actions for one inoperable channel. Therefore, the NUREG-1431 actions for one inoperable Underfrequency Bus A01 and A02 trip channel have been adopted with the following exception. NUREG-1431 Required Action M.1 requires the inoperable channel to be placed in trip within 6 hours, as will ITS Required Action E.1. However, NUREG-1431 Required Action M.1 has been revised to only allow one hour to place inoperable channels in trip because Point Beach Nuclear Plant has not performed a plant specific evaluation to confirm the applicable ability of the generic analysis contained in WCAP-10271-P-A, Supplement 2 (upon which the 6 hour allowance of RA M.1 is based). The 6 hour allowance to place an inoperable channel of the Underfrequency Bus A01 and A02 trip function in trip is based on the time required to call in an Electrical Maintenance technician (who may not be on site during back shifts, weekends and holidays) to be dispatched to the location of the relay to place the channel in trip. This difference necessitates the addition of a new Condition and Required Actions to address the unique aspects of the Underfrequency Bus A01 and A02 trip function. NUREG-1431, Condition E and associated Required Actions have been utilized and revised in ITS as Condition E and associated Required Actions, for one inoperable Underfrequency Bus A01 and A02 trip channel.	
	ITS:	**NUREG:**
	LCO 3.03.01 COND E	LCO 3.03.01 COND E
	LCO 3.03.01 COND E RA E.1	LCO 3.03.01 COND E RA E.1
	LCO 3.03.01 COND E RA E.2	LCO 3.03.01 COND E RA E.2
74 Rev. D	The SERs for WCAP-10271-P-A, Supplement 2, require individual plants to confirm the applicability of the generic analysis of the WCAP. Point Beach Nuclear Plant has not confirmed the applicability of the generic analysis of WCAP-10271. However, adoption of the requirements for an operable Source Range Neutron Flux trip function in MODES 3, 4 and 5 with the RTBs closed and the Rod Control System capable of rod withdrawal is a new, more restrictive requirement, as are the required actions for an inoperable Source Range Neutron Flux channel in these MODES. Therefore, the Completion Times of NUREG-1431 have been adopted.	
	ITS:	**NUREG:**
	B 3.03.01	B 3.03.01
75 Rev. D	The SERs for WCAP-10271-P-A, Supplement 2, require individual plants to confirm the applicability of the generic analysis of the WCAP. Point Beach Nuclear Plant has not confirmed the applicability of the generic analysis of WCAP-10271, and therefore, will retain the frequency requirements of the current licensing basis.	
	ITS:	**NUREG:**
	B 3.03.01	B 3.03.01

60

Protection (RPS)

3.3 INSTRUMENTATION

3.3.1 Reactor Trip System (RTS) Instrumentation

RPS 60

LCO 3.3.1 The RTS instrumentation for each Function in Table 3.3.1-1 shall be OPERABLE.

APPLICABILITY: According to Table 3.3.1-1.

ACTIONS

------------------------------------NOTE------------------------------------
Separate Condition entry is allowed for each Function.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more Functions with one or more required channels or trains inoperable. Approved TSTF-135	A.1 Enter the Condition referenced in Table 3.3.1-1 for the channel(s) or train(s).	Immediately Approved TSTF-135
B. One Manual Reactor Trip channel inoperable.	B.1 Restore channel to OPERABLE status. OR Approved TSTF-135 B.2~~.1~~ Be in MODE 3. ~~AND~~ ~~B.2.2 Open reactor trip breakers (RTBs).~~	48 hours 54 hours ~~55 hours~~ Approved TSTF-135

D
RAI 3.3.1-1

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
C. One channel ~~or train~~ inoperable. Manual Reactor Trip 4	C.1 Restore channel ~~or train~~ to OPERABLE status. 4	48 hours
	OR	
	C.2 Open RTBs.	49 hours
7 D. One ~~Power Range Neutron Flux – High~~ channel inoperable.	~~NOTE~~ ~~The inoperable channel may be bypassed for up to 4 hours for surveillance testing and setpoint adjustment of other channels.~~ 71	
	D.1~~.1~~ Place channel in trip.	6 hours 1 hour 72
	~~AND~~	
7	~~D.1.2 Reduce THERMAL POWER to ≤ 75% RTP.~~	~~12 hours~~
	~~OR~~	
	~~D.2.1 Place channel in trip.~~	~~6 hours~~
	~~AND~~	
		(continued)

RAI 3.3.1-1 D

RAI 3.3.1-30 D

RAI 3.3.1-2 D

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
D. (continued)	~~-------------NOTE------------ Only required to be performed when the Power Range Neutron Flux input to QPTR is inoperable. ---------------------------~~	
	~~D.2.2 Perform SR 3.2.4.2.~~	~~Once per 12 hours~~ 7
	OR	
7 D.2	D.3 Be in MODE 3.	12 hours 7 72
E. One channel inoperable.	~~-------------NOTE------------ The inoperable channel may be bypassed for up to 4 hours for surveillance testing of other channels. ---------------------------~~ 71	
	E.1 Place channel in trip.	6 hours
	OR	
73 Reduce THERMAL POWER to < P-7.	E.2 Be in MODE 3.	12 hours
Approved TSTF-135 F. ~~THERMAL POWER > P-6 and < P-10,~~ one Intermediate Range Neutron Flux channel inoperable.	F.1 Reduce THERMAL POWER to < P-6.	2 hours 24
	OR Approved TSTF-246	
	F.2 Increase THERMAL POWER to > P-10.	2 hours 24

D RAI 3.3.1-2

D RAI 3.3.1-30

D RAI 3.3.1-2 Errata #145

D Errata #145

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
G. ~~THERMAL POWER > P-6 and < P-10,~~ two Intermediate Range Neutron Flux channels inoperable. Approved TSTF-135	G.1 Suspend operations involving positive reactivity additions.	Immediately
	AND	
	G.2 Reduce THERMAL POWER to < P-6.	2 hours
~~H. THERMAL POWER < P-6, one or two Intermediate Range Neutron Flux channels inoperable.~~	~~H.1 Restore channel(s) to OPERABLE status.~~ Approved TSTF-135	~~Prior to increasing THERMAL POWER to > P-6~~
I. One Source Range Neutron Flux channel inoperable.	I.1 Suspend operations involving positive reactivity additions.	Immediately
J. Two Source Range Neutron Flux channels inoperable. Approved TSTF-135	J.1 Open RTBs.	Immediately
K. One Source Range Neutron Flux channel inoperable. Approved TSTF-135	K.1 Restore channel to OPERABLE status.	48 hours
	OR	
	K.2 Open RTBs.	49 hours

Errata #145

RAI 3.3.1-1
Errata #145

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
~~L. Required Source Range Neutron Flux channel[(s)] inoperable.~~ Approved TSTF-135	~~L.1 Suspend operations involving positive reactivity additions.~~ ~~AND~~ ~~L.2 Close unborated water source isolation valves.~~ ~~AND~~ ~~L.3 Perform SR 3.1.1.1.~~	~~Immediately~~ ~~1 hour~~ ~~1 hour~~ ~~AND~~ ~~Once per 12 hours thereafter~~
[M] K One channel inoperable. Approved TSTF-135	~~-----NOTE----- The inoperable channel may be bypassed for up to 4 hours for surveillance testing of other channels.~~ 71 K [M].1 Place channel in trip. OR K [M].2 Reduce THERMAL POWER to < P-7.	[6 hours] 1 hour 72 [12] hours 7

D RAI 3.3.1-1 RAI 3.3.1-30 RAI 3.3.1-2 Errata #145

D RAI 3.3.1-4 RAI 3.3.1-8

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
N L One Reactor Coolant Flow - Low (Single Loop) channel inoperable. Approved TSTF-135	~~-----NOTE----- The inoperable channel may be bypassed for up to 4 hours for surveillance testing of other channels.~~ 71	
	L N.1 Place channel in trip.	6 hours 1 hour 72
	OR	
	L N.2 Reduce THERMAL POWER to < P-8.	5 10 hours
O M One Reactor Coolant Pump Breaker Position channel inoperable. Approved TSTF-135	~~-----NOTE----- The inoperable channel may be bypassed for up to 4 hours for surveillance testing of other channels.~~ 13	
	M O.1 Restore channel to OPERABLE status.	6 hours 1 hour 13
	OR	
	O.2 Reduce THERMAL POWER to < P-8.	5 10 hours

D RAI 3.3.1-1 RAI 3.3.1-30 RAI 3.3.1-2 Errata #145

D RAI 3.3.1-5

D RAI 3.3.1-1 Errata #145

D RAI 3.3.1-10

(continued)

Insert N 14

D RAI 3.3.1-1 Errata #145

Insert N

N. One channel inoperable.	N.1	Restore channel to OPERABLE status.	1 hour	D RAI 3.3.1-1 RAI 3.3.1-8 RAI 3.3.1-10 Errata #145
	OR			
	N.2	Reduce THERMAL POWER to < P-7.	7 hours	

60 RPS

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
P One Turbine Trip channel inoperable.	~~NOTE The inoperable channel may be bypassed for up to 4 hours for surveillance testing of other channels.~~	
	P.1 Place channel in trip.	6 hours 1 hour
	OR	
	P.2 Reduce THERMAL POWER to < [P-9]	5 10 hours
Q. One train inoperable.	NOTE One train may be bypassed for up to [4] hours for surveillance testing provided the other train is OPERABLE.	
	Q.1 Restore train to OPERABLE status.	6 hours
	OR	
	Q.2 Be in MODE 3.	12 hours

Approved TSTF-135 O P 71 72 16 P-9 8 16

D RAI 3.3.1-1 RAI 3.3.1-30 RAI 3.3.1-2 Errata #145

D RAI 3.3.1-7

D RAI 3.3.1-1 Errata #145

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME	
R. One RTB ~~train~~ inoperable. (42; Q; Approved TSTF-135)	RTB 42 ------------NOTES------------ ~~1.~~ One train may be bypassed for up to 2 hours ~~for surveillance testing,~~ provided the other train is OPERABLE. (17; 8 17; RTB 42) ~~2. One RTB may be bypassed for up to 2 hours for maintenance on undervoltage or shunt trip mechanisms, provided the other train is OPERABLE.~~ -----------------------------------		D RAI 3.3.1-1 Errata #145
Approved TSTF-135 Q	42 RTB R.1 Restore train to OPERABLE status.	1 hour	
	OR R.2 Be in MODE 3.	7 hours	D RAI 3.3.1-1 Errata #145
or more channel(s) S. One channel inoperable. (R; Approved TSTF-135)	R S.1 Verify interlock is in required state for existing unit conditions.	1 hour	D RAI 3.3.1-1 Errata #145
	OR R S.2 Be in MODE 3.	7 hours	

(continued)

Insert 0

NOT USED

D

RAI 3.3.1-1
Errata #145

ACTIONS (continued)

CONDITION	REQUIRED ACTION		COMPLETION TIME
T. One channel or more channel(s) inoperable. Approved TSTF-135	T.1	Verify interlock is in required state for existing unit conditions.	1 hour
	OR		
	T.2	Be in MODE 2.	7 hours
U. One trip mechanism inoperable for one RTB.	U.1	Restore inoperable trip mechanism to OPERABLE status.	48 hours
	OR		
	U.2.~~1~~	Be in MODE 3.	54 hours
	~~AND~~		Approved TSTF-135
	~~U.2.2~~	~~Open RTB.~~	~~55 hours~~
~~V. Two RTS trains inoperable.~~	~~V.1~~	~~Enter LCO 3.0.3.~~	~~Immediately~~

Insert T ← 20

Insert V and W ← 43

Insert X ← 44

D RAI 3.3.1-1 Errata #145

D RAI 3.3.1-1 Errata #145

D RAI 3.3.1-1 Errata #145

Insert T

T.	One RTB or trip mechanism for one RTB inoperable.	T.1	Restore RTB or RTB trip mechanism to OPERABLE status.	48 hours	△D RAI 3.3.1-1 Errata #145
		OR			
		T.2	Open RTBs.	49 hours	

Inserts V and W

V.	One reactor trip bypass breaker (RTBB) or trip mechanism for one RTBB inoperable.	V.1	Restore RTBB or RTBB trip mechanism to OPERABLE status.	1 hour
		OR		
		V.2	Be in MODE 3.	7 hours
W.	One RTBB or trip mechanism for one RTBB inoperable.	W.1	Restore RTBB or RTBB trip mechanism to OPERABLE status.	48 hours
		OR		
		W.2	Open RTBs and RTBBs.	49 hours

Insert X

X. One train inoperable.	X.1 Restore train to OPERABLE status. OR X.2 Open RTBs.	48 hours 49 hours

D

RAI 3.3.1-1
Errata #145

SR 3.3.1.5 Notes Insert

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

1. Not required to be performed for the Source Range Neutron Flux trip function until 8 hours after power is below P-6.

2. Not required to be performed for the RCP Breaker Position (Two Loop), Reactor Coolant Flow - Low (Two Loop) and Underfrequency Bus A01 and A02 Trip Functions and the P-6, P-7, P-8, P-9 and P-10 Interlocks.

D

RAI 3.3.1-22
RAI TR-2
Errata #46

60 → RPS → RTS

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.3.1.8	------------------NOTE------------------ This Surveillance shall include verification that interlocks P-6 and P-10 are in their required state for existing unit conditions. --------------------------------------- Perform COT.	-----NOTE----- Only required when not performed within previous [92] days -------------- Prior to reactor startup AND Four hours after reducing power below P-10 for power and intermediate instrumentation AND Four hours after reducing power below P-6 for source range instrumentation AND Every 92 days thereafter

16 → 92

D RAI 3.3.1-17

23 → range

D RAI 3.3.1-17

(continued)

60 ~~RPS~~

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.3.1.14 (55 → 13)	~~-----NOTE----- Verification of setpoint is not required.~~ (24) Perform TADOT.	18 (16) [18] months
SR 3.3.1.15 (55 → 14)	~~-----NOTE----- Verification of setpoint is not required.~~ (24) Perform TADOT.	-----NOTE----- Only required when not performed within previous 31 days ----- Prior to reactor startup (16 → P-9 → Approved TSTF-311: Prior to exceeding the [P-9] interlock whenever the unit has been in MODE 3, if not performed within previous 31 days.)
~~SR 3.3.1.16~~	-----~~NOTE~~----- ~~Neutron detectors are excluded from response time testing.~~ ----- ~~Verify RTS RESPONSE TIME is within limits.~~	(25) ~~[18] months on a STAGGERED TEST BASIS~~

D RAI 3.3.1-7

D RAI 3.3.1-26

33

D RAI 3.3.1-22

SR 3.3.1.15	-----NOTE----- This Surveillance shall be performed on the RCP Breaker Position (Two Loop), Reactor Coolant Flow - Low (Two Loop) and Underfrequency Bus A01 and A02 Trip Functions and the P-6, P-7, P-8, P-9 and P-10 Interlocks. ----- Perform ACTUATION LOGIC TEST.	18 months

Table 3.3.1-1 (page 1 of 8)
Reactor Trip Protection System Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS	CONDITIONS	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE	TRIP SETPOINT(a)
1. Manual Reactor Trip	1,2	2	B	SR 3.3.1.13	NA	NA
	3(b), 4(b), 5(b)	2	C	SR 3.3.1.13	NA	NA
2. Power Range Neutron Flux						
a. High	1,2	4	D	SR 3.3.1.1 SR 3.3.1.2 SR 3.3.1.7 SR 3.3.1.11 ~~SR 3.3.1.16~~	≤ [108]% RTP	≤ [109]% RTP
b. Low	1(c),2	4	E	SR 3.3.1.1 SR 3.3.1.8 SR 3.3.1.11 ~~SR 3.3.1.16~~	≤ [25]% RTP	≤ [25]% RTP
~~3. Power Range Neutron Flux Rate~~						
~~a. High Positive Rate~~	~~1,2~~	~~4~~	~~E~~	~~SR 3.3.1.7~~ ~~SR 3.3.1.11~~	~~≤ [6.8]% RTP with time constant ≥ [2] sec~~	≤ [5]% RTP with time constant ≥ [2] sec
~~b. High Negative Rate~~	~~1,2~~	~~4~~	~~E~~	~~SR 3.3.1.7~~ ~~SR 3.3.1.11~~ ~~SR 3.3.1.16~~	~~≤ [6.8]% RTP with time constant ≥ [2] sec~~	≤ [5]% RTP with time constant ≥ [2] sec
3. Intermediate Range Neutron Flux	1(c), 2(d)	2	F,G	SR 3.3.1.1 SR 3.3.1.8 SR 3.3.1.11	≤ [40]% RTP	≤ [25]% RTP
	~~2(e)~~	~~2~~	~~H~~	~~SR 3.3.1.1~~ ~~SR 3.3.1.8~~ ~~SR 3.3.1.11~~	~~≤ [31]% RTP~~	≤ [25]% RTP

Approved TSTF-135

(continued)

~~(a) Reviewer's Note: Unit specific implementations may contain only Allowable Value depending on Setpoint Study methodology used by the unit.~~

(a) With Reactor Trip Breakers (RTBs) closed and Rod Control System capable of rod withdrawal.

(b) Below the P-10 (Power Range Neutron Flux) interlocks.

(c) Above the P-6 (Intermediate Range Neutron Flux) interlocks.

~~(e) Below the P-6 (Intermediate Range Neutron Flux) interlocks.~~

Approved TSTF-135

RAI 3.3.1-1
Errata #145

Insert 2.a NOT USED

Insert 2.b NOT USED

D

Errata #145

60 → RPS

Table 3.3.1-1 (page 2 of 8)
Reactor Trip System Instrumentation

60 → Protection

15 | 27

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS	CONDITIONS	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE	TRIP SETPOINT(a)
29 → 4 → 5 Source Range Neutron Flux	2(e) ← (d) ← 26	2	H,I → I,J	SR 3.3.1.1 SR 3.3.1.8 SR 3.3.1.11 ~~SR 3.3.1.16~~	≤ [1.4 E5] cps within span of instrumentation	≤ [1.0 E5] cps
	26 → (a) → 3(b), 4(b), 5(b)	2	I,J → J,K	SR 3.3.1.1 SR 3.3.1.7 SR 3.3.1.11 ~~SR 3.3.1.16~~	≤ [1.4 E5] cps	≤ [1.0 E5] cps
Approved TSTF-135 →	~~3(f), 4(f), 5(f)~~	~~[1]~~	~~L~~	~~SR 3.3.1.1~~ ~~SR 3.3.1.11~~	~~N/A~~	N/A
29 → 5 → 6 Overtemperature ΔT	1,2	16 → 4 → [4]	7 → D → E	SR 3.3.1.1 SR 3.3.1.3 SR 3.3.1.6 SR 3.3.1.7 SR 3.3.1.12 ~~SR 3.3.1.16~~	Refer to Note 1 (Page 3.3-21)	Refer to Note 1 (Page 3.3-21)
29 → 6 → 7 Overpower ΔT	1,2	4 → [4]	D → E	SR 3.3.1.1 SR 3.3.1.7 SR 3.3.1.12 ~~SR 3.3.1.16~~	Refer to Note 2 (Page 3.3-22)	Refer to Note 2 (Page 3.3-22)

Approved TSTF-135 | 25 | 16 | 25 | 55 → 11

D RAI 3.3.1-1 Errata #145

(continued)

26

~~(a) Reviewer's Note: Unit specific implementations may contain only Allowable Value depending on Setpoint Study methodology used by the unit.~~

(a) → (b) With RTBs closed and Rod Control System capable of rod withdrawal.

(d) → (e) Below the P-6 (Intermediate Range Neutron Flux) interlocks.

26 ~~(f) With the RTBs open. In this condition, source range Function does not provide reactor trip but does provide [input to the Boron Dilution Protection System (LCO 3.3.9), and] indication.~~ Approved TSTF-135

D RAI 3.3.1-12 Errata #145

60 → RPS → RTS

Table 3.3.1-1 (page 3 of 8)
Reactor Trip Protection System Instrumentation

60 → Protection → Trip

15 → APPLICABLE MODES → APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS

27 → TRIP SETPOINT(a)

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS	CONDITIONS	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE	TRIP SETPOINT(a)
29 → 7 → 8 Pressurizer Pressure						
a. Low	26 → (e) → 1(g)	16: 4 → [4]	Approved TSTF-135 → K; M	SR 3.3.1.1 SR 3.3.1.7 SR 3.3.1.10 ~~SR 3.3.1.16~~	16 → (h) → ≥ [1886] psig	~~≥ [1900] psig~~
b. High	1,2	16: 3 → [4]	7 → D → E	SR 3.3.1.1 SR 3.3.1.7 SR 3.3.1.10 ~~SR 3.3.1.16~~ (25)	16 → (i) → ≤ [2396] psig; 32 → 11	~~≤ [2385] psig~~
29 → 8 → 9 Pressurizer Water Level – High	26 → (e) → 1(g)	3	M; Approved TSTF-135 → K	SR 3.3.1.1 SR 3.3.1.7 SR 3.3.1.10	≤ [93.8]% 95% of span	~~≤ [92]%~~
29 → 9 → 10 Reactor Coolant Flow – Low						
a. Single Loop	26 → (f) → 1(h)	3 per loop	L → N	SR 3.3.1.1 SR 3.3.1.7 SR 3.3.1.10 ~~SR 3.3.1.16~~ (25)	16 → 90 → ≥ [89.2]%	~~≥ [90]%~~
b. Two Loops	26 → (g) → 1(i)	3 per loop	K → M	SR 3.3.1.1 SR 3.3.1.7 SR 3.3.1.10 (32 → 11) ~~SR 3.3.1.16~~ (25)	16 → 90 → ≥ [89.2]%	~~≥ [90]%~~

D RAI 3.3.1-1 Errata #145

(continued)

26

~~(a) Reviewer's Note: Unit specific implementations may contain only Allowable Value depending on Setpoint Study methodology used by the unit.~~

26

(e) → (g) Above the P-7 (Low Power Reactor Trips Block) interlock.

(f) → (h) Above the P-8 (Power Range Neutron Flux) interlock.

(g) → (i) Above the P-7 (Low Power Reactor Trips Block) interlock and below the P-8 (Power Range Neutron Flux) interlock.

D RAI 3.3.1-4 RAI 3.3.1-5 Errata #145

(h) ≥ 1905 psig during operation at 2250 psia, or ≥ 1800 psig during operation at 2000 psia.

(i) ≤ 2385 psig during operation at 2250 psia, or ≤ 2210 psig during operation at 2000 psia. ← 16

60 → RPS

Table 3.3.1-1 (page 4 of 8)
Reactor [Trip] System Instrumentation

Protection ← 60

15 → APPLICABLE MODES

27

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS	CONDITIONS	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE	TRIP SETPOINT(a)
[11] [10] Reactor Coolant Pump (RCP) Breaker Position (29)			Approved TSTF-135		16 → 3120 V	
a. Single Loop	1[h] (f) (26)	1 per RCP	[M] [O]	SR 3.3.1.[14] (55 → [13])	NA	NA
[11] b. Two Loops	1[(i)] (g)	1 per RCP (14)	[N] [M]	SR 3.3.1.[14]	NA	NA
[12] Undervoltage [RCPs] [Bus A01 and A02] (34)	1[(g)] (e) (26)	[2] [3] per bus	[M] Approved TSTF-135 [K]	SR 3.3.1.9 SR 3.3.1.10 ~~SR 3.3.1.16~~	≥ [4760] V [55.0]	≥ [4830] V
[13] Underfrequency [RCPs] [Bus A01 and A02] (29) [12]	1[(g)] (16)	[2] [3] per bus (73)	[E] [M]	~~SR 3.3.1.9~~ SR 3.3.1.10 ~~SR 3.3.1.16~~ (25)	≥ [57.1] Hz [20% of span] (16) (35)	≥ [57.5] Hz
[14] [13] Steam Generator (SG) Water Level – Low Low	1,2	[3 per SG] [4 per SG] (16)	[E] (7) [D]	SR 3.3.1.1 SR 3.3.1.7 SR 3.3.1.[10] ~~SR 3.3.1.16~~ (25)	≥ [30.4]%	≥ [32.3]%
[15] [14] SG Water Level – Low	1,2	2 per SG	[E] (7) [D]	SR 3.3.1.1 SR 3.3.1.7 SR 3.3.1.[10] ~~SR 3.3.1.16~~ (25)	[(j)] ← 70; ≥ [30.4]%; [11] ← 32	≥ [32.3]%
Coincident with Steam Flow/Feedwater Flow Mismatch	1,2	2 per SG	[E]	SR 3.3.1.1 SR 3.3.1.7 SR 3.3.1.[10] ~~SR 3.3.1.16~~ (25)	≤ [42.5]% full steam flow at RTP; 16 → [≤ 1 E6 lbm/hr]	≤ [40]% full steam flow at RTP

(continued)

D RAI 3.3.1-1 Errata #145

D RAI 3.3.1-26 RAI 3.3.1-1

26

~~(a) Reviewer's Note: Unit specific implementations may contain only Allowable Value depending on Setpoint Study methodology used by the unit.~~

26

[(e)] [(g)] Above the P-7 (Low Power Reactor Trips Block) interlock.

[(f)] [(h)] Above the P-8 (Power Range Neutron Flux) interlock.

[(g)] [(i)] Above the P-7 (Low Power Reactor Trips Block) interlock and below the P-8 (Power Range Neutron Flux) interlock.

[(j) Field Setting of ≥ 30% of span (nominal).] ← 70

D RAI 3.3.1-8 RAI 3.3.1-10 RAI 3.3.1-26 Errata #145

15 APPLICABLE MODES

Table 3.3.1-1 (page 5 of 8)
Reactor Trip Protection 60 System Instrumentation

27

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS	CONDITIONS	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE	TRIP SETPOINT(a)
29 15 16. Turbine Trip Autostop 36				55 14	(m) 70	
a. Low Fluid Oil Pressure	26 (k) 1(j)	37 3 2 4	O P	38 ~~SR 3.3.1.10~~ SR 3.3.1.15	≥ [750] psig	≥ [800] psig
b. Turbine Stop Valve Closure	1(j)	4	P	~~SR 3.3.1.10~~ SR 3.3.1.15	≥ [1]% open NA 16	≥ [1]% open
16 17. Safety Injection (SI) Input from Engineered Safety Feature Actuation System (ESFAS)	1,2	2 trains	Q Approved TSTF-135 P	SR 3.3.1.14 55 13	NA	NA
29 17 18. Reactor Trip System Interlocks	26	Approved TSTF-135		55 12	16 1E-10	
a. Intermediate Range Neutron Flux, P-6	2(d) (e)	2	R S	SR 3.3.1.11 SR 3.3.1.13	≥ [6E-11] amp	≥ [1E-10] amp
Replace with Insert 17.b 39						
~~b. Low Power Reactor Trips Block, P-7~~	~~1~~	~~1 per train~~	~~T~~	~~SR 3.3.1.11~~ ~~SR 3.3.1.13~~	~~NA~~	NA
c. Power Range Neutron Flux, P-8	1	4	Approved TSTF-135 S T	SR 3.3.1.11 SR 3.3.1.13 55 12	16 50 ≤ [50.2]% RTP	≤ [48]% RTP
d. Power Range Neutron Flux, P-9	1 (l) 40	4	S T	SR 3.3.1.11 SR 3.3.1.13 12	16 50 ≤ [52.2]% RTP	≤ [50]% RTP
e. Power Range Neutron Flux, P-10	1,2	4	Approved TSTF-135 R S	SR 3.3.1.11 SR 3.3.1.13 55 12	16 8 ≥ [7.8]% RTP and ≤ [12.2]% RTP 10	≥ [10]% RTP 16
39 ~~f. Turbine Impulse Pressure, P-13~~	~~1~~	~~2~~	~~T~~	~~[SR 3.3.1.1]~~ ~~SR 3.3.1.10~~ ~~SR 3.3.1.13~~	~~≤ [12.2]% turbine power~~	≤ [10]% turbine power

D RAI 3.3.1-1 RAI 3.3.1-27 Errata #145
D RAI 3.3.1-1 RAI 3.3.1-12 Errata #145
D RAI 3.3.1-1 RAI 3.3.1-5 RAI 3.3.1-14 RAI 3.3.1-27
D RAI 3.3.1-1 RAI 3.3.1-4 RAI 3.3.1-14

(continued)

26
26 ~~(a) Reviewer's Note: Unit specific implementations may contain only Allowable Value depending on Setpoint Study methodology used by the unit.~~

(d) (e) Below the P-6 (Intermediate Range Neutron Flux) interlocks.

(k) (j) Above the P-9 (Power Range Neutron Flux) interlock.

26 (l) With 1 of 2 circulating water pump breakers closed and vacuum ≥ 22"Hg. 40

(m) Field setting of ≥ 45 psig (nominal). 70

D RAI 3.3.1-7 RAI 3.3.1-12 RAI 3.3.1-26 RAI 3.3.1-27 Errata #145

Insert 17.a-02

NOT USED

Insert 17.b

(1) Power Range Neutron Flux	1	4	S	SR 3.3.1.11 SR 3.3.1.12	≤ 10% RTP
(2) Turbine Impulse Pressure	1	2	S	SR 3.3.1.11 SR 3.3.1.12	≤ 10% turbine power

Insert 17.d

Not used.

D

RAI 3.3.1-1
RAI 3.3.1-8
RAI 3.3.1-12
RAI 3.3.1-14
RAI 3.3.1-27
Errata #145

15

Table 3.3.1-1 (page 6 of 8)
Reactor Trip System Instrumentation

60 Protection

Approved TSTF-135

27

29 18 FUNCTION 42	15 APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS	CONDITIONS	Approved TSTF-135 SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE	~~TRIP SETPOINT(a)~~
19. Reactor Trip Breakers ~~(k)~~ (RTBs) 42	26 1, 2	2 ~~trains~~ 42 20	R Q	SR 3.3.1.4	NA	~~NA~~
	3(a), 4(a), 5(a) [(b) changed to (a)]	2 ~~trains~~	T C	SR 3.3.1.4	NA	~~NA~~
29 19 20. Reactor Trip Breaker Undervoltage and Shunt Trip Mechanisms	(a) 1, 2	1 each per RTB	U U Approved TSTF-135	SR 3.3.1.4	NA	~~NA~~
	3(a), 4(a), 5(a) [(b) changed to (a)]	1 each per RTB	T C	SR 3.3.1.4	NA	~~NA~~
21. Automatic Trip Logic	1, 2	2 trains	Q P Approved TSTF-135	SR 3.3.1.5 SR 3.3.1.15 33	NA	~~NA~~
	3(b), 4(b), 5(b)	2 trains	C X 44	SR 3.3.1.5	NA	~~NA~~

D RAI 3.3.1-1 RAI 3.3.1-26

26

~~(a) Reviewer's Note: Unit specific implementations may contain only Allowable Value depending on Setpoint Study methodology used by the unit.~~ 26

(a) (b) With RTBs closed and Rod Control System capable of rod withdrawal.

~~(k) Including any reactor trip bypass breakers that are racked in and closed for bypassing an RTB.~~ 42

(n) When a Reactor Trip Bypass Breaker is racked in and closed and the Rod Control System is capable of rod withdrawal.

D Errata #145

20. Reactor Trip Bypass Breaker and associated Undervoltage Trip Mechanism	1(n), 2(n)	1	V	SR 3.3.1.4	NA	NA
	3(n), 4(n), 5(n)	1	W	SR 3.3.1.4	NA	NA

43

60 RPS

Table 3.3.1-1 (page 7 of 8)
Reactor Trip System Instrumentation

Protection 60

Note 1: Overtemperature ΔT

Errata #70

The Overtemperature ΔT Function Allowable Value shall not exceed the following Trip Setpoint by more than [3.8]% of ΔT span.

$$\Delta T\frac{(1+\tau_1 s)}{(1+\tau_2 s)}\left(\frac{1}{1+\tau_3 s}\right) \le \Delta T_0\left\{K_1 - K_2\frac{(1+\tau_4 s)}{(1+\tau_5 s)}\left[T\frac{1}{(1+\tau_6 s)} - T'\right] + K_3(P - P') - f_1(\Delta I)\right\}$$

Where: ΔT is measured RCS ΔT, °F.
ΔT_0 is the indicated ΔT at RTP, °F.
s is the Laplace transform operator, sec $^{-1}$.
T is the measured RCS average temperature, °F.
T' is the nominal T_{avg} at RTP, ≤ [588]°F.

Approved TSTF 339

*

P is the measured pressurizer pressure, psig
P' is the nominal RCS operating pressure, ≤ [2235] psig

$K_1 \le$ [1.09] $K_2 \ge$ [0.0138]/°F K_3 = [0.000671]/psig
$\tau_1 \ge$ [8] sec $\tau_2 \le$ [3] sec $\tau_3 \le$ [2] sec
$\tau_4 \ge$ [33] sec $\tau_5 \le$ [4] sec $\tau_6 \le$ [2] sec

$f_1(\Delta I)$ = 1.26{35 + $(q_t - q_b)$} when $q_t - q_b \le$ - [35]% RTP
0% of RTP when -[35]% RTP < $q_t - q_b \le$ [7]% RTP
-1.05{$(q_t - q_b)$ - 7} when $q_t - q_b$ > [7]% RTP

Where q_t and q_b are percent RTP in the upper and lower halves of the core, respectively, and $q_t + q_b$ is the total THERMAL POWER in percent RTP.

The values denoted with [*] are specified in the COLR.

[*] Approved TSTF 339

Insert Note 1

45

60 RPS

Table 3.3.1-1 (page 8 of 8)
Reactor Trip System Instrumentation

Protection 60

Note 2: Overpower ΔT

The Overpower ΔT Function Allowable Value shall not exceed the following Trip Setpoint by more than [3]% of ΔT span.

$$\Delta T\frac{(1+\tau_1 s)}{(1+\tau_2 s)}\left(\frac{1}{1+\tau_3 s}\right) \le \Delta T_0\left\{K_4 - K_5\frac{\tau_7 s}{(1+\tau_7 s)}\left(\frac{1}{1+\tau_6 s}\right)T - K_6\left[T\frac{1}{1+\tau_6 s} - T''\right] - f_2(\Delta I)\right\}$$

Where: ΔT is measured RCS ΔT, °F.
ΔT_0 is the indicated ΔT at RTP, °F.
s is the Laplace transform operator, sec^{-1}.
T is the measured RCS average temperature, °F.
T" is the nominal T_{avg} at RTP, ≤ [588]°F.

K_4 ≤ [1.09] K_5 ≥ [0.02]/°F for increasing T_{avg} K_6 ≥ [0.00128]/°F when T > T"
[0]/°F for decreasing T_{avg} [0]/°F when T ≤ T"
τ_1 ≥ [8] sec τ_2 ≤ [3] sec τ_3 ≤ [2] sec
τ_6 ≤ [2] sec τ_7 ≥ [10] sec

$f_2(\Delta I)$ = 0% RTP for all ΔI.

The values denoted with [*] are specified in the COLR.

* Approved TSTF 339 [*]

Insert Note 2

45

D
Errata #70

(NUREG Markup for RPS Instrumentation 3.3.1)

Insert Note 1

$$\Delta T \left(\frac{1}{1+\tau_3 S}\right) \leq \Delta T_o \left(K_1 - K_2\left(T\left(\frac{1}{1+\tau_4 S}\right) - T'\right)\left(\frac{1+\tau_1 S}{1+\tau_2 S}\right) + K_3(P - P') - f(\Delta I)\right)$$

Errata #70

where (values are applicable to operation at both 2000 psia and 2250 psia unless otherwise indicated)

ΔT_o	=	indicated ΔT at rated power, °F
T	=	average temperature, °F
T'	≤	[*]°F (for cores containing 422V+ fuel assemblies)
T'	≤	[*]°F (for cores not containing 422V+ fuel assemblies)
P	=	pressurizer pressure, psig
P'	=	[*] psig (for 2250 psia operation)
P'	=	[*] psig (for 2000 psia operation and cores not containing 422V+ fuel assemblies)
K_1	≤	[*] (for 2250 psia operation and cores containing 422V+ fuel assemblies)
K_1	≤	[*] (for 2250 psia operation and cores not containing 422V+ fuel assemblies)
K_1	≤	[*] (for 2000 psia operation and cores not containing 422V+ fuel assemblies)
K_2	=	[*] (for 2250 psia operation and cores containing 422V+ fuel assemblies)
K_2	=	[*] (for 2250 psia operation and cores not containing 422V+ fuel assemblies)
K_2	=	[*] (for 2000 psia operation and cores not containing 422V+ fuel assemblies)
K_3	=	[*] (for 2250 psia operation and cores containing 422V+ fuel assemblies)
K_3	=	[*] (for 2250 psia operation and cores not containing 422V+ fuel assemblies)
K_3	=	[*] (for 2000 psia operation and cores not containing 422V+ fuel assemblies)
τ_1	=	[*] sec
τ_2	=	[*] sec
τ_3	=	[*] sec for Rosemont or equivalent RTD
	=	[*] sec for Sostman or equivalent RTD
τ_4	=	[*] sec for Rosemont or equivalent RTD
	=	[*] sec for Sostman or equivalent RTD

and $f(\Delta I)$ is an even function of the indicated difference between top and bottom detectors of the power-range nuclear ion chambers; with gains to be selected based on measured instrument response during plant startup tests, where q_t and q_b are the percent power in the top and bottom halves of the core respectively, and $q_t + q_b$ is total core power in percent of rated power, such that:

(a) for $q_t - q_b$ within –[*], +[*] percent, $f(\Delta I) = 0$ for cores not containing 422V+ fuel assemblies; for $q_t - q_b$ within –[*], +[*] percent, $f(\Delta I) = 0$ for cores containing 422V+ fuel assemblies.

(b) for each percent that the magnitude of $q_t - q_b$ exceeds +[*] percent, the ΔT trip setpoint shall be automatically reduced by an equivalent of [*] percent of rated power for cores not containing 422V+ fuel assemblies and reduced by an equivalent of [*] percent of rated power for cores containing 422V+ fuel assemblies.

(c) for cores not containing 422V+ fuel assemblies, for each percent that the magnitude of $q_t - q_b$ exceeds –[*] percent, the ΔT trip setpoint shall be automatically reduced by an equivalent of [*] percent of rated power; for cores containing 422V+ fuel assemblies, for each percent that the magnitude of $q_t - q_b$ exceeds –[*] percent, the ΔT trip setpoint shall be automatically reduced by an equivalent of [*] percent of rated power.

The values denoted with [*] are specified in the COLR.

(NUREG Markup for RPS Instrumentation 3.3.1)

Insert Note 2

D

Errata #70

$$\Delta T\left(\frac{1}{1+\tau_3 S}\right) \le \Delta T_o\left[K_4 - K_5\left(\frac{\tau_5 S}{\tau_5 S+1}\right)\left(\frac{1}{1+\tau_4 S}\right)T - K_6\left[T\left(\frac{1}{1+\tau_4 S}\right) - T'\right]\right]$$

where (values are applicable to operation at both 2000 psia and 2250 psia)

ΔT_o	=	indicated ΔT at rated power, °F
T	=	average temperature, °F
T'	≤	[*]°F (for cores containing 422V+ fuel assemblies)
T'	≤	[*]°F (for cores not containing 422V+ fuel assemblies)
K_4	≤	[*] of rated power (for cores containing 422V+ fuel assemblies)
K_4	≤	[*] of rated power (for cores not containing 422V+ fuel assemblies)
K_5	=	[*] for increasing T
	=	[*] for decreasing T
K_6	=	[*] for T ≥ T' (for cores containing 422V+ fuel assemblies)
K_6	=	[*] for T ≥ T' (for cores not containing 422V+ fuel assemblies)
	=	[*] for T < T'
τ_5	=	[*] sec
τ_3	=	[*] sec for Rosemont or equivalent RTD
		[*] sec for Sostman or equivalent RTD
τ_4	=	[*] sec for Rosemont or equivalent RTD
		[*] sec for Sostman or equivalent RTD

The values denoted with [*] are specified in the COLR.

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

60 RPS

this case, the RTS will still provide protection, even with random failure of one of the other three protection channels. Three ~~operable~~ [OPERABLE — Approved TSTF-135] instrumentation channels in a two-out-of-three configuration are generally required when there is no potential for control system and protection system interaction that could simultaneously create a need for RTS [60 RPS] trip and disable one RTS [60 RPS] channel. The two-out-of-three and two-out-of-four configurations allow one channel to be tripped during maintenance or testing without causing a reactor trip. Specific exceptions to the above general philosophy exist and are discussed below.

60 Protection

Reactor Trip System Functions

60 RPS

The safety analyses and OPERABILITY requirements applicable to each RTS Function are discussed below:

1. Manual Reactor Trip

The Manual Reactor Trip ensures that the control room operator can initiate a reactor trip at any time by using either of two [28 any one of four] reactor trip switches in the control room. A Manual Reactor Trip accomplishes the same results as any one of the automatic trip Functions. It is used by the reactor operator to shut down the reactor whenever any parameter is rapidly trending toward its Trip Setpoint. [Allowable Value 27]

D RAI 3.3.1-1

28

The LCO requires two Manual Reactor Trip channels to be OPERABLE. Each channel is controlled by a manual ~~reactor trip switch~~ [28 consists of two reactor trip switches (one in each train).]. Each channel activates the reactor trip breaker in both trains. Two independent channels are required to be OPERABLE so that no single random failure will disable the Manual Reactor Trip Function.

D RAI 3.3.1-1 Errata #126

In MODE 1 or 2, manual initiation of a reactor trip must be OPERABLE. These are the MODES in which the shutdown rods and/or control rods are partially or fully withdrawn from the core. In MODE 3, 4, or 5, the manual initiation Function must also be OPERABLE if the shutdown rods or control rods are withdrawn or the Control Rod Drive (CRD) System is capable of withdrawing the shutdown rods or the control rods. [18 with the RTBs closed and the Rod Control System capable of rod withdrawal.] In this condition, inadvertent control rod withdrawal is

D RAI 3.3.1-1 Errata #126

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

will terminate the reactivity excursion and shut down the reactor prior to reaching a power level that could damage the fuel. In MODE 3, 4, 5, or 6, the NIS power range detectors cannot detect neutron levels in this range. In these MODES, the Power Range Neutron Flux – High does not have to be OPERABLE because the reactor is shut down and reactivity excursions into the power range are extremely unlikely. Other RTS Functions and administrative controls provide protection against reactivity additions when in MODE 3, 4, 5, or 6.

D Errata #145

b. <u>Power Range Neutron Flux – Low</u>

The LCO requirement for the Power Range Neutron Flux – Low trip Function ensures that protection is provided against a positive reactivity excursion from low power or subcritical conditions.

The LCO requires all four of the Power Range Neutron Flux – Low channels to be OPERABLE.

In MODE 1, below the Power Range Neutron Flux (P-10 setpoint), and in MODE 2, the Power Range Neutron Flux – Low trip must be OPERABLE. This Function may be manually blocked by the operator when two out of four power range channels are greater than approximately 10% RTP (P-10 setpoint). This Function is automatically unblocked when three out of four power range channels are below the P-10 setpoint. Above the P-10 setpoint, positive reactivity additions are mitigated by the Power Range Neutron Flux – High trip Function.

D Errata #145

In MODE 3, 4, 5, or 6, the Power Range Neutron Flux – Low trip Function does not have to be OPERABLE because the reactor is shut down and the NIS power range detectors cannot detect neutron levels in this range. Other RTS trip Functions and administrative controls provide protection

60 RPS

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

against positive reactivity additions or power excursions in MODE 3, 4, 5, or 6.

D Errata #145

~~3. Power Range Neutron Flux Rate~~

~~The Power Range Neutron Flux Rate trips use the same channels as discussed for Function 2 above.~~

~~a. Power Range Neutron Flux - High Positive Rate~~

~~The Power Range Neutron Flux - High Positive Rate trip Function ensures that protection is provided against rapid increases in neutron flux that are characteristic of an RCCA drive rod housing rupture and the accompanying ejection of the RCCA. This Function compliments the Power Range Neutron Flux - High and Low Setpoint trip Functions to ensure that the criteria are met for a rod ejection from the power range.~~

29 ~~The LCO requires all four of the Power Range Neutron Flux - High Positive Rate channels to be OPERABLE.~~

~~In MODE 1 or 2, when there is a potential to add a large amount of positive reactivity from a rod ejection accident (REA), the Power Range Neutron Flux - High Positive Rate trip must be OPERABLE. In MODE 3, 4, 5, or 6, the Power Range Neutron Flux - High Positive Rate trip Function does not have to be OPERABLE because other RTS trip Functions and administrative controls will provide protection against positive reactivity additions. Also, since only the shutdown banks may be withdrawn in MODE 3, 4, or 5, the remaining complement of control bank worth ensures a sufficient degree of SDM in the event of an REA. In MODE 6, no rods are withdrawn and the SDM is increased during refueling operations. The reactor vessel head is also removed or the closure bolts are detensioned preventing any pressure buildup. In addition, the NIS power range detectors cannot detect neutron levels present in this mode.~~

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

22 do not provide any input to control systems. ~~Note that this Function also provides a signal to prevent automatic and manual rod withdrawal prior to initiating a reactor trip. Limiting further rod withdrawal may terminate the transient and eliminate the need to trip the reactor.~~

The LCO requires two channels of Intermediate Range Neutron Flux to be OPERABLE. Two OPERABLE channels are sufficient to ensure no single random failure will disable this trip Function. D RAI 3.3.1-1

Because this trip Function is important only during startup, there is generally no need to disable channels for testing while the Function is required to be OPERABLE. Therefore, a third channel is unnecessary. D RAI 3.3.1-1

In MODE 1 below the P-10 setpoint, and in MODE 2, when there is a potential for an uncontrolled RCCA bank rod withdrawal accident during reactor startup, the Intermediate Range Neutron Flux trip must be OPERABLE. Above the P-10 setpoint, the Power Range Neutron Flux-High Setpoint trip ~~and the Power Range Neutron Flux-High Positive Rate trip~~ 29 provides core protection for a rod withdrawal accident. In MODE 3, 4, or 5, the Intermediate Range Neutron Flux trip does not have to be OPERABLE because the control rods must be fully inserted and only the shutdown rods may be withdrawn. 29 The reactor cannot be started up in this condition. The core also has the required SDM to mitigate the consequences of a positive reactivity addition accident. In MODE 6, all rods are fully inserted and the core has a required increased SDM. Also, the NIS intermediate range detectors cannot detect neutron levels present in this MODE. D Errata #145

29 4 5 Source Range Neutron Flux

The LCO requirement for the Source Range Neutron Flux trip Function ensures that protection is provided against an uncontrolled RCCA bank rod withdrawal accident from a subcritical condition during startup. This trip Function provides redundant protection to

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

Approved TSTF-135

the Power Range Neutron Flux – Low ~~Setpoint and Intermediate Range Neutron Flux~~ trip Function~~s~~. In MODES 3, 4, and 5, administrative controls also prevent the uncontrolled withdrawal of rods. The NIS source range detectors are located external to the reactor vessel and measure neutrons leaking from the core. The NIS source range detectors do not provide any inputs to control systems. The source range trip is the only RTS automatic protection function required in MODES 3, 4, and 5. Therefore, the functional capability at the specified Trip Setpoint is assumed to be available.

60 RPS

Errata #145

RAI 3.3.1-1

The LCO requires two channels of Source Range Neutron Flux to be OPERABLE. Two OPERABLE channels are sufficient to ensure no single random failure will disable this trip Function. ~~The LCO also requires one channel of the Source Range Neutron Flux to be OPERABLE in MODE 3, 4, or 5 with RTBs open. In this case, the source range Function is to provide control room indication and input to the Boron Dilution Protection System (BDPS). The outputs of the Function to RTS logic are not required OPERABLE when the RTBs are open.~~

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48

The Source Range Neutron Flux Function provides protection for control rod withdrawal from subcritical~~, boron dilution~~ and control rod ejection events. ~~The Function also provides visual neutron flux indication in the control room.~~

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RAI 3.3.1-12

In MODE 2 when below the P-6 setpoint ~~during a reactor startup~~ the Source Range Neutron Flux trip must be OPERABLE. Above the P-6 setpoint, the Intermediate Range Neutron Flux trip and the Power Range Neutron Flux – Low Setpoint trip will provide core protection for reactivity accidents. Above the P-6 setpoint, the NIS source range detectors are de-energized ~~and inoperable~~.

and in MODES 3, 4 and 5 when there is a potential for an uncontrolled RCCA bank rod withdrawal accident

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In MODES 3, 4 and 5 with the Rod Control System not capable of rod withdrawal, and in MODE 6, the Function is not required to be OPERABLE.

In MODE 3, 4, or 5 with the reactor shut down, the Source Range Neutron Flux trip Function must also be OPERABLE. If the CRD System is capable of rod withdrawal, the Source Range Neutron Flux trip must be

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued) RPS 60

channels shared with other RTS Functions. Failures that affect multiple Functions require entry into the Conditions applicable to all affected Functions.

In MODE 1 or 2, the Overpower ΔT trip Function must be OPERABLE. These are the only times that enough heat is generated in the fuel to be concerned about the heat generation rates and overheating of the fuel. In MODE 3, 4, 5, or 6, this trip Function does not have to be OPERABLE because the reactor is not operating and there is insufficient heat production to be concerned about fuel overheating and fuel damage.

29 7 8 Pressurizer Pressure

The same sensors provide input to the Pressurizer Pressure – High and – Low trips and the Overtemperature ΔT trip. 16 ~~At some units,~~ the Pressurizer Pressure channels are also used to provide input to the Pressurizer Pressure Control System. ~~For those units,~~ the actuation logic must be able to withstand an input failure to the control system, which may then require the protection function actuation, and a single failure in the other channels providing the protection function actuation.

a. Pressurizer Pressure – Low

The Pressurizer Pressure – Low trip Function ensures that protection is provided against violating the DNBR limit due to low pressure.

16 The LCO requires four channels ~~for two and four loop units (three channels for three loop units)~~ of Pressurizer Pressure – Low to be OPERABLE.

In MODE 1, when DNB is a major concern, the Pressurizer Pressure – Low trip must be OPERABLE. This trip Function is automatically enabled on increasing power by the P-7 interlock (NIS power range P-10 or turbine impulse pressure greater than approximately 10% of full power equivalent D RAI 3.3.1-4

60 → RPS

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

29 → 8 → 9

Pressurizer Water Level - High

The Pressurizer Water Level - High trip Function provides a backup signal for the Pressurizer Pressure - High trip and also provides protection against water relief through the pressurizer safety valves. These valves are designed to pass steam in order to achieve their design energy removal rate. A reactor trip is actuated prior to the pressurizer becoming water solid. The LCO requires three channels of Pressurizer Water Level - High to be OPERABLE. The pressurizer level channels are used as input to the Pressurizer Level Control System. A fourth channel is not required to address control/protection interaction concerns. The level channels do not actuate the safety valves, and the high pressure reactor trip is set below the safety valve setting. Therefore, with the slow rate of charging available, pressure overshoot due to level channel failure cannot cause the safety valve to lift before reactor high pressure trip.

In MODE 1, when there is a potential for overfilling the pressurizer, the Pressurizer Water Level - High trip must be OPERABLE. This trip Function is automatically enabled on increasing power by the P-7 interlock. On decreasing power, this trip Function is automatically blocked below P-7. Below the P-7 setpoint, transients that could raise the pressurizer water level will be slow and the operator will have sufficient time to evaluate unit conditions and take corrective actions. RAI 3.3.1-4

61 → interlock

29 → 9 → 10

Reactor Coolant Flow - Low

a. Reactor Coolant Flow - Low (Single Loop)

The Reactor Coolant Flow - Low (Single Loop) trip Function ensures that protection is provided against violating the DNBR limit due to low flow in one or more RCS loops, while avoiding reactor trips due to normal variations in loop flow. Above the P-8 setpoint, which is approximately 48% RTP, a loss of flow in any RCS loop will actuate a reactor trip. Each RCS loop has three flow detectors to monitor flow. The flow signals are not used for any control system input. RAI 3.3.1-5

16 → 50% → 48%

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

The LCO requires three Reactor Coolant Flow – Low channels per loop to be OPERABLE in MODE 1 above P-8.

16 29 9

In MODE 1 above the P-8 setpoint, a loss of flow in one RCS loop could result in DNB conditions in the core. In MODE 1 below the P-8 setpoint, a loss of flow in two ~~or more~~ loops is required to actuate a reactor trip (Function 10b) because of the lower power level and the greater margin to the design limit DNBR.

b. Reactor Coolant Flow – Low (Two Loops)

16

The Reactor Coolant Flow – Low (Two Loops) trip Function ensures that protection is provided against violating the DNBR limit due to low flow in two ~~or more~~ RCS loops while avoiding reactor trips due to normal variations in loop flow.

61 interlock 16 D RAI 3.3.1-4

Above the P-7 setpoint and below the P-8 setpoint, a loss of flow in two ~~or more~~ loops will initiate a reactor trip. Each loop has three flow detectors to monitor flow. The flow signals are not used for any control system input.

The LCO requires three Reactor Coolant Flow – Low channels per loop to be OPERABLE.

61 interlock D RAI 3.3.1-4 61 interlock 16 16

In MODE 1 above the P-7 setpoint and below the P-8 setpoint, the Reactor Coolant Flow – Low (Two Loops) trip must be OPERABLE. Below the P-7 setpoint, all reactor trips on low flow are automatically blocked since no conceivable power distributions could occur that would cause a DNB concern at this low power level. Above the P-7 setpoint the reactor trip on low flow in two ~~or more~~ RCS loops is automatically enabled. Above the P-8 setpoint, a loss of flow in any one loop will actuate a reactor trip because of the higher power level and the reduced margin to the design limit DNBR.

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

29 10 11

Reactor Coolant Pump (RCP) Breaker Position

Both RCP Breaker Position trip Functions operate together on two sets of auxiliary contacts, with one set on each RCP breaker. These Functions anticipate the Reactor Coolant Flow - Low trips to avoid RCS heatup that would occur before the low flow trip actuates.

a. Reactor Coolant Pump Breaker Position (Single Loop)

The RCP Breaker Position (Single Loop) trip Function ensures that protection is provided against violating the DNBR limit due to a loss of flow in one RCS loop. The position of each RCP breaker is monitored. If one RCP breaker is open above the P-8 setpoint, a reactor trip is initiated. This trip Function will generate a reactor trip before the Reactor Coolant Flow - Low (Single Loop) Trip Setpoint is reached. D RAI 3.3.1-10

62

A channel consists of the RCP Breaker auxiliary contact and the associated RCP Loss of Power Trip Matrix Relay.

The LCO requires one RCP Breaker Position channel per RCP to be OPERABLE. One OPERABLE channel is sufficient for this trip Function because the RCS Flow-Low trip alone provides sufficient protection of unit SLs for loss of flow events. The RCP Breaker Position trip serves only to anticipate the low flow trip, minimizing the thermal transient associated with loss of a pump.

This Function measures only the discrete position (open or closed) of the RCP breaker, using a position switch. Therefore, the Function has no adjustable trip setpoint with which to associate an LSSS.

In MODE 1 above the P-8 setpoint, when a loss of flow in any RCS loop could result in DNB conditions in the core, the RCP Breaker Position (Single Loop) trip must be OPERABLE. In MODE 1 below the P-8 setpoint, a loss of flow in two ~~or~~ 16 16 ~~more~~ loops is required to actuate a reactor trip because of the lower power level and the greater margin to the design limit DNBR.

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

b. Reactor Coolant Pump Breaker Position (Two Loops)

16 → The RCP Breaker Position (Two Loops) trip Function ensures that protection is provided against violating the DNBR limit due to a loss of flow in two ~~or more~~ RCS loops. The position of each RCP breaker is monitored. Above the P-7 setpoint and below the P-8 setpoint, a loss of flow in two ~~or more~~ loops will initiate a reactor trip. This trip Function will generate a reactor trip before the Reactor Coolant Flow - Low (Two Loops) Trip Setpoint is reached.

61 → interlock → setpoint

16

D RAI 3.3.1-10

62 → A channel consists of the RCP Breaker auxiliary contact and the associated RCP Loss of Power Trip Matrix Relay.

The LCO requires one RCP Breaker Position channel per RCP to be OPERABLE. One OPERABLE channel is sufficient for this Function because the RCS Flow - Low trip alone provides sufficient protection of unit SLs for loss of flow events. The RCP Breaker Position trip serves only to anticipate the low flow trip, minimizing the thermal transient associated with loss of an RCP.

This Function measures only the discrete position (open or closed) of the RCP breaker, using a position switch. Therefore, the Function has no adjustable trip setpoint with which to associate an LSSS.

D RAI 3.3.1-10

61 → interlock → setpoint

In MODE 1 above the P-7 setpoint and below the P-8 setpoint, the RCP Breaker Position (Two Loops) trip must be OPERABLE. Below the P-7 setpoint all reactor trips on loss of flow are automatically blocked since no conceivable power distributions could occur that would cause a DNB concern at this low power level. Above the P-7 setpoint the reactor trip on loss of flow in two RCS loops is automatically enabled. Above the P-8 setpoint, a loss of flow in any one loop will actuate a reactor trip because of the higher power level and the reduced margin to the design limit DNBR.

61 → interlock → setpoint

60 → RPS

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

29 → 11 → 12

Undervoltage Reactor Coolant Pumps

34 → Bus A01 and A02

16 → both

61 → interlock

D RAI 3.3.1-8

The Undervoltage RCPs reactor trip Function ensures that protection is provided against violating the DNBR limit due to a loss of flow in two or more RCS loops. The voltage to each RCP is monitored. Above the P-7 setpoint a loss of voltage detected on two or more ~~RCP~~ buses will initiate a reactor trip. This trip Function will generate a reactor trip before the Reactor Coolant Flow – Low (Two Loops) Trip Setpoint is reached. Time delays are incorporated into the Undervoltage RCPs channels to prevent reactor trips due to momentary electrical power transients.

16 → two

The LCO requires three Undervoltage ~~RCPs~~ channels ~~(one per phase)~~ per bus to be OPERABLE.

61 → interlock

In MODE 1 above the P-7 setpoint, the Undervoltage RCP trip must be OPERABLE. Below the P-7 setpoint, all reactor trips on loss of flow are automatically blocked since no conceivable power distributions could occur that would cause a DNB concern at this low power level. Above the P-7 setpoint, the reactor trip on loss of flow in two or more RCS loops is automatically enabled. ~~This Function uses the same relays as the ESFAS Function 6.f, "Undervoltage Reactor Coolant Pump (RCP)" start of the auxiliary feedwater (AFW) pumps.~~

59

29 → 12 → 13

Underfrequency Reactor Coolant Pumps

34 → Bus A01 and A02

The Underfrequency RCPs reactor trip Function ensures that protection is provided against violating the DNBR limit due to a loss of flow in two ~~or more~~ RCS loops from a major network frequency disturbance. An underfrequency condition will slow down the pumps, thereby reducing their coastdown time following a pump trip. The proper coastdown time is required so that reactor heat can be removed immediately after reactor trip. The frequency of each RCP bus is monitored. Above the P-7 setpoint, a loss of frequency detected on two ~~or more~~ RCP buses will initiate a reactor trip. This trip Function will generate a reactor trip before the Reactor Coolant Flow – Low (Two Loops) Trip Setpoint is reached. Time delays are incorporated into the Underfrequency RCPs channels to prevent reactor trips due to momentary electrical power transients.

50 → trip both RCP breakers

50 → Tripping both RCP breakers

D RAI 3.3.1-8

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

The LCO requires two Underfrequency Bus A01 channels and two Underfrequency Bus A02 channels to be OPERABLE.

In MODE 1 above the P-7 interlock, the Underfrequency Bus A01 and A02 RCP breaker trip must be OPERABLE. Below the P-7 interlock, this trip and all reactor trips on loss of flow are automatically blocked, because no conceivable power distributions could occur that would cause a DNB concern at this low power level. Above the P-7 interlock, the Underfrequency Bus A01 and A02 RCP breaker trip is automatically enabled.

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14
13
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D
RAI 3.3.1-1

The LCO requires three Underfrequency RCPs channels per bus to be OPERABLE.

In MODE 1 above the P-7 setpoint, the Underfrequency RCPs trip must be OPERABLE. Below the P-7 setpoint, all reactor trips on loss of flow are automatically blocked since no conceivable power distributions could occur that would cause a DNB concern at this low power level. Above the P-7 setpoint, the reactor trip on loss of flow in two or more RCS loops is automatically enabled.

Steam Generator Water Level – Low Low

The SG Water Level – Low Low trip Function ensures that protection is provided against a loss of heat sink and actuates the AFW System prior to uncovering the SG tubes. The SGs are the heat sink for the reactor. In order to act as a heat sink, the SGs must contain a minimum amount of water. A narrow range low low level in any SG is indicative of a loss of heat sink for the reactor. The level transmitters provide input to the SG Level Control System. Therefore, the actuation logic must be able to withstand an input failure to the control system, which may then require the protection function actuation, and a single failure in the other channels providing the protection function actuation. This Function also performs the ESFAS function of starting the AFW pumps on low low SG level.

three 16

The LCO requires four channels of SG Water Level – Low Low per SG to be OPERABLE ~~for four loop units in which these channels are shared between protection and control. In two, three, and four loop units where three SG Water Levels are dedicated to the RTS, only three channels per SG are required to be OPERABLE~~.

16

In MODE 1 or 2, when the reactor requires a heat sink, the SG Water Level – Low Low trip must be OPERABLE. The normal source of water for the SGs is the Main Feedwater (MFW) System (not safety related). The MFW System is only in operation in MODE 1 or 2. The AFW System is the safety related backup source of water to

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

ensure that the SGs remain the heat sink for the reactor. During normal startups and shutdowns, the AFW System provides feedwater to maintain SG level. In MODE 3, 4, 5, or 6, the SG Water Level - Low Low Function does not have to be OPERABLE because the MFW System is not in operation and the reactor is not operating or even critical. Decay heat removal is accomplished by the AFW System in MODE 3 and by the Residual Heat Removal (RHR) System in MODE 4, 5, or 6.

29 14 15

Steam Generator Water Level - Low, Coincident With Steam Flow/Feedwater Flow Mismatch

63

SG Water Level - Low, in conjunction with the Steam Flow/Feedwater Flow Mismatch, ensures that protection is provided against a loss of heat sink ~~and actuates the AFW System prior to uncovering the SG tubes~~. In addition to a decreasing water level in the SG, the difference between feedwater flow and steam flow is evaluated to determine if feedwater flow is significantly less than steam flow. With less feedwater flow than steam flow, SG level will decrease at a rate dependent upon the magnitude of the difference in flow rates. There are two SG level channels and two Steam Flow/Feedwater Flow Mismatch channels per SG. One narrow range level channel sensing a low level coincident with one Steam Flow/Feedwater Flow Mismatch channel sensing flow mismatch (steam flow greater than feed flow) will actuate a reactor trip.

70

The field setting for SG Water Level-Low was developed outside of the setpoint methodology and has been provided by the NSSS supplier. No Analytical Value is assumed in the accident analysis for this function.

The LCO requires two channels of SG Water Level - Low coincident with Steam Flow/Feedwater Flow Mismatch. per SG

54 D RAI 3.3.1-1

In MODE 1 or 2, when the reactor requires a heat sink, the SG Water Level - Low coincident with Steam Flow/Feedwater Flow Mismatch trip must be OPERABLE. The normal source of water for the SGs is the MFW System (not safety related). The MFW System is only in operation in MODE 1 or 2. The AFW System is the safety related backup source of water to ensure that the SGs remain the heat sink for the reactor. During normal startups and shutdowns, the AFW System provides feedwater to maintain SG level. In MODE 3, 4, 5,

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

or 6, the SG Water Level – Low coincident with Steam Flow/Feedwater Flow Mismatch Function does not have to be OPERABLE because the MFW System is not in operation and the reactor is not operating or even critical. Decay heat removal is accomplished by the AFW System in MODE 3 and by the RHR System in MODE 4, 5, or 6. The MFW System is in operation only in MODE 1 or 2 and, therefore, this trip Function need only be OPERABLE in these MODES.

29 15 16

Turbine Trip

a. Turbine Trip – Low Fluid Oil Pressure Autostop 36

D RAI 3.3.1-7

The Turbine Trip – Low Fluid Oil Pressure trip Function anticipates the loss of heat removal capabilities of the secondary system following a turbine trip. This trip Function acts to minimize the pressure/temperature transient on the reactor. Any turbine trip from a power level below the P-9 setpoint, approximately 50% power, will not actuate a reactor trip. Three pressure switches monitor the control oil pressure in the Turbine Electrohydraulic Control System. A low pressure condition sensed by two-out-of-three pressure switches will actuate a reactor trip. These pressure switches do not provide any input to the control system. The unit is designed to withstand a complete loss of load and not sustain core damage or challenge the RCS pressure limitations. Core protection is provided by the Pressurizer Pressure – High trip Function and RCS integrity is ensured by the pressurizer safety valves.

D RAI 3.3.1-27

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with at least one circulating water pump breaker closed and condenser pressure not high,

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The field setting for Turbine Trip – Low Autostop Oil Pressure was developed outside of the setpoint methodology and has been provided by the NSSS supplier. No Analytical Value is assumed in the accident analysis for this function.

D RAI 3.3.1-7

36 Autostop

The LCO requires three channels of Turbine Trip – Low Fluid Oil Pressure to be OPERABLE in MODE 1 above P-9.

Below the P-9 setpoint, a turbine trip does not actuate a reactor trip. In MODE 2, 3, 4, 5, or 6, there is no potential for a turbine trip, and the

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

Autostop 36

Turbine Trip – Low Fluid Oil Pressure trip Function does not need to be OPERABLE.

b. Turbine Trip – Turbine Stop Valve Closure

D RAI 3.3.1-7

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with at least one circulating water pump breaker closed and condenser pressure not high,

The Turbine Trip – Turbine Stop Valve Closure trip Function anticipates the loss of heat removal capabilities of the secondary system following a turbine trip from a power level below the P-9 setpoint, approximately 50% power. This action will not actuate a reactor trip. The trip Function anticipates the loss of secondary heat removal capability that occurs when the stop valves close. Tripping the reactor in anticipation of loss of secondary heat removal acts to minimize the pressure and temperature transient on the reactor. This trip Function will not and is not required to operate in the presence of a single channel failure. The unit is designed to withstand a complete loss of load and not sustain core damage or challenge the RCS pressure limitations. Core protection is provided by the Pressurizer Pressure – High trip Function, and RCS integrity is ensured by the pressurizer safety valves. This trip Function is diverse to the Turbine Trip – Low Fluid Oil Pressure trip Function. Each turbine stop valve is equipped with one limit switch that inputs to the RTS. If all four limit switches indicate that the stop valves are all closed, a reactor trip is initiated.

D RAI 3.3.1-27

36 Autostop

37 both

No Analytical Value is assumed in the accident analysis for this function.

3

The LSSS for this Function is set to assure channel trip occurs when the associated stop valve is completely closed.

two 37

The LCO requires four Turbine Trip – Turbine Stop Valve Closure channels, one per valve, to be OPERABLE in MODE 1 above P-9. All four channels must trip to cause reactor trip.

Both 37

D RAI 3.3.1-8

Below the P-9 setpoint, a load rejection can be accommodated by the Steam Dump System. In MODE 2, 3, 4, 5, or 6, there is no potential for a load

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

when the associated reactor trip functions are outside the applicable MODES. These are:

a. Intermediate Range Neutron Flux, P -6

The Intermediate Range Neutron Flux, P -6 interlock is actuated when any NIS intermediate range channel goes approximately one decade above the minimum channel reading. If both channels drop below the setpoint, the permissive will automatically be defeated. The LCO requirement for the P-6 interlock ensures that the following Functions are performed:

- on increasing power, the P -6 interlock allows the manual block of the NIS Source Range, Neutron Flux reactor trip. This prevents a premature block of the source range trip and allows the operator to ensure that the intermediate range is OPERABLE prior to leaving the source range. When the source range trip is blocked, the high voltage to the detectors is also removed;

- on decreasing power, the P -6 interlock automatically energizes the NIS source range detectors and enables the NIS Source Range Neutron Flux reactor trip ~~; and~~ 51

- ~~on increasing power, the P -6 interlock provides a backup block signal to the source range flux doubling circuit. Normally, this Function is manually blocked by the control room operator during the reactor startup.~~

D
RAI 3.3.1-12
Errata #145

The LCO requires two channels of Intermediate Range Neutron Flux, P -6 interlock to be OPERABLE in MODE 2 when below the P -6 interlock setpoint.

Above the P -6 interlock setpoint, the NIS Source Range Neutron Flux reactor trip will be blocked, and this Function will no longer be necessary.

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

In MODE 3, 4, 5, or 6, the P-6 interlock does not have to be OPERABLE because the NIS Source Range is providing core protection. D Errata #145

b. Low Power Reactor Trips Block, P-7 39

The Low Power Reactor Trips Block, P-7 interlock is actuated by input from either ~~the~~ Power Range Neutron Flux, ~~P-10,~~ or ~~the~~ Turbine Impulse Pressure~~, P-13 interlock~~. The LCO requirement for the P-7 interlock ensures that the following Functions are performed: 41 D RAI 3.3.1-14

(1) on increasing power, the P-7 interlock automatically enables reactor trips on the following Functions:

- Pressurizer Pressure - Low;
- Pressurizer Water Level - High;
- Reactor Coolant Flow - Low (Two Loops);
- RCPs Breaker Open (Two Loops);
- Undervoltage RCPs; and Bus A01 and A02
- Underfrequency RCPs. 34

These reactor trips are only required when operating above the P-7 setpoint (approximately 10% power). The reactor trips provide protection against violating the DNBR limit. Below the P-7 setpoint, the RCS is capable of providing sufficient natural circulation without any RCP running.

(2) on decreasing power, the P-7 interlock automatically blocks reactor trips on the following Functions:

- Pressurizer Pressure - Low;

60 RPS

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

- Pressurizer Water Level – High;
- Reactor Coolant Flow – Low (Two Loops);
- RCP Breaker Position (Two Loops);
- Undervoltage RCPs; and
- Underfrequency RCPs.

Bus A01 and A02

34

D

RAI 3.3.1-14

39

~~Trip Setpoint and Allowable Value are not applicable to the P-7 interlock because it is a logic Function and thus has no parameter with which to associate an LSSS.~~

~~The P-7 interlock is a logic Function with train and not channel identity. Therefore, the LCO requires one channel per train of Low Power Reactor Trips Block, P-7 interlock to be OPERABLE in MODE 1.~~

The low power trips are blocked below the P-7 setpoint and unblocked above the P-7 setpoint. In MODE 2, 3, 4, 5, or 6, this Function does not have to be OPERABLE because the interlock performs its Function when power level drops below 10% power, which is in MODE 1.

39 Insert 17.b

c. Power Range Neutron Flux, P-8

16 50

The Power Range Neutron Flux, P-8 interlock is actuated at approximately 48% power as determined by two-out-of-four NIS power range detectors. The P-8 interlock automatically enables the Reactor Coolant Flow – Low (Single Loop) and RCP Breaker Position (Single Loop) reactor trips ~~on~~ ~~low flow in one or more RCS loops~~ on increasing power. The LCO requirement for this trip Function ensures that protection is provided against a loss of flow in any RCS loop that could result in DNB conditions in the core when greater than approximately 48% power. On decreasing

64

64

16 50

Insert 17.b

1. Power Range Neutron Flux

Power Range Neutron Flux is actuated by two-out-of-four NIS power range channels. The LCO requirement for this function ensures that this input to the P-7 interlock is available.

The LCO requires four channels of Power Range Neutron Flux to be OPERABLE in MODE 1.

OPERABILITY in MODE 1 ensures the Function is available to perform its increasing power Functions.

2. Turbine Impulse Pressure

The Turbine Impulse Pressure interlock is actuated when the pressure in the first stage of the high pressure turbine is greater than approximately 10% of the rated full power pressure. This is determined by one -out-of-two pressure detectors. The LCO requirement for this Function ensures that one of the inputs to the P-7 interlock is available.

The LCO requires two channels of Turbine Impulse Pressure interlock to be OPERABLE in MODE 1.

The Turbine Impulse Chamber Pressure interlock must be OPERABLE when the turbine generator is operating. The interlock Function is not required OPERABLE in MODE 2, 3, 4, 5, or 6 because the turbine generator is not operating.

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

power, the reactor trip on low flow in any loop is automatically blocked.

The LCO requires four channels of Power Range Neutron Flux, P-8 interlock to be OPERABLE in MODE 1.

In MODE 1, a loss of flow in one RCS loop could result in DNB conditions, so the Power Range Neutron Flux, P-8 interlock must be OPERABLE. In MODE 2, 3, 4, 5, or 6, this Function does not have to be OPERABLE because the core is not producing sufficient power to be concerned about DNB conditions. D RAI 3.3.1-14

d. Power Range Neutron Flux, P-9

40 if the Steam Dump System is available

36 Autostop

The Power Range Neutron Flux, P-9 interlock is actuated at approximately 50% power as determined by two-out-of-four NIS power range detectors. The LCO requirement for this Function ensures that the Turbine Trip – Low Fluid Oil Pressure and Turbine Trip – Turbine Stop Valve Closure reactor trips are enabled above the P-9 setpoint. Above the P-9 setpoint, a turbine trip will cause a load rejection beyond the capacity of the Steam Dump System. A reactor trip is automatically initiated on a turbine trip when it is above the P-9 setpoint, to minimize the transient on the reactor. D RAI 3.3.1-27

The LCO requires four channels of Power Range Neutron Flux, P-9 interlock to be OPERABLE in MODE 1.

with one of two circulating water pump breakers closed and condenser vacuum ≥ 22 "Hg.

40

In MODE 1, a turbine trip could cause a load rejection beyond the capacity of the Steam Dump System, so the Power Range Neutron Flux interlock must be OPERABLE. In MODE 2, 3, 4, 5, or 6, this Function does not have to be OPERABLE because the reactor is not at a power level sufficient to have a load rejection beyond the capacity of the Steam Dump System.

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

D RAI 3.3.1-14

e. Power Range Neutron Flux, P-10

The Power Range Neutron Flux, P-10 interlock is actuated at approximately 10% power, as determined by two-out-of-four NIS power range detectors. If power level falls below 10% RTP on 3 of 4 channels, the nuclear instrument trips will be automatically unblocked. The LCO requirement for the P-10 interlock ensures that the following Functions are performed:

- on increasing power, the P-10 interlock allows the operator to manually block the Intermediate Range Neutron Flux reactor trip. 65 ~~Note that blocking the reactor trip also blocks the signal to prevent automatic and manual rod withdrawal;~~
- on increasing power, the P-10 interlock allows the operator to manually block the Power Range Neutron Flux – Low reactor trip;
- on increasing power, the P-10 interlock automatically provides a backup signal to block the Source Range Neutron Flux reactor trip, and also to de-energize the NIS source range detectors;
- 39 ~~the P-10 interlock provides one of the two inputs to the P-7 interlock; and~~ D RAI 3.3.1-14
- on decreasing power, the P-10 interlock automatically enables the Power Range Neutron Flux – Low reactor trip and the Intermediate Range Neutron Flux reactor trip 65 ~~(and rod stop)~~.

The LCO requires four channels of Power Range Neutron Flux, P-10 interlock to be OPERABLE in MODE 1 or 2.

OPERABILITY in MODE 1 ensures the Function is available to perform its decreasing power Functions in the event of a reactor shutdown. This Function must be OPERABLE in MODE 2 to ensure that core protection is provided during a

APPLICABLE SAFETY ANALYSES, LCO, and APPLICABILITY (continued)

startup or shutdown by the Power Range Neutron Flux–Low and Intermediate Range Neutron Flux reactor trips. In MODE 3, 4, 5, or 6, this Function does not have to be OPERABLE because the reactor is not at power and the Source Range Neutron Flux reactor trip provides core protection.

~~f. Turbine Impulse Pressure~~

~~The Turbine Impulse Pressure interlock is actuated when the pressure in the first stage of the high pressure turbine is greater than approximately 10% of the rated full power pressure. This is determined by one -out-of-two pressure detectors. The LCO requirement for this Function ensures that one of the inputs to the P-7 interlock is available.~~

~~The LCO requires two channels of Turbine Impulse Pressure interlock to be OPERABLE in MODE 1.~~

~~The Turbine Impulse Chamber Pressure interlock must be OPERABLE when the turbine generator is operating. The interlock Function is not required OPERABLE in MODE 2, 3, 4, 5, or 6 because the turbine generator is not operating.~~

D

RAI 3.3.1-14

29 18 19. Reactor Trip Breakers

This trip Function applies to the RTBs exclusive of individual trip mechanisms. The LCO requires two OPERABLE RTBs trains of trip breakers. ~~A trip breaker train consists of all trip breakers associated with a single RTS logic train that are racked in, closed, and capable of supplying power to the CRD System. Thus, the train may consist of the main breaker, bypass breaker, or main breaker and bypass breaker, depending upon the system configuration.~~ 42 Two OPERABLE trains RTBs ensure no single random failure can disable the RTS trip capability.

42

ACTIONS (continued)

at the same time. The Required Action is to refer to Table 3.3.1-1 and to take the Required Actions for the protection functions affected. The Completion Times are those from the referenced Conditions and Required Actions.

D RAI 3.3.1-1

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B.1 ~~, B.2.1,~~ and B.2 ~~.2~~

Condition B applies to the Manual Reactor Trip in MODE 1 or 2. ~~This action addresses the train orientation of the SSPS for this Function.~~ With one channel inoperable, the inoperable channel must be restored to OPERABLE status within 48 hours. In this Condition, the remaining OPERABLE channel is adequate to perform the safety function. 46

The Completion Time of 48 hours is reasonable considering that there are two automatic actuation trains and another manual initiation channel OPERABLE, and the low probability of an event occurring during this interval.

If the Manual Reactor Trip Function cannot be restored to OPERABLE status within the allowed 48 hour Completion Time, the unit must be brought to a MODE in which the requirement does not apply. To achieve this status, the unit must be brought to at least MODE 3 within 6 additional hours ~~(54 hours total time) followed by opening the RTBs within 1 additional hour (55 hours total time).~~ The 6 additional hours to reach MODE 3 ~~and the 1 hour to open the RTBs are~~ is reasonable, based on operating experience, to reach MODE 3 ~~and open the RTBs~~ from full power operation in an orderly manner and without challenging unit systems. With ~~the RTBs open and~~ the unit in MODE 3, this trip Function is no longer required to be OPERABLE.

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C.1 and C.2

20 Manual Reactor Trip Function

Condition C applies to the following react or trip Functions in MODE 3, 4, or 5 with the RTBs closed and the CRD System capable of rod withdrawal:

18 Rod Control

D RAI 3.3.1-1

ACTIONS (continued)

20
- ~~Manual Reactor Trip;~~
- ~~RTBs;~~
- ~~RTB Undervoltage and Shunt Trip Mechanisms; and~~
- ~~Automatic Trip Logic.~~

20 4 46

~~This action addresses the train orientation of the SSPS for these Functions.~~ With one channel ~~or train~~ inoperable, the inoperable channel ~~or train~~ must be restored to OPERABLE status within 48 hours. If the Reactor Manual trip channel ~~affected Function(s)~~ cannot be restored to OPERABLE status within the allowed 48 hour Completion Time, the unit must be placed in a MODE in which the requirement does not apply. To achieve this status, the RTBs must be opened within the next hour. The additional hour provides sufficient time to accomplish the action in an orderly manner. With the RTBs open, the Manual Reactor Trip Function is ~~these Functions are~~ no longer required. 20

D RAI 3.3.1-1

28
~~The Completion Time is reasonable considering that in this Condition, the remaining OPERABLE train is adequate to perform the safety function, and given the low probability of an event occurring during this interval.~~

~~D.1.1, D.1.2, D.2.1, D.2.2, and D.3~~

7
~~Condition D applies to the Power Range Neutron Flux - High Function.~~

~~The NIS power range detectors provide input to the CRD System and the SG Water Level Control System and, therefore, have a two-out-of-four trip logic. A known inoperable channel must be placed in the tripped condition. This results in a partial trip condition requiring only one-out-of three logic for actuation. The 6 hours allowed to place the inoperable channel in the tripped condition is justified in WCAP-10271-P-A (Ref. 7).~~

~~In addition to placing the inoperable channel in the tripped condition, THERMAL POWER must be reduced to ≤ 75% RTP within 12 hours. Reducing the power level prevents operation of the core with radial power distributions beyond the design~~

ACTIONS (continued) D 7

E 1 and E 2

Condition E applies to the following reactor trip Functions:

- Power Range Neutron Flux — High; 7
- Power Range Neutron Flux – Low;
- Overtemperature ΔT;
- Overpower ΔT;
- ~~Power Range Neutron Flux – High Positive Rate;~~ 29
- ~~Power Range Neutron Flux – High Negative Rate;~~
- Pressurizer Pressure – High;
- SG Water Level – Low Low; and
- SG Water Level – Low coincident with Steam Flow/Feedwater Flow Mismatch. D RAI 3.3.1-1

1 hour 72 A known inoperable channel must be placed in the tripped condition within 6 hours. Placing the channel in the tripped condition results in a partial trip condition requiring only one-out-of-two logic for actuation of the two-out-of-three trips and one-out-of-three logic for actuation of the two-out-of-four trips. ~~The 6 hours allowed to place the inoperable channel in the tripped condition is justified in Reference 7.~~ D RAI 3.3.1-2

If the operable channel cannot be placed in the trip condition within the specified Completion Time, the unit must be placed in a MODE where these Functions are not required OPERABLE. An additional 6 hours is allowed to place the unit in MODE 3. Six hours is a reasonable time, based on operating experience, to place the unit in MODE 3 from full power in an orderly manner and without challenging unit systems. D RAI 3.3.1-30

71 ~~The Required Actions have been modified by a Note that allows placing the inoperable channel in the bypassed condition for up to 4 hours while performing routine surveillance testing of the other channels. The 4 hour time limit is justified in Reference 7.~~

ACTIONS (continued)

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73

Insert E.1 and E.2

F.1 and F.2

Condition F applies to the Intermediate Range Neutron Flux trip when THERMAL POWER is above the P-6 setpoint and below the P-10 setpoint and one channel is inoperable. Above the P-6 setpoint and below the P-10 setpoint, the NIS intermediate range detector performs the monitoring Functions. If THERMAL POWER is greater than the P-6 setpoint but less than the P-10 setpoint, 24 2 hours is allowed to reduce THERMAL POWER below the P-6 setpoint or increase to THERMAL POWER above the P-10 setpoint. The NIS Intermediate Range Neutron Flux channels must be OPERABLE when the power level is above the capability of the source range, P-6, and below the capability of the power range, P-10. If THERMAL POWER is greater than the P-10 setpoint, the NIS power range detectors perform the monitoring and protection functions and the intermediate range is not required. The Completion Times allow for a slow and controlled power adjustment above P-10 or below P-6 and take into account the redundant capability afforded by the redundant OPERABLE channel, and the low probability of its failure during this period. This action does not require the inoperable channel to be tripped because the Function uses one-out-of-two logic. Tripping one channel would trip the reactor. Thus, the Required Actions specified in this Condition are only applicable when channel failure does not result in reactor trip.

D
RAI 3.3.1-1
Errata #145

G.1 and G.2

Condition G applies to two inoperable Intermediate Range Neutron Flux trip channels in MODE 2 when THERMAL POWER is above the P-6 setpoint and below the P-10 setpoint. Required Actions specified in this Condition are only applicable when channel failures do not result in reactor trip. Above the P-6 setpoint and below the P-10 setpoint, the NIS intermediate range detector performs the monitoring Functions. With no intermediate range channels OPERABLE, the Required Actions are to suspend operations involving positive reactivity additions immediately. This will preclude any power level increase since there are no OPERABLE Intermediate Range Neutron Flux channels. The operator must also reduce THERMAL POWER below the P-6 setpoint within two hours. Below P-6, the Source Range Neutron Flux channels will be able to monitor the core power

E.1 and E.2 Insert

Condition E applies to the Underfrequency Bus A01 and A02 trip function. With one channel inoperable, the inoperable channel must be placed in the tripped condition within 6 hours. Placing the channel in the tripped condition results in a partial trip condition requiring only one additional channel to initiate a reactor trip above the P-7 setpoint. The 6 hours to place the channel in the tripped condition is necessary due to plant design requiring maintenance personnel to effect the trip of the channel outside of the Control Room. An additional 6 hours is allowed to reduce THERMAL POWER to below P-7 if the inoperable channel cannot be restored to OPERABLE status or placed in trip within the specified Completion Time.

Allowance of this time interval takes into consideration the redundant capability provided by the remaining redundant OPERABLE channel and the low probability of occurrence of an event during this period that may require the protection afforded by this trip function.

D

RAI 3.3.1-1
Errata #145

ACTIONS (continued)

level. The Completion Time of 2 hours will allow a slow and controlled power reduction to less than the P -6 setpoint and takes into account the low probability of occurrence of an event during this period that may require the protection afforded by the NIS Intermediate Range Neutron Flux trip.

D
Errata #145

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~~H.1~~

~~Condition H applies to the Intermediate Range Neutron Flux trip when THERMAL POWER is below the P -6 setpoint and one or two channels are inoperable. Below the P -6 setpoint, the NIS source range performs the monitoring and protection functions. The inoperable NIS intermediate range channel(s) must be returned to OPERABLE status prior to increasing power above the P -6 setpoint. The NIS intermediate range channels must be OPERABLE when the power level is above the capability of the source range, P -6, and below the capability of the power range, P -10.~~

H → I.1

Condition H I applies to one inoperable Source Range Neutron Flux trip channel when in MODE 2, below the P -6 setpoint, and performing a reactor startup. With the unit in this Condition, below P -6, the NIS source range performs the monitoring and protection functions. With one of the two channels inoperable, operations involving positive reactivity additions shall be suspended immediately.

D
RAI 3.3.1-1
Errata #145

This will preclude any power escalation. With only one source range channel OPERABLE, core protection is severely reduced and any actions that add positive reactivity to the core must be suspended immediately.

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I → J.1

Condition I J applies to two inoperable Source Range Neutron Flux trip channels when in MODE 2, below the P -6 setpoint, and performing a reactor startup, or in MODE 3, 4, or 5 with the RTBs closed and the CRD System capable of rod withdrawal. With the unit in this Condition, below P-6, the

18 → Rod Control

ACTIONS (continued)

NIS source range performs the monitoring and protection functions. With both source range channels inoperable, the RTBs must be opened immediately. With the RTBs open, the core is in a more stable condition ~~and the unit enters Condition L~~

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J

Errata #145

K.1 and K.2

18 Rod Control

Condition K applies to one inoperable source range channel in MODE 3, 4, or 5 with the RTBs closed and the CRD System capable of rod withdrawal. With the unit in this Condition, below P-6, the NIS source range performs the monitoring and protection functions. With one of the source range channels inoperable, 48 hours is allowed to restore it to an OPERABLE status. If the channel cannot be returned to an OPERABLE status, 1 additional hour is allowed to open the RTBs. Once the RTBs are open, the core is in a more stable condition ~~and the unit enters Condition L~~. ~~The allowance of 48 hours to restore the channel to OPERABLE status, and the additional hour to open the RTBs, are justified in Reference 7.~~

RAI 3.3.1-1

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74

~~L.1, L.2, and L.3~~

~~Condition L applies when the required number of OPERABLE Source Range Neutron Flux channels is not met in MODE 3, 4, or 5 with the RTBs open. With the unit in this Condition, the NIS source range performs the monitoring and protection functions. With less than the required number of source range channels OPERABLE, operations involving positive reactivity additions shall be suspended immediately. This will preclude any power escalation. In addition to suspension of positive reactivity additions, all valves that could add unborated water to the RCS must be closed within 1 hour as specified in LCO 3.9.2. The isolation of unborated water sources will preclude a boron dilution accident.~~

~~Also, the SDM must be verified within 1 hour and once every 12 hours thereafter as per SR 3.1.1.1, SDM verification. With no source range channels OPERABLE, core protection is severely reduced. Verifying the SDM within 1 hour allows~~

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ACTIONS (continued)

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~~sufficient time to perform the calculations and determine that the SDM requirements are met. The SDM must also be verified once per 12 hours thereafter to ensure that the core reactivity has not changed. Required Action L.1 precludes any positive reactivity additions; therefore, core reactivity should not be increasing, and a 12 hour Frequency is adequate. The Completion Times of within 1 hour and once per 12 hours are based on operating experience in performing the Required Actions and the knowledge that unit conditions will change slowly.~~

K.1 and K.2 Approved TSTF-135

M.1 and M.2 K

Condition M applies to the following reactor trip Functions:

- Pressurizer Pressure - Low;
- Pressurizer Water Level - High;
- Reactor Coolant Flow - Low (Two Loops);
- ~~RCP Breaker Position (Two Loops);~~ 14
- Undervoltage RCPs; and Bus A01 & A02; 34
- Underfrequency RCPs.

1 hour 72

With one channel inoperable, the inoperable channel must be placed in the tripped condition within 6 hours. Placing the channel in the tripped condition results in a partial trip condition requiring only one additional channel to initiate a reactor trip above the P-7 setpoint and below the P-8 setpoint. These Functions do not have to be OPERABLE below the P-7 setpoint because there are no loss of flow trips below the P-7 setpoint. ~~The 6 hours allowed to place the channel in the tripped condition is justified in Reference 7.~~ An additional 6 hours is allowed to reduce THERMAL POWER to below P-7 if the inoperable channel cannot be restored to OPERABLE status or placed in trip within the specified Completion Time.

61 interlock
61 interlock
72

Allowance of this time interval takes into consideration the redundant capability provided by the remaining redundant

D RAI 3.3.1-1 Errata #145
D RAI 3.3.1-1
D RAI 3.3.1-4 RAI 3.3.1-2
D RAI 3.3.1-2
D RAI 3.3.1-4

ACTIONS (continued)

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OPERABLE channel, and the low probability of occurrence of an event during this period that may require the protection afforded by the Functions associated with Condition ~~K~~ M. D RAI 3.3.1-1 Errata#145

71 ~~The Required Actions have been modified by a Note that allows placing the inoperable channel in the bypassed condition for up to 4 hours while performing routine surveillance testing of the other channels. The 4 hour time limit is justified in Reference 7.~~ D RAI 3.3.1-30

~~L~~ Approved TSTF-135

N.1 and N.2

Condition N applies to the Reactor Coolant Flow-Low (Single Loop) reactor trip Function. With one channel inoperable, the inoperable channel must be placed in trip within ~~6 hours~~ 1 hour (72). If the channel cannot be restored to OPERABLE status or the channel placed in trip within the ~~6 hours~~ 1 hour (72), then THERMAL POWER must be reduced below the P-8 setpoint within the next 4 hours. This places the unit in a MODE where the LCO is no longer applicable. This trip Function does not have to be OPERABLE below the P-8 setpoint because other RTS trip Functions provide core protection below the P-8 setpoint. 72 ~~The 6 hours allowed to restore the channel to OPERABLE status or place in trip and the 4 additional hours allowed to reduce THERMAL POWER to below the P-8 setpoint are justified in Reference 7.~~ D RAI 3.3.1-5 RAI 3.3.1-2

71 ~~The Required Actions have been modified by a Note that allows placing the inoperable channel in the bypassed condition for up to 4 hours while performing routine surveillance testing of the other channels. The 4 hour time limit is justified in Reference 7.~~ D RAI 3.3.1-30

~~M~~ Approved TSTF-135

O.1 and O.2

Condition O applies to the RCP Breaker Position (Single Loop) reactor trip Function. There is one breaker position device per RCP breaker. With one channel inoperable, the inoperable channel must be restored to OPERABLE status within ~~6 hours~~ 1 hour (13). If the channel cannot be restored to OPERABLE status within the ~~6 hours~~ 1 hour (13), then THERMAL POWER must be reduced below the P-8 setpoint within the next 4 hours.

D RAI 3.3.1-10

ACTIONS (continued)

60 RPS

This places the unit in a MODE where the LCO is no longer applicable. This Function does not have to be OPERABLE below the P-8 setpoint because other RTS Functions provide core protection below the P-8 setpoint. ~~The 6 hours allowed to restore the channel to OPERABLE status and the 4 additional hours allowed to reduce THERMAL POWER to below the P-8 setpoint are justified in Reference 7.~~ ~~The Required Actions have been modified by a Note that allows placing the inoperable channel in the bypassed condition for up to 4 hours while performing routine surveillance testing of the other channels. The 4 hour time limit is justified in Reference 7.~~

D RAI 3.3.1-10

13

72

D RAI 3.3.1-2

13

14 Insert N.1 and N.2

P.1 and P.2

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Autostop 36

D RAI 3.3.1-1 Errata #145

Condition P applies to Turbine Trip on Low Fluid Oil Pressure or on Turbine Stop Valve Closure. With one channel inoperable, the inoperable channel must be placed in the trip condition within 6 hours. If placed in the tripped condition, this results in a partial trip condition requiring only one additional channel to initiate a reactor trip. If the channel cannot be restored to OPERABLE status or placed in the trip condition, then power must be reduced below the P-9 setpoint within the next 4 hours. ~~The 6 hours allowed to place the inoperable channel in the tripped condition and the 4 hours allowed for reducing power are justified in Reference 7.~~

72

D RAI 3.3.1-2

72

D RAI 3.3.1-7 RAI 3.3.1-2

71

~~The Required Actions have been modified by a Note that allows placing the inoperable channel in the bypassed condition for up to 4 hours while performing routine surveillance testing of the other channels. The 4 hour time limit is justified in Reference 7.~~

D RAI 3.3.1-30

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Q.1 and Q.2 P.1 and P.2 Approved TSTF-135

P

60 RPS

Condition Q applies to the SI Input from ESFAS reactor trip and the RTS Automatic Trip Logic in MODES 1 and 2. These actions address the train orientation of the RTS for these Functions. With one train inoperable, 6 hours are allowed to restore the train to OPERABLE status (Required Action Q.1) or the unit must be placed in MODE 3 within the

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D RAI 3.3.1-1 Errata #145

N.1 and N.2 Insert

Condition N applies to the RCP Breaker Position (Two Loop) reactor trip Function. With one channel inoperable, the inoperable channel must be restored to OPERABLE status within 1 hour. If the channel cannot be restored to OPERABLE status in 1 hour, then THERMAL POWER must be reduced below the P-7 interlock within the next 6 hours. This places the unit in a MODE where the LCO is no longer applicable. This function does not have to be OPERABLE below the P-7 interlock because there are no loss of flow trips below the P-7 interlock. The Completion Time of 6 hours is reasonable, based on operating experience, to reduce THERMAL POWER to below the P-7 interlock from full power in an orderly manner without challenging unit systems.

D

RAI 3.3.1-8
RAI 3.3.1-10
RAI 3.3.1-1
Errata #145

ACTIONS (continued)

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next 6 hours. The Completion Time of 6 hours (Required Action Q.1) is reasonable considering that in this Condition, the remaining OPERABLE train is adequate to perform the safety function and given the low probability of an event during this interval. The Completion Time of 6 hours (Required Action Q.2) is reasonable, based on operating experience, to reach MODE 3 from full power in an orderly manner and without challenging unit systems. D RAI 3.3.1-1 Errata #145

The Required Actions have been modified by a Note that allows bypassing one train up to [4] hours for surveillance testing, provided the other train is OPERABLE.

8 16

Approved TSTF-135 Q R.1 and R.2 Q.1 and Q.2 Approved TSTF-135 42 D RAI 3.3.1-1 Errata #145

Condition R applies to the RTBs in MODES 1 and 2. ~~These actions address the train orientation of the RTS for the RTBs.~~ With one train inoperable, 1 hour is allowed to restore the train to OPERABLE status or the unit must be placed in MODE 3 within the next 6 hours. The Completion Time of 6 hours is reasonable, based on operating experience, to reach MODE 3 from full power in an orderly manner and without challenging unit systems. The 1 hour and 6 hour Completion Times are equal to the time allowed by LCO 3.0.3 for shutdown actions in the event of a complete loss of RTS Function. Placing the unit in MODE 3 removes the requirement for this particular Function.

42 RTB RTB 42 60 RPS RTB 42 a Note

17 allowing 8 17 RTB 42 17

The Required Actions have been modified by two Notes. Note 1 allows one channel to be bypassed for up to 2 hours ~~for surveillance testing~~, provided the other channel is OPERABLE. ~~Note 2 allows one RTB to be bypassed for up to 2 hours for maintenance on undervoltage or shunt trip mechanisms if the other RTB train is OPERABLE. The 2 hour time limit is justified in Reference 7.~~

Approved TSTF-135 S.1 and S.2 R.1 and R.2 R interlock (in MODE 2) and the 56 D RAI 3.3.1-1 Errata #145 one or more channels Approved TSTF-135

Condition S applies to the P-6 and P-10 interlock~~s~~. With one channel inoperable for one-out-of-two or two-out-of-four coincidence logic, the associated interlock must be verified to be in its required state for the existing unit condition

ACTIONS (continued)

within 1 hour or the unit must be placed in MODE 3 within the next 6 hours. Verifying the interlock status manually accomplishes the interlock's Function. The Completion Time of 1 hour is based on operating experience and the minimum amount of time allowed for manual operator actions. The Completion Time of 6 hours is reasonable, based on operating experience, to reach MODE 3 from full power in an orderly manner and without challenging unit systems. The 1 hour and 6 hour Completion Times are equal to the time allowed by LCO 3.0.3 for shutdown actions in the event of a complete loss of RTS Function.

60 RPS

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T.1 and T.2 S.1 and S.2 and P-9

S , two-out-of-two

Condition T applies to the P-7, P-8, P-9, and P-13 interlocks. With one channel inoperable for one-out-of-two or two-out-of-four coincidence logic, the associated interlock must be verified to be in its required state for the existing unit condition within 1 hour or the unit must be placed in MODE 2 within the next 6 hours. These actions are conservative for the case where power level is being raised. Verifying the interlock status manually accomplishes the interlock's Function. The Completion Time of 1 hour is based on operating experience and the minimum amount of time allowed for manual operator actions. The Completion Time of 6 hours is reasonable, based on operating experience, to reach MODE 2 from full power in an orderly manner and without challenging unit systems.

one or more channels

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D RAI 3.3.1-1 Errata #145

20 Insert T.1 and T.2

U.1, U.2.1, and U.2.2 U.1 and U.2 Approved TSTF-135

D RAI 3.3.1-1 Errata #145

U

Condition U applies to the RTB Undervoltage and Shunt Trip Mechanisms, or diverse trip features, in MODES 1 and 2. With one of the diverse trip features inoperable, it must be restored to an OPERABLE status within 48 hours or the unit must be placed in a MODE where the requirement does not apply. This is accomplished by placing the unit in MODE 3 within the next 6 hours (54 hours total time) ~~followed by opening the RTBs in 1 additional hour (55 hours total time).~~ The Completion Time of 6 hours is a reasonable time, based on operating experience, to reach MODE 3 from full power in an orderly manner and without challenging unit systems.

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RAI 3.3.1-1
Errata #145

T.1 and T.2 Insert

RAI 3.3.1-1
Errata #145

Condition T applies to the RTBs and the RTB Undervoltage and Shunt Trip Mechanisms in MODES 3, 4, or 5 with the RTBs closed and the Rod Control System capable of rod withdrawal.

With one trip mechanism or RTB inoperable, the inoperable trip mechanism or RTB must be restored to OPERABLE status within 48 hours. The Completion Time is reasonable considering that the remaining OPERABLE trip mechanism or RTB is adequate to perform the safety function, and given the low probability of an event occurring during this interval.

If the RTB or trip mechanism cannot be restored to OPERABLE status within 48 hours, the unit must be placed in a MODE in which the requirement does not apply. This is accomplished by opening the RTBs within the next hour (49 hours total time). The Completion Time of 1 hour provides sufficient time to accomplish this action in an orderly manner and takes into account the low probability of an event occurring in this interval.

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Condition T would apply to any inoperable RTB trip mechanism.

D Errata #145

ACTIONS (continued)

With ~~the RTBs open and~~ the unit in MODE 3, this trip Function is no longer required to be OPERABLE. The affected RTB shall not be bypassed while one of the diverse features is inoperable except for the time required to perform maintenance to one of the diverse features. The allowable time for performing maintenance of the diverse features is 2 hours for the reasons stated under Condition R. Q

D RAI 3.3.1-1 Errata #145

The Completion Time of 48 hours ~~for Required Action U.1~~ is reasonable considering that in this Condition there is one remaining diverse feature for the affected RTB, and one OPERABLE RTB capable of performing the safety function and given the low probability of an event occurring during this interval.

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~~V.1~~

Insert V.1, V.2, W.1 and W.2

Insert X.1 and X.2

D RAI 3.3.1-1 Errata #145

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~~With two RTS trains inoperable, no automatic capability is available to shut down the reactor, and immediate plant shutdown in accordance with LCO 3.0.3 is required.~~

SURVEILLANCE REQUIREMENTS

The SRs for each RPS RTS Function are identified by the SRs column of Table 3.3.1-1 for that Function.

A Note has been added to the SR Table stating that Table 3.3.1-1 determines which SRs apply to which RTS RPS Functions.

Note that each channel of process protection supplies both trains of the RTS RPS. When testing Channel I, Train A and Train B must be examined. Similarly, Train A and Train B must be examined when testing Channel II, Channel III, and Channel IV (if applicable). The CHANNEL CALIBRATION and COTs are performed in a manner that is consistent with the assumptions used in analytically calculating the required channel accuracies.

~~Reviewer's Note: Certain Frequencies are based on approval topical reports. In order for a licensee to use these times, the licensee must justify the Frequencies as required by the staff SER for the topical report.~~

V.1 and V.2 Insert

Condition V applies to the Reactor Trip Bypass Breaker (RTBB) and associated Undervoltage Trip Mechanism in MODE 1 or 2, when the RTBB is racked in and closed. With the required RTBB inoperable, 1 hour is allowed to restore the RTBB to OPERABLE status or the unit must be placed in MODE 3 within the next 6 hours. The Completion Time of 6 hours is reasonable, based on operating experience, to reach MODE 3 from full power in an orderly manner and without challenging unit systems. The 1 hour and 6 hour completion times are equal to the time allowed by LCO 3.0.3 for shutdown action in the event of a complete loss of RTS Function. Placing the unit in MODE 3 removes the requirement for this particular Function.

D
RAI 3.3.1-1
Errata #145

W.1 and W.2 Insert

Condition W applies to the Reactor Trip Bypass Breaker (RTBB) and associated Undervoltage Trip Mechanism in MODES 3, 4, or 5, when an RTBB is racked in and closed and the Rod Control System is capable of rod withdrawal. With the required RTBB inoperable, 48 hours is allowed to restore the RTBB to OPERABLE status or the unit must be placed in a MODE in which the requirement does not apply. To achieve this status, the RTBs and RTBBs must be opened within the next 1 hour (49 hours total time). The Completion Time of 1 hour provides sufficient time to accomplish the action in an orderly manner. With the RTBs and RTBBs open, this Function is no longer required.

D
RAI 3.3.1-1
Errata #145

X.1 and X.2 Insert

Condition X applies to the RPS Automatic Trip Logic in MODES 3, 4 or 5 with the RTBs closed and the Rod Control System capable of rod withdrawal. With one train inoperable, 48 hours are allowed to restore the train to an OPERBALE status. The Completion Time of 48 hours is reasonable considering that in this condition, the remaining OPERABLE train is adequate to perform the safety function, and given the low probability of an event occurring in this interval.

D
RAI 3.3.1-1
Errata #145

If the RPS Automatic Trip Logic cannot be restored to OPERABLE status within 48 hours, the unit must be placed in a MODE where this Function is not required to be OPERABLE. To achieve this status, the RTBs must be opened within the next 1 hour (49 hours total time). The additional hour provides sufficient time to accomplish the action in an orderly manner. With the RTBs open, the Automatic Trip Logic is no longer required.

SURVEILLANCE REQUIREMENTS (continued)

allowed for performing the first Surveillance after reaching 15% RTP. At lower power levels, calorimetric data are inaccurate.

The Frequency of every 24 hours is adequate. It is based on unit operating experience, considering instrument reliability and operating history data for instrument drift. Together these factors demonstrate the change in the absolute difference between NIS and heat balance calculated powers rarely exceeds 2% in any 24 hour period.

In addition, control room operators periodically monitor redundant indications and alarms to detect deviations in channel outputs.

69

SR 3.3.1.3 is performed by means of the moveable incore detector system.

SR 3.3.1.3

SR 3.3.1.3 compares the incore system to the NIS channel output every 31 EFPD. If the absolute difference is ≥ 3%, the NIS channel is still OPERABLE, but must be readjusted.

D

RAI 3.3.1-18

If the NIS channel cannot be properly readjusted, the channel is declared inoperable. This Surveillance is performed to verify the f(ΔI) input to the overtemperature ΔT Function.

Two Notes modify SR 3.3.1.3. Note 1 indicates that the excore NIS channel shall be adjusted if the absolute difference between the incore and excore AFD is ≥ 3%. Note 2 clarifies that the Surveillance is required only if reactor power is ≥ [15%] RTP and that 24 hours is allowed for performing the first Surveillance after reaching [15%] RTP.

16 → 50%

The Frequency of every 31 EFPD is adequate. It is based on unit operating experience, considering instrument reliability and operating history data for instrument drift. Also, the slow changes in neutron flux during the fuel cycle can be detected during this interval.

Insert SR 3.3.1.5 Notes

SR 3.3.1.5 is modified by two Notes. Note 1 provides an 8 hour delay in the requirement to perform this Surveillance for the Source Range Neutron Flux trip function instrumentation when power is reduced to below P-6. This Note allows a normal shutdown to proceed without a delay for testing in MODE 2 and for a short time in MODE 3 until the RTBs are open and SR 3.3.1.5 is no longer required to be performed. If the unit is to be in MODE 2 with power below P-6 for > 8 hours, this Surveillance must be performed prior to 8 hours after reducing power below P-6.

Note 3 excludes the RCP Breaker Position (Two Loop) Reactor Coolant Flow - Low (Two Loop) and Underfrequency Bus A01 and A02 Trip Functions, and the P-6, P-7, P-8, P-9 and P-10 Interlocks. These functions/interlocks are tested at an 18 month frequency via SR 3.3.1.15

D

RAI TR-2
RAI 3.3.1-22

SURVEILLANCE REQUIREMENTS (continued)

A Note modifies SR 3.3.1.6. The Note states that this Surveillance is required only if reactor power is > 50% RTP and that [24] hours is allowed for performing the first surveillance after reaching 50% RTP.

16 24

The Frequency of 92 EFPD is adequate. It is based on industry operating experience, considering instrument reliability and operating history data for instrument drift.

SR 3.3.1.7

16 92

SR 3.3.1.7 is the performance of a COT every [92] days.

A COT is performed on each required channel to ensure the entire channel will perform the intended Function.

Setpoints must be within the Allowable Values specified in Table 3.3.1-1.

The difference between the current "as found" values and the previous test "as left" values must be consistent with the drift allowance used in the setpoint methodology. The setpoint shall be left set consistent with the assumptions of the current unit specific setpoint methodology.

67

and verified to be within the required limits.

The "as found" and "as left" values must also be recorded and reviewed for consistency with the assumptions of Reference 7.

SR 3.3.1.7 is modified by a Note that provides a 4 hour delay in the requirement to perform this Surveillance for source range instrumentation when entering MODE 3 from MODE 2. This Note allows a normal shutdown to proceed without a delay for testing in MODE 2 and for a short time in MODE 3 until the RTBs are open and SR 3.3.1.7 is no longer required to be performed. If the unit is to be in MODE 3 with the RTBs closed for > 4 hours this Surveillance must be performed prior to 4 hours after entry into MODE 3.

75 ~~The Frequency of [92] days is justified in Reference 7.~~

D
Errata #155

SURVEILLANCE REQUIREMENTS (continued)

SR 3.3.1.8

SR 3.3.1.8 is the performance of a COT as described in SR 3.3.1.7, except it is modified by a Note that this test shall include verification that the P-6 and P-10 interlocks are in their required state for the existing unit condition. The Frequency is modified by a Note that allows this surveillance to be satisfied if it has been performed within 16 92 [92] days of the Frequencies prior to reactor startup and four hours after reducing power below P-10 and P-6. The Frequency of "prior to startup" ensures this surveillance is performed prior to critical operations and applies to the source, intermediate and power range low instrument channels. The Frequency of "4 hours after reducing power below P-10" (applicable to intermediate and power range low channels) and "4 hours after reducing power below P-6" (applicable to source range channels) allows a normal shutdown to be completed and the unit removed from the MODE of Applicability for this surveillance without a delay to perform the testing required by this surveillance. The Frequency of every 92 days thereafter applies if the plant remains in the MODE of Applicability after the initial performances of prior to reactor startup and four hours after reducing power below P-10 or P-6. The MODE of Applicability for this surveillance is < P-10 for the power range low and intermediate range channels and < P-6 for the source range channels. Once the unit is in MODE 3, this surveillance is no longer required. If power is to be maintained < P-10 or < P-6 for more than 4 hours, then the testing required by this surveillance must be performed prior to the expiration of the 4 hour limit. Four hours is a reasonable time to complete the required testing or place the unit in a MODE where this surveillance is no longer required. This test ensures that the NIS source, intermediate, and power range low channels are OPERABLE prior to taking the reactor critical and after reducing power into the applicable MODE (< P-10 or < P-6) for periods > 4 hours. D RAI 3.3.1-17

SR 3.3.1.9

SR 3.3.1.9 is the performance of a TADOT and is performed every [92] days~~, as justified in Reference 7~~. 75 D Errata #155

31 16

SURVEILLANCE REQUIREMENTS (continued)

SR 3.3.1.14 13 55

SR 3.3.1.14 is the performance of a TADOT of the Manual Reactor Trip, RCP Breaker Position, ~~and the~~ SI Input from ESFAS, and the Condenser Pressure-High and Circulating Water Pump Breaker Position inputs to the P-9 Interlock. This TADOT is performed every [18] months. The test shall independently verify the OPERABILITY of the undervoltage and shunt trip mechanisms circuits for the Manual Reactor Trip Function for the Reactor Trip Breakers and Reactor Trip Bypass Breakers. ~~The Reactor Trip Bypass Breaker test shall include testing of the automatic undervoltage trip.~~

40 18 16 40 53 and/or 68 Errata #101

The Frequency is based on the known reliability of the Functions and the multichannel redundancy available, and has been shown to be acceptable through operating experience.

24 ~~The SR is modified by a Note that excludes verification of setpoints from the TADOT. The Functions affected have no setpoints associated with them.~~

16 P-9

SR 3.3.1.15 14 55

SR 3.3.1.15 is the performance of a TADOT of Turbine Trip Functions. This TADOT is as described in SR 3.3.1.4, except that this test is performed prior to exceeding the [P-9] interlock when the unit has been in MODE 3. ~~reactor startup. A Note states that~~ this Surveillance is not required if it has been performed within the previous 31 days. ~~Verification of the Trip Setpoint does not have to be performed for this Surveillance.~~ Performance of this test will ensure that the turbine trip Function is OPERABLE prior to exceeding the [P-9] interlock ~~taking the reactor critical. This test cannot be performed with the reactor at power and must therefore be performed prior to reactor startup.~~

Approved TSTF-135 24 P-9 16 RAI 3.3.1-7 RAI 3.3.1-7

25 ~~SR 3.3.1.16~~

~~SR 3.3.1.16 verifies that the individual channel/train actuation response times are less than or equal to the maximum values assumed in the accident analysis. Response time testing acceptance criteria are included in Technical Requirements Manual, Section 15 (Ref. 8). Individual component response times are not modeled in the analyses.~~

Insert SR 3.3.1.15 33 RAI 3.3.1-26

RAI 3.3.1-26

Insert SR 3.3.1.15

SR 3.3.1.15 is the performance of an ACTUATION LOGIC TEST on the RCP Breaker Position (Two Loop), Reactor Coolant Flow - Low (Two Loop) and Underfrequency Bus A01 and A02 Trip Functions, and P-6, P-7, P-8, P-9 and P-10 Interlocks, every 18 months. 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 an unplanned transient if the surveillance were performed with the reactor at power.

RAI 3.3.1-22

REFERENCES (continued)

16 4 5. 10 CFR 50.49.

16 5 6. RTS/ESFAS Setpoint Methodology Study. DG-I01, Instrument Setpoint Methodology 16

71, 72, 73, 74 ~~7. WCAP-10271-P-A, Supplement 2, Rev. 1, June 1990.~~

~~8. Technical Requirements Manual, Section 15, "Response Times."~~

D

RAI 3.3.1-2
RAI 3.3.1-25
RAI 3.3.1-30

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.
L.01 Rev. D	Not used.
L.02 Rev. D	Not used.

NSHC Number	NSHC Text
L.03 Rev. D	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 addition of a Note that allows one RPS train to be bypassed for up to 8 hours for surveillance testing, provided the other train is OPERABLE. This Note is consistent with the allowance to bypass a RTB for 8 hours, if the other RTB is OPERABLE. The RPS system is assumed to function in the mitigation of various design basis events, but is not assumed to be an initiator of any analyzed event. Based on the change not presenting any new equipment or operational modes, the probability of previously evaluated accidents is not significantly altered. 2. Does the change create the possibility of a new or different kind of accident from any accident previously evaluated? The proposed change to RPS allowed out-of-service time does not result in a change in the manner in which the RPS provides plant protection. No change is being made which alters the functioning of the RPS (other than in a test mode). The proposed changes to RPS out-of-service times and surveillance intervals do not involve hardware changes. Therefore, the proposed change does not create the possibility of a new or different kind of accident. 3. Does this change involve a significant reduction in a margin of safety? The proposed changes to RPS out-of-service time does not alter the manner in which safety limits, limiting safety system setpoints or limiting conditions for operation are determined. Implementation of the proposed change is expected to result in an overall improvement in safety due to fewer inadvertent reactor trips, and higher quality repairs leading to improved equipment reliability resulting from longer repair times. Therefore, this change does not involve a significant reduction in a margin of safety.

NSHC Number	NSHC Text
L.04 Rev. D	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 deletes CTS Table 15.3.5-2, Note ***. This Note requires the unit to be in cold shutdown within 48 hours, if the minimum conditions for SI input to RPS are not met within 24 hours after reaching hot shutdown. Although certain elements of RPS and ESFAS SI function are required to be operable in Modes 3, 4 and 5, the SI actuation input to RPS is only credited for shutting down the core to reduce the heat generation to decay levels required by 10CFR50, Appendix K. 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 RPS are properly maintained. Therefore, this change does not involve a reduction in a margin of safety.

NSHC Number	NSHC Text
L.05 Rev. D	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 an increase in the time allowed to commence a shutdown if the minimum operable channels of SI input to RPS cannot be met. RPS and SI are assumed to function in the mitigation of various design basis events, but are not assumed to be an initiator of any analyzed event. Based on the change not presenting any new equipment or operational modes, the probability of previously evaluated accidents is not significantly altered. 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. The proposed change to the time allowed to commence a shutdown if the minimum operable channels of SI input to RPS does not result in a change in the manner in which the RPS provides plant protection. No change is being made which alters the functioning of the RPS. Therefore, the proposed changes do not create the possibility of a new or different kind of accident. 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 RPS are properly maintained. Therefore, this change does not involve a reduction in a margin of safety.

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 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. This change modifies CTS Table 15.4.1-1, item #1, Daily Heat Balance surveillance requirement for the Nuclear Power Range instrumentation. The Note allows 12 hours after THERMAL POWER is greater than or equal to 15% RTP to perform this surveillance. This change is acceptable, because at lower power levels calorimetric data is inaccurate. Therefore the change provides an allowance to delay performance of the SR until conditions necessary to perform the SR are established while ensuring the SR is performed at the earliest reasonable opportunity. 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 RPS are properly maintained. Therefore, this change does not involve a reduction in 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 revises the frequency of the monthly comparison of incore detector measurements to NIS axial flux difference from monthly to 31 EFPD. This change is acceptable, based on unit operating experience, considering instrument reliability and operating history data for instrument drift. Additionally, the slow changes in neutron flux during the fuel cycle can be detected 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 RPS are properly maintained. Therefore, this change does not involve a reduction in a margin of safety.
L.08 Rev. D	Not used.

NSHC Number	NSHC Text
L.09 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. This change revises the CTS Table 15.4.1-1, surveillance requirement Frequency of P, "Prior to reactor criticality, if not performed during the previous week," to the proposed ITS SR 3.3.1.8, Frequency of, "Prior to reactor startup," which is modified by a Note stating, "Only required when not performed within previous 92 days." This is acceptable because the 92 days is consistent with the frequency of performance for this type of surveillance on similar instrumentation and will not result in an impact to the safety of the unit. 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 RPS are properly maintained. Therefore, this change does not involve a reduction in a margin of safety.

NSHC Number	NSHC Text
L.10 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. This change revises the CTS Table 15.4.1-1, item #4, Reactor Coolant Temperature (OP deltaT and OT deltaT), Plant Conditions When Required, from "PWR, HOT S/D, COLD S/D", to "MODES 1, 2", in ITS Table 3.3.1-1, functions 5 and 6. This is acceptable because in MODES 3, 4, 5 and 6 the OT deltaT function is not required to be OPERABLE due to insufficient heat production to be concerned about DNB. The OP deltaT function is not required to be OPERABLE in MODES 3, 4, 5 and 6, because the reactor is not operating and there is insufficient heat production to be concerned about fuel overheating and fuel damage. 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 RPS are properly maintained. Therefore, this change does not involve a reduction in a margin of safety.

NSHC Number	NSHC Text
L.11 Rev. D	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. The proposed change modifies the Actuation Logic Test requirement by the addition of 2 Notes. Note 1 allows a delay in the performance of this surveillance for the Source Range Neutron Flux Trip Function for up to 8 hours after power is reduced below P-6. Note 2 allows an exception to the performance of this surveillance for the RCP Breaker Position (Two Loops), Reactor Coolant Flow – Low (Two Loops) and Underfrequency Bus A01 and A02 Trip Functions and the P-6, P-7, P-8, P-9 and P-10 Interlocks. 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 RPS system is assumed to function in the mitigation of various design basis events, but is not assumed to be an initiator of any analyzed event. Based on the change not presenting any new equipment or operational modes, the probability of previously evaluated accidents is not significantly altered. 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 RPS are properly maintained. Therefore, this change does not involve a reduction in a margin of safety.
L.12 Rev. D	Not used.

NSHC Number	NSHC Text
L.13 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 RPS are properly maintained. Therefore, this change does not involve a reduction in a margin of safety.

NSHC Number	NSHC Text
L.14 Rev. D	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 modifying when the Undervoltage Bus A01 and A02 and Underfrequency Bus A01 and A02 trip functions are required to be operable. These reactor trip functions provide protection against core DNB during loss of coolant flow events that result from a loss of RCPs. Per PBNP design, these trips are automatically blocked below P-7. Below the P-7 interlock, natural circulation in the RCS can provide adequate core cooling, such that a reactor trip is not required. Furthermore, although these trip functions are assumed to function in the mitigation of a design basis event, they are not assumed to be an initiator of any analyzed event. Based on the change not presenting any new equipment or operational modes, the probability of previously evaluated accidents is not significantly altered. 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. The proposed change to the mode of applicability for the Undervoltage Bus A01 and A02 and Underfrequency Bus A01 and A02 trip functions does not result in a change in the manner in which the RPS provides plant protection. No change is being made which alters the functioning of the RPS. Therefore, the proposed change does not create the possibility of a new or different kind of accident. 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 RPS are properly maintained. Therefore, this change does not involve a reduction in a margin of safety.

NSHC Number	NSHC Text
L.15 Rev. D	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 a relaxation of the testing requirement for the Turbine Trip Functions. Although the Turbine Trip Functions may actuate to mitigate a design basis event, they are not assumed to be initiators of any analyzed event. Based on the change not presenting any new equipment or operational modes, the probability of previously evaluated accidents is not significantly altered. 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. No change is being made which alters the functioning of the RPS. Therefore, the proposed change does not create the possibility of a new or different kind of accident. 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 RPS are properly maintained. Therefore, this change does not involve a reduction in a margin of safety.

NSHC Number	NSHC Text
L.16 Rev. D	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 a relaxation of the testing requirements for the RTBs. Although the RPS system is assumed to function in the mitigation of various design basis events, it is not assumed to be an initiator of any analyzed event. Based on the change not presenting any new equipment or operational modes, the probability of previously evaluated accidents is not significantly altered. 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. No change is being made which alters the functioning of the RPS. Therefore, the proposed change does not create the possibility of a new or different kind of accident. 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 RPS are properly maintained. Therefore, this change does not involve a reduction in a margin of safety.

NSHC Number	NSHC Text
L.17 Rev. D	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 a relaxation of the logic testing frequency for the Turbine Trip and SI input from ESFAS functions. Although the RPS system is assumed to function in the mitigation of various design basis events, it is not assumed to be an initiator of any analyzed event. Based on the change not presenting any new equipment or operational modes, the probability of previously evaluated accidents is not significantly altered. 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. No change is being made which alters the functioning of the RPS. Therefore, the proposed change does not create the possibility of a new or different kind of accident. 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 RPS are properly maintained. Therefore, this change does not involve a reduction in a margin of safety.

NSHC Number	NSHC Text
L.18 Rev. D	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 the setpoint for the low voltage trip of a RCP motor breaker. The low voltage trip of an RCP Breaker does not cause a direct reactor trip, although an undervoltage condition on the RCP Buses (A01 and A02) does provide a reactor trip using a different set of undervoltage relays than the RCP Breaker trip. Furthermore, Point Beach accident analysis does not credit the indirect trip of the RCP Breakers on low voltage for the mitigation of a loss of flow event. Accordingly, this setpoint may be deleted from the technical specifications as it is not required to provide adequate protection of public health and safety. 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. No change is being made which alters the functioning of the RPS. Therefore, the proposed change does not create the possibility of a new or different kind of accident. 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 RPS are properly maintained. Therefore, this change does not involve a reduction in a margin of safety.

NSHC Number	NSHC Text
L.19 Rev. D	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.

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.

3.3 INSTRUMENTATION

3.3.1 Reactor Protection System (RPS) Instrumentation

LCO 3.3.1 The RPS instrumentation for each Function in Table 3.3.1-1 shall be OPERABLE.

APPLICABILITY: According to Table 3.3.1-1.

ACTIONS

----------NOTE----------
Separate Condition entry is allowed for each Function.

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. One or more Functions with one or more required channels or trains inoperable.	A.1	Enter the Condition referenced in Table 3.3.1-1 for the channel(s) or train(s).	Immediately
B. One Manual Reactor Trip channel inoperable.	B.1	Restore channel to OPERABLE status.	48 hours
	<u>OR</u>		
	B.2	Be in MODE 3.	54 hours
C. One Manual Reactor Trip channel inoperable.	C.1	Restore channel to OPERABLE status.	48 hours
	<u>OR</u>		
	C.2	Open reactor trip breakers (RTBs).	49 hours

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(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION		COMPLETION TIME
D. One channel inoperable.	D.1	Place channel in trip.	1 hour
	OR		
	D.2	Be in MODE 3.	7 hours
E. One channel inoperable.	E.1	Place channel in trip.	6 hours
	OR		
	E.2	Reduce THERMAL POWER to < P-7.	12 hours
F. One Intermediate Range Neutron Flux channel inoperable.	F.1	Reduce THERMAL POWER to < P-6.	24 hours
	OR		
	F.2	Increase THERMAL POWER to > P-10.	24 hours
G. Two Intermediate Range Neutron Flux channels inoperable.	G.1	Suspend operations involving positive reactivity additions.	Immediately
	AND		
	G.2	Reduce THERMAL POWER to < P-6.	2 hours
H. One Source Range Neutron Flux channel inoperable.	H.1	Suspend operations involving positive reactivity additions.	Immediately

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RAI 3.3.1-2
RAI 3.3.1-30

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RAI 3.3.1-2
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(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION		COMPLETION TIME
I. Two Source Range Neutron Flux channels inoperable.	I.1	Open RTBs.	Immediately
J. One Source Range Neutron Flux channel inoperable.	J.1	Restore channel to OPERABLE status.	48 hours
	OR		
	J.2	Open RTBs.	49 hours
K. One channel inoperable.	K.1	Place channel in trip.	1 hour
	OR		
	K.2	Reduce THERMAL POWER to < P-7.	7 hours
L. One Reactor Coolant Flow-Low (Single Loop) channel inoperable.	L.1	Place channel in trip.	1 hour
	OR		
	L.2	Reduce THERMAL POWER to < P-8.	5 hours
M. One Reactor Coolant Pump Breaker Position (Single Loop) channel inoperable.	M.1	Restore channel to OPERABLE status.	1 hour
	OR		
	M.2	Reduce THERMAL POWER to < P-8.	5 hours

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RAI 3.3.1-2
RAI 3.3.1-4
RAI 3.3.1-5
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(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
N. One inoperable channel.	N.1 Restore channel to OPERABLE status. OR N.2 Reduce THERMAL POWER to < P-7.	1 hour 7 hours
O. One turbine trip channel inoperable.	O.1 Place channel in trip. OR O.2 Reduce THERMAL POWER to < P-9.	1 hour 5 hours
P. One train inoperable.	-----------------NOTE----------------- One train may be bypassed for up to 8 hours for surveillance testing provided the other train is OPERABLE. -- P.1 Restore train to OPERABLE status. OR P.2 Be in MODE 3.	 6 hours 12 hours

(continued)

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RAI 3.3.1-1
RAI 3.3.1-2
RAI 3.3.1-7
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ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
Q. One RTB inoperable.	-----------------NOTE----------------- One RTB may be bypassed for up to 8 hours provided the other RTB is OPERABLE. ---	
	Q.1 Restore RTB to OPERABLE status.	1 hour
	OR	
	Q.2 Be in MODE 3.	7 hours
R. One or more channel(s) inoperable.	R.1 Verify interlock is in required state for existing unit conditions.	1 hour
	OR	
	R.2 Be in MODE 3.	7 hours
S. One or more channel(s) inoperable.	S.1 Verify interlock is in required state for existing unit conditions.	1 hour
	OR	
	S.2 Be in MODE 2.	7 hours
T. One RTB or trip mechanism for one RTB inoperable.	T.1 Restore RTB or RTB trip mechanism to OPERABLE status.	48 hours
	OR	
	T.2 Open RTBs.	49 hours

D
RAI 3.3.1-1
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(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION		COMPLETION TIME
U. One trip mechanism inoperable for one RTB.	U.1	Restore inoperable trip mechanism to OPERABLE status.	48 hours
	OR		
	U.2	Be in MODE 3.	54 hours
V. One reactor trip bypass breaker (RTBB) or trip mechanism for one RTBB inoperable.	V.1	Restore RTBB or RTBB trip mechanism to OPERABLE status.	1 hour
	OR		
	V.2	Be in MODE 3.	7 hours
W. One reactor trip bypass breaker (RTBB) or trip mechanism for one RTBB inoperable.	W.1	Restore RTBB or RTBB trip mechanism to OPERABLE status.	48 hours
	OR		
	W.2	Open RTBs and RTBBs.	49 hours
X. One train inoperable.	X.1	Restore train to OPERABLE status.	48 hours
	OR		
	X.2	Open RTBs.	49 hours

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SURVEILLANCE REQUIREMENTS

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

SURVEILLANCE		FREQUENCY
SR 3.3.1.1	Perform CHANNEL CHECK.	12 hours
SR 3.3.1.2	-----NOTES----- 1. Adjust NIS channel if absolute difference is > 2%. 2. Not required to be performed until 12 hours after THERMAL POWER is ≥ 15% RTP. ---------- Compare results of calorimetric heat balance calculation to Nuclear Instrumentation System (NIS) channel output.	24 hours
SR 3.3.1.3	-----NOTES----- 1. Adjust NIS channel if absolute difference is ≥ 3%. 2. Not required to be performed until 24 hours after THERMAL POWER is ≥ 50% RTP. ---------- Compare results of the incore detector measurements to NIS AFD.	31 effective full power days (EFPD)

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.3.1.4	-------------------------NOTE------------------------- This Surveillance must be performed on the reactor trip bypass breaker prior to placing the bypass breaker in service. -- Perform TADOT.	 31 days on a STAGGERED TEST BASIS
SR 3.3.1.5	-------------------------NOTES------------------------- 1. Not required to be performed for the Source Range Neutron Flux Trip Function until 8 hours after power is below P-6. 2. Not required to be performed for the RCP Breaker Position (Two Loops), Reactor Coolant Flow — Low (Two Loops) and Underfrequency Bus A01 and A02 Trip Functions and the P-6, P-7, P-8, P-9 and P-10 Interlocks. -- Perform ACTUATION LOGIC TEST.	 31 days on a STAGGERED TEST BASIS
SR 3.3.1.6	-------------------------NOTE------------------------- Not required to be performed until 24 hours after THERMAL POWER is ≥ 50% RTP. -- Calibrate excore channels to agree with incore detector measurements.	 92 EFPD

(continued)

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RAI 3.3.1-22
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SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.3.1.7	-----NOTE----- Not required to be performed for source range instrumentation prior to entering MODE 3 from MODE 2 until 4 hours after entry into MODE 3. ----- Perform COT.	92 days

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.3.1.8	-----NOTE----- This Surveillance shall include verification that interlocks P-6 and P-10 are in their required state for existing unit conditions. ----- Perform COT.	-----NOTE----- Only required when not performed within previous 92 days ----- Prior to reactor startup AND Four hours after reducing power below P-10 for power and intermediate range instrumentation AND Four hours after reducing power below P-6 for source range instrumentation AND Every 92 days thereafter

D RAI 3.3.1-17

D RAI 3.3.1-17

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.3.1.9	Perform TADOT.	31 days
SR 3.3.1.10	-----NOTE----- This Surveillance shall include verification that the time delays are adjusted to the prescribed values. ----- Perform CHANNEL CALIBRATION.	18 months
SR 3.3.1.11	-----NOTE----- Neutron detectors are excluded from CHANNEL CALIBRATION. ----- Perform CHANNEL CALIBRATION.	18 months
SR 3.3.1.12	Perform COT.	18 months
SR 3.3.1.13	Perform TADOT.	18 months
SR 3.3.1.14	Perform TADOT.	Prior to exceeding the P-9 interlock whenever the unit has been in MODE 3, if not performed within previous 31 days.

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RAI 3.3.1-7

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.3.1.15	--------------------------NOTE-------------------------- This Surveillance must be performed on the RCP Breaker Position (Two Loop), Reactor Coolant Flow - Low (Two Loop) and Underfrequency Bus A01 and A02 Trip Functions and the P-6 , P-7, P-8, P-9 and P-10 Interlocks. --- Perform ACTUATION LOGIC TEST.	18 months

D

RAI 3.3.1-22
RAI 3.3.1-26

Table 3.3.1-1 (page 1 of 8)
Reactor Protection System Instrumentation

FUNCTION	APPLICABLE MODES	REQUIRED CHANNELS	CONDITIONS	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE
1. Manual Reactor Trip	1,2	2	B	SR 3.3.1.13	NA
	3(a), 4(a), 5(a)	2	C	SR 3.3.1.13	NA
2. Power Range Neutron Flux					
a. High	1,2	4	D	SR 3.3.1.1 SR 3.3.1.2 SR 3.3.1.7 SR 3.3.1.11	≤ 108% RTP
b. Low	1(b),2	4	D	SR 3.3.1.1 SR 3.3.1.8 SR 3.3.1.11	≤ 25% RTP
3. Intermediate Range Neutron Flux	1(b), 2(c)	2	F,G	SR 3.3.1.1 SR 3.3.1.8 SR 3.3.1.11	≤ 40% RTP
4. Source Range Neutron Flux	2(d)	2	H,I	SR 3.3.1.1 SR 3.3.1.8 SR 3.3.1.11	within span of instrumentation
	3(a), 4(a), 5(a)	2	I,J	SR 3.3.1.1 SR 3.3.1.7 SR 3.3.1.11	within span of instrumentation
5. Overtemperature ΔT	1,2	4	D	SR 3.3.1.1 SR 3.3.1.3 SR 3.3.1.6 SR 3.3.1.7 SR 3.3.1.11	Refer to Note 1 (Page 3.3.1-18)
6. Overpower ΔT	1,2	4	D	SR 3.3.1.1 SR 3.3.1.7 SR 3.3.1.11	Refer to Note 2 (Page 3.3.1-20)

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(continued)

(a) With Reactor Trip Breakers (RTBs) closed and Rod Control System capable of rod withdrawal.

(b) Below the P-10 (Power Range Neutron Flux) interlocks.

(c) Above the P-6 (Intermediate Range Neutron Flux) interlock.

(d) Below the P-6 (Intermediate Range Neutron Flux) interlock.

D
RAI 3.3.1-12
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Table 3.3.1-1 (page 2 of 8)
Reactor Protection System Instrumentation

FUNCTION	APPLICABLE MODES	REQUIRED CHANNELS	CONDITIONS	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE
7. Pressurizer Pressure					
a. Low	1(e)	4	K	SR 3.3.1.1 SR 3.3.1.7 SR 3.3.1.11	(h)
b. High	1,2	3	D	SR 3.3.1.1 SR 3.3.1.7 SR 3.3.1.11	(i)
8. Pressurizer Water Level — High	1(e)	3	K	SR 3.3.1.1 SR 3.3.1.7 SR 3.3.1.11	≤ 95% of span
9. Reactor Coolant Flow-Low					
a. Single Loop	1(f)	3 per loop	L	SR 3.3.1.1 SR 3.3.1.7 SR 3.3.1.11	≥90%
b. Two Loops	1(g)	3 per loop	K	SR 3.3.1.1 SR 3.3.1.7 SR 3.3.1.11	≥90%
10. Reactor Coolant Pump (RCP) Breaker Position					
a. Single Loop	1(f)	1 per RCP	M	SR 3.3.1.13	NA
b. Two Loops	1(g)	1 per RCP	N	SR 3.3.1.13	NA
11. Undervoltage Bus A01 & A02	1(e)	2 per bus	K	SR 3.3.1.9 SR 3.3.1.10	≥ 3120 V

(continued)

(e) Above the P-7 (Low Power Reactor Trips Block) interlock.

(f) Above the P-8 (Power Range Neutron Flux) interlock.

(g) Above the P-7 (Low Power Reactor Trips Block) interlock and below the P-8 (Power Range Neutron Flux) interlock.

(h) ≥ 1905 psig during operation at 2250 psia, or ≥ 1800 psig during operation at 2000 psia.

(i) ≤ 2385 psig during operation at 2250 psia, or ≤ 2210 psig during operation at 2000 psia.

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RAI 3.3.1-10
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Table 3.3.1-1 (page 3 of 8)
Reactor Protection System Instrumentation

FUNCTION	APPLICABLE MODES	REQUIRED CHANNELS	CONDITIONS	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE
12. Underfrequency Bus A01 & A02	1(e)	2 per bus	E	SR 3.3.1.10	≥ 55.0 Hz
13. Steam Generator (SG) Water Level — Low Low	1,2	3 per SG	D	SR 3.3.1.1 SR 3.3.1.7 SR 3.3.1.11	≥ 20% of span
14. SG Water Level — Low	1,2	2 per SG	D	SR 3.3.1.1 SR 3.3.1.7 SR 3.3.1.11	(j)
Coincident with Steam Flow/Feedwater Flow Mismatch	1,2	2 per SG	D	SR 3.3.1.1 SR 3.3.1.7 SR 3.3.1.11	≤ 1 E6 lbm/hr
15. Turbine Trip					
a. Low Autostop Oil Pressure	1(k)	3	O	SR 3.3.1.14	(m)
b. Turbine Stop Valve Closure	1(k)	2	O	SR 3.3.1.14	NA
16. Safety Injection (SI) Input from Engineered Safety Feature Actuation System (ESFAS)	1,2	2 trains	P	SR 3.3.1.13	NA

(continued)

(e) Above the P-7 (Low Power Reactor Trips Block) interlock.

(j) Field setting of ≥ 30% of span (nominal).

(k) Above the P-9 (Power Range Neutron Flux) interlock.

(m) Field setting of ≥ 45 psig (nominal).

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RAI 3.3.1-1
RAI 3.3.1-7
RAI 3.3.1-26
RAI 3.3.1-27
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Table 3.3.1-1 (page 4 of 8)
Reactor Protection System Instrumentation

FUNCTION	APPLICABLE MODES	REQUIRED CHANNELS	CONDITIONS	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE
17. Reactor Trip System Interlocks					
a. Intermediate Range Neutron Flux, P-6	2(d)	2	R	SR 3.3.1.11 SR 3.3.1.12	≥ 1E-10 amp
b. Low Power Reactor Trips Block, P-7					
(1) Power Range Neutron Flux	1	4	S	SR 3.1.1.11 SR 3.3.1.12	≤ 10% RTP
(2) Turbine Impulse Pressure	1	2	S	SR 3.3.1.11 SR 3.3.1.12	≤ 10% turbine power
c. Power Range Neutron Flux, P-8	1	4	S	SR 3.3.1.11 SR 3.3.1.12	≤ 50% RTP
d. Power Range Neutron Flux, P-9	1(l)	4	S	SR 3.3.1.11 SR 3.3.1.12	≤ 50% RTP
e. Power Range Neutron Flux, P-10	1,2	4	R	SR 3.3.1.11 SR 3.3.1.12	≥ 8% RTP and ≤ 10% RTP
18. Reactor Trip Breakers (RTBs)	1,2	2 trains	Q	SR 3.3.1.4	NA
	3(a), 4(a), 5(a)	2 trains	T	SR 3.3.1.4	NA
19. Reactor Trip Breaker Undervoltage and Shunt Trip Mechanisms	1,2	1 each per RTB	U	SR 3.3.1.4	NA
	3(a), 4(a), 5(a)	1 each per RTB	T	SR 3.3.1.4	NA

(continued)

(a) With the RTBs closed and the Rod Control System capable of rod withdrawal.

(d) Below the P-6 (Intermediate Range Neutron Flux) interlock.

(l) With 1 of 2 circulating water pump breakers closed and condenser vacuum ≥ 22 "Hg.

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RAI 3.3.1-1
RAI 3.3.1-4
RAI 3.3.1-5
RAI 3.3.1-7
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Table 3.3.1-1 (page 5 of 8)
Reactor Protection System Instrumentation

	FUNCTION	APPLICABLE MODES	REQUIRED CHANNELS	CONDITIONS	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE
20.	Reactor Trip Bypass Breaker and associated Undervoltage Trip Mechanism	$1^{(n)}$, $2^{(n)}$	1	V	SR 3.3.1.4	NA
		$3^{(n)}$, $4^{(n)}$, $5^{(n)}$	1	W	SR 3.3.1.4	NA
21.	Automatic Trip Logic	1, 2,	2 trains	P	SR 3.3.1.5 SR 3.3.1.15	NA
		$3^{(a)}$, $4^{(a)}$, $5^{(a)}$	2 trains	X	SR 3.3.1.5	NA

(a) With RTBs closed and Rod Control System capable of rod withdrawal.

(n) When Reactor Trip Bypass Breakers are racked in and closed and the Rod Control System is capable of rod withdrawal.

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Table 3.3.1-1 (page 6 of 8)
Reactor Protection System Instrumentation

Note 1: Overtemperature ΔT

$$\Delta T\,(\frac{1}{1+\tau_3 S}) \leq \Delta T_o\,(K_1 - K_2(T(\frac{1}{1+\tau_4 S}) - T')(\frac{1+\tau_1 S}{1+\tau_2 S}) + K_3(P - P') - f(\Delta I))$$

where (values are applicable to operation at both 2000 psia and 2250 psia unless otherwise indicated)

D
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ΔT_o	=	indicated ΔT at rated power, °F
T	=	average temperature, °F
T'	≤	[*]°F (for cores containing 422V+ fuel assemblies)
T'	≤	[*]°F (for cores not containing 422V+ fuel assemblies)
P	=	pressurizer pressure, psig
P'	=	[*] psig (for 2250 psia operation)
P'	=	[*] psig (for 2000 psia operation and cores not containing 422V+ fuel assemblies)
K_1	≤	[*] (for 2250 psia operation and cores containing 422V+ fuel assemblies)
K_1	≤	[*] (for 2250 psia operation and cores not containing 422V+ fuel assemblies)
K_1	≤	[*] (for 2000 psia operation and cores not containing 422V+ fuel assemblies)
K_2	=	[*] (for 2250 psia operation and cores containing 422V+ fuel assemblies)
K_2	=	[*] (for 2250 psia operation and cores not containing 422V+ fuel assemblies)
K_2	=	[*] (for 2000 psia operation and cores not containing 422V+ fuel assemblies)
K_3	=	[*] (for 2250 psia operation and cores containing 422V+ fuel assemblies)
K_3	=	[*] (for 2250 psia operation and cores not containing 422V+ fuel assemblies)
K_3	=	[*] (for 2000 psia operation and cores not containing 422V+ fuel assemblies)
τ_1	=	[*] sec
τ_2	=	[*] sec
τ_3	=	[*] sec for Rosemont or equivalent RTD
	=	[*] sec for Sostman or equivalent RTD
τ_4	=	[*] sec for Rosemont or equivalent RTD
	=	[*] sec for Sostman or equivalent RTD

and f(ΔI) is an even function of the indicated difference between top and bottom detectors of the power-range nuclear ion chambers; with gains to be selected based on measured instrument response during plant startup tests, where q_t and q_b are the percent power in the top and bottom halves of the core respectively, and $q_t + q_b$ is total core power in percent of rated power, such that:

(a) for $q_t - q_b$ within –[*], +[*] percent, f(ΔI) = 0 for cores not containing 422V+ fuel assemblies; for $q_t - q_b$ within –[*], +[*] percent, f(ΔI) = 0 for cores containing 422V+ fuel assemblies.

(b) for each percent that the magnitude of $q_t - q_b$ exceeds +[*] percent, the ΔT trip setpoint shall be automatically reduced by an equivalent of [*] percent of rated power for cores not containing 422V+ fuel assemblies and reduced by an equivalent of [*] percent of rated power for cores containing 422V+ fuel assemblies.

Table 3.3.1-1 (page 7 of 8)
Reactor Protection System Instrumentation

Note 1: Overtemperature ΔT (continued)

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(c) for cores not containing 422V+ fuel assemblies, for each percent that the magnitude of $q_t - q_b$ exceeds –[*] percent, the ΔT trip setpoint shall be automatically reduced by an equivalent of [*] percent of rated power; for cores containing 422V+ fuel assemblies, for each percent that the magnitude of $q_t - q_b$ exceeds –[*] percent, the ΔT trip setpoint shall be automatically reduced by an equivalent of [*] percent of rated power.

The values denoted with [*] are specified in the COLR.

Table 3.3.1-1 (page 8 of 8)
Reactor Protection System Instrumentation

Note 2: Overpower ΔT

$$\Delta T\left(\frac{1}{1+\tau_3 S}\right) \leq \Delta T_o\left[K_4 - K_5\left(\frac{\tau_5 S}{\tau_5 S+1}\right)\left(\frac{1}{1+\tau_4 S}\right)T - K_6\left[T\left(\frac{1}{1+\tau_4 S}\right) - T'\right]\right]$$

where (values are applicable to operation at both 2000 psia and 2250 psia)

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ΔT_o	=	indicated ΔT at rated power, °F
T	=	average temperature, °F
T'	≤	[*]°F (for cores containing 422V+ fuel assemblies)
T'	≤	[*]°F (for cores not containing 422V+ fuel assemblies)
K_4	≤	[*] of rated power (for cores containing 422V+ fuel assemblies)
K_4	≤	[*] of rated power (for cores not containing 422V+ fuel assemblies)
K_5	=	[*] for increasing T
	=	[*] for decreasing T
K_6	=	[*] for T ≥ T' (for cores containing 422V+ fuel assemblies)
K_6	=	[*] for T ≥ T' (for cores not containing 422V+ fuel assemblies)
	=	[*] for T < T'
τ_5	=	[*] sec
τ_3	=	[*] sec for Rosemont or equivalent RTD
		[*] sec for Sostman or equivalent RTD
τ_4	=	[*] sec for Rosemont or equivalent RTD
		[*] sec for Sostman or equivalent RTD

The values denoted with [*] are specified in the COLR.

B 3.3 INSTRUMENTATION

B 3.3.1 Reactor Protection System (RPS) Instrumentation

BASES

BACKGROUND

The RPS initiates a unit shutdown, based on the values of selected unit parameters, to protect against violating the core fuel design limits and Reactor Coolant System (RCS) pressure boundary during anticipated operational occurrences (AOOs) and to assist the Engineered Safety Features (ESF) Systems in mitigating accidents.

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

The LSSS, defined in this specification as the Allowable Value Setpoints, in conjunction with the LCOs, establish the threshold for protective system action to prevent exceeding acceptable limits during Design Basis Accidents (DBAs).

During AOOs, which are those events expected to occur one or more times during the unit life, the acceptable limits are:

1. The Departure from Nucleate Boiling Ratio (DNBR) shall be maintained above the Safety Limit (SL) value to prevent departure from nucleate boiling (DNB);
2. Fuel centerline melt shall not occur; and
3. The RCS pressure SL of 2750 psia shall not be exceeded.

Operation within the SLs of Specification 2.0, "Safety Limits (SLs)," also maintains the above values and assures that offsite dose will be within the 10 CFR 50 and 10 CFR 100 criteria during AOOs.

Accidents are events that are analyzed even though they are not expected to occur during the unit life. The acceptable limit during accidents is that offsite dose shall be maintained within an acceptable fraction of 10 CFR 100 limits. Different accident categories are allowed a different fraction of these limits, based on probability of occurrence. Meeting the acceptable dose limit for an accident category is considered having acceptable consequences for that event.

BACKGROUND (continued)

The RPS instrumentation is segmented into four distinct but interconnected modules as identified below:

1. Field transmitters or process sensors: provide a measurable electronic signal based upon the physical characteristics of the parameter being measured;

2. Signal Process Control and Protection System, including Analog Protection System, Nuclear Instrumentation System (NIS), field contacts, and protection channel sets: provides signal conditioning, compatible electrical signal output to protection system devices, and control board/control room/miscellaneous indications;

3. Relay Logic System, including input, logic, and output devices: initiates proper unit shutdown in accordance with the defined logic, which is based on bistable, setpoint comparators, or contact outputs from the signal process control and protection systems; and

4. Reactor trip switchgear, including reactor trip breakers (RTBs) and bypass breakers: provides the means to interrupt power to the control rod drive mechanisms (CRDMs) and allows the rod cluster control assemblies (RCCAs), or "rods," to fall into the core and shut down the reactor. The bypass breakers allow testing of the RTBs at power.

<u>Field Transmitters or Sensors</u>

To meet the design demands for redundancy and reliability, more than one, and often as many as four, field transmitters or sensors are used to measure unit parameters. To account for the calibration tolerances and instrument drift, which are assumed to occur between calibrations, statistical allowances are provided in the Allowable Values. The OPERABILITY of each transmitter or sensor can be evaluated when its "as found" calibration data are compared against its documented acceptance criteria.

<u>Signal Process Control and Protection System</u>

Generally, three or four channels of process control equipment are used for the signal processing of unit parameters measured by the field instruments. The process control equipment provides signal conditioning, comparable output signals for instruments located on the main control board, and comparison of measured input signals with setpoints established by safety analyses. If the measured value of a unit parameter exceeds the predetermined setpoint, an output from a bistable is forwarded to the logic relays.

BASES

BACKGROUND (continued)

Generally, if a parameter is used only for input to the protection circuits, three channels with a two-out-of-three logic are sufficient to provide the required reliability and redundancy. If one channel fails in a direction that would not result in a partial Function trip, the Function is still OPERABLE with a two-out-of-two logic. If one channel fails, such that a partial Function trip occurs, a trip will not occur and the Function is still OPERABLE with a one-out-of-two logic.

Generally, if a parameter is used for input to the relay logic system and a control function, four channels with a two-out-of-four logic are sufficient to provide the required reliability and redundancy. The circuit must be able to withstand both an input failure to the control system, which may then require the protection function actuation, and a single failure in the other channels providing the protection function actuation. Again, a single failure will neither cause nor prevent the protection function actuation. These requirements are described in IEEE-279-1968 (Ref. 3). The actual number of channels required for each unit parameter is specified in Reference 1.

Two logic channels are required to ensure no single random failure of a logic channel will disable the RPS. The logic channels are designed such that testing required while the reactor is at power may be accomplished without causing trip. Provisions to allow removing logic channels from service during maintenance are unnecessary because of the logic system's designed reliability.

Allowable Values

To allow for calibration tolerances, instrumentation uncertainties, instrument drift, and severe environment errors for those RPS channels that must function in harsh environments as defined by 10 CFR 50.49 (Ref. 4), the Allowable Values specified in Table 3.3.1-1 in the accompanying LCO are conservatively adjusted with respect to the analytical limits. The actual nominal Trip Setpoint entered into the bistable is more conservative than that specified by the Allowable Value to account for changes in random measurement errors detectable by a COT. One example of such a change in measurement error is drift during the surveillance interval. If the measured setpoint does not exceed the Allowable Value, the bistable is considered OPERABLE.

Setpoints in accordance with the Allowable Value ensure that SLs are not violated during AOOs (and that the consequences of DBAs will be acceptable, providing the unit is operated from within the LCOs at the onset of the AOO or DBA and the equipment functions as designed). Note that in the accompanying LCO 3.3.1, the Allowable Values of Table 3.3.1-1 are the LSSS.

BASES

BACKGROUND (continued)

Each channel of the process control equipment can be tested on line to verify that the signal or setpoint accuracy is within the specified allowance requirements. Once a designated channel is taken out of service for testing, a simulated signal is injected in place of the field instrument signal. The process equipment for the channel in test is then tested, verified, and calibrated. SRs for the channels are specified in the SRs section.

The Allowable Values listed in Table 3.3.1-1 are based on the methodology, which incorporates all of the known uncertainties applicable for each channel. The magnitudes of these uncertainties are factored into the determination of each Allowable Value. All field sensors and signal processing equipment for these channels are assumed to operate within the allowances of these uncertainty magnitudes.

Relay Logic System

The Relay Logic System equipment is used for the decision logic processing of outputs from the signal processing equipment bistables. To meet the redundancy requirements, two trains of Relay Logic System, each performing the same functions, are provided. If one train is taken out of service for maintenance or test purposes, the second train will provide reactor trip for the unit. Each train is packaged in its own cabinet for physical and electrical separation to satisfy separation and independence requirements. The system has been designed to trip in the event of a loss of power, directing the unit to a safe shutdown condition.

The Relay Logic System performs the decision logic for actuating a reactor trip, generates the electrical output signal that will initiate the required trip, and provides the status, permissive, and annunciator output signals to the main control room of the unit.

The bistable outputs from the signal processing equipment are sensed by the Relay Logic System equipment and combined into logic matrices that represent combinations indicative of various unit upset and accident transients. If a required logic matrix combination is completed, the system will initiate a reactor trip. Examples are given in the Applicable Safety Analyses, LCO, and Applicability sections of this Bases.

Reactor Trip Switchgear

The RTBs are in the electrical power supply line from the control rod drive motor generator set power supply to the CRDMs. Opening of the RTBs interrupts power to the CRDMs, which allows the shutdown rods

BASES

BACKGROUND (continued)	and control rods to fall into the core by gravity. Each RTB is equipped with a bypass breaker to allow testing of the RTB while the unit is at power. During normal operation the output from the relay logic system is a voltage signal that energizes the undervoltage coils in the RTBs and bypass breakers, if in use. When the required logic matrix combination is completed, the relay logic system output voltage signal is removed, the undervoltage coils are de-energized, the breaker trip lever is actuated by the de-energized undervoltage coil, and the RTBs and bypass breakers are tripped open. This allows the shutdown rods and control rods to fall into the core. In addition to the de-energization of the undervoltage coils, each RTB is also equipped with a shunt trip device that is energized to trip the breaker open upon receipt of a reactor trip signal from the relay logic system. Either the undervoltage coil or the shunt trip mechanism is sufficient by itself, thus providing a diverse trip mechanism.
APPLICABLE SAFETY ANALYSES, LCO, AND APPLICABILITY	The RPS functions to maintain the SLs during all AOOs and mitigates the consequences of DBAs in all MODES in which the RTBs are closed. Each of the analyzed accidents and transients can be detected by one or more RPS Functions. The accident analysis described in Reference 2 takes credit for most RPS trip Functions. RPS trip Functions not specifically credited in the accident analysis are qualitatively credited in the safety analysis and the NRC staff approved licensing basis for the unit. These RPS trip Functions may provide protection for conditions that do not require dynamic transient analysis to demonstrate Function performance. They may also serve as backups to RPS trip Functions that were credited in the accident analysis. The LCO requires all instrumentation performing an RPS Function, listed in Table 3.3.1-1 in the accompanying LCO, to be OPERABLE. Failure of any instrument renders the affected channel(s) inoperable and reduces the reliability of the affected Functions. The LCO generally requires OPERABILITY of four or three channels in each instrumentation Function, one channel of Manual Reactor Trip in each logic Function, and two trains in each Automatic Trip Logic Function. Four OPERABLE instrumentation channels in a two-out-of-four configuration are generally required when one RPS channel is also used as a control system input. This configuration accounts for the possibility of the shared channel failing in such a manner that it creates a transient that requires RPS action. In this case, the RPS will still provide protection, even with random failure of one of the other three protection channels. Three OPERABLE instrumentation channels in a two-out-of-three configuration are generally required when there is no

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APPLICABLE SAFETY ANALYSES, LCO, AND APPLICABILITY (continued)

potential for control system and protection system interaction that could simultaneously create a need for RPS trip and disable one RPS channel. The two-out-of-three and two-out-of-four configurations allow one channel to be tripped during maintenance or testing without causing a reactor trip. Specific exceptions to the above general philosophy exist and are discussed below.

Reactor Protection System Functions

The safety analyses and OPERABILITY requirements applicable to each RPS Function are discussed below:

1. Manual Reactor Trip

   The Manual Reactor Trip ensures that the control room operator can initiate a reactor trip at any time by using one of four reactor trip switches in the control room. A Manual Reactor Trip accomplishes the same results as any one of the automatic trip Functions. It is used by the reactor operator to shut down the reactor whenever any parameter is rapidly trending toward its Allowable Value.

   The LCO requires two Manual Reactor Trip channels to be OPERABLE. Each channel consists of two reactor trip switches (one in each train). Each channel activates the reactor trip breaker in both trains. Two independent channels are required to be OPERABLE so that no single random failure will disable the Manual Reactor Trip Function. D RAI 3.3.1-1 Errata #126

   In MODE 1 or 2, manual initiation of a reactor trip must be OPERABLE. These are the MODES in which the shutdown rods and/or control rods are partially or fully withdrawn from the core. In MODE 3, 4, or 5, the manual initiation Function must also be OPERABLE with the RTBs closed and the Rod Control System capable of rod withdrawal. In this condition, inadvertent control rod withdrawal is possible. In MODE 3, 4, or 5, manual initiation of a reactor trip does not have to be OPERABLE if the Rod Control System is not capable of withdrawing the shutdown rods or control rods. If the rods cannot be withdrawn from the core or all of the rods are inserted, there is no need to be able to trip the reactor. In MODE 6, neither the shutdown rods nor the control rods are permitted to be withdrawn and the CRDMs are disconnected from the control rods and shutdown rods. Therefore, the manual initiation Function is not required.

BASES

APPLICABLE SAFETY ANALYSES, LCO AND APPLICABILITY (continued)

2. Power Range Neutron Flux

The NIS power range detectors are located external to the reactor vessel and measure neutrons leaking from the core. The NIS power range detectors provide input to the Rod Control System. Therefore, the actuation logic must be able to withstand an input failure to the control system, which may then require the protection function actuation, and a single failure in the other channels providing the protection function actuation. Note that this Function also provides a signal to prevent automatic and manual rod withdrawal prior to initiating a reactor trip. Limiting further rod withdrawal may terminate the transient and eliminate the need to trip the reactor.

a. Power Range Neutron Flux-High

The Power Range Neutron Flux-High trip Function ensures that protection is provided, from all power levels, against a positive reactivity excursion leading to DNB during power operations. These can be caused by rod withdrawal or reductions in RCS temperature.

The LCO requires all four of the Power Range Neutron Flux-High channels to be OPERABLE.

In MODE 1 or 2, the Power Range Neutron Flux-High trip must be OPERABLE. This Function will terminate the reactivity excursion and shut down the reactor prior to reaching a power level that could damage the fuel. In MODE 3, 4, 5, or 6, the NIS power range detectors cannot detect neutron levels in this range. In these MODES, the Power Range Neutron Flux - High does not have to be OPERABLE because the reactor is shut down and reactivity excursions into the power range are extremely unlikely. Other RTS Functions and administrative controls provide protection against reactivity additions when in MODE 3, 4, 5, or 6.

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Errata #145

b. Power Range Neutron Flux-Low

The LCO requirement for the Power Range Neutron Flux-Low trip Function ensures that protection is provided against a positive reactivity excursion from low power or subcritical conditions.

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APPLICABLE SAFETY ANALYSES, LCO AND APPLICABILITY (continued)

The LCO requires all four of the Power Range Neutron Flux-Low channels to be OPERABLE.

In MODE 1, below the Power Range Neutron Flux (P-10 setpoint), and in MODE 2, the Power Range Neutron Flux-Low trip must be OPERABLE. This Function may be manually blocked by the operator when two out of four power range channels are greater than approximately 10% RTP (P-10 setpoint). This Function is automatically unblocked when three out of four power range channels are below the P-10 setpoint. Above the P-10 setpoint, positive reactivity additions are mitigated by the Power Range Neutron Flux-High trip Function. D Errata #145

In MODE 3, 4, 5, or 6, the Power Range Neutron Flux - Low trip Function does not have to be OPERABLE because the reactor is shut down and the NIS power range detectors cannot detect neutron levels in this range. Other RPS trip Functions and administrative controls provide protection against positive reactivity additions or power excursions in MODE 3, 4, 5, or 6.

3. Intermediate Range Neutron Flux

The Intermediate Range Neutron Flux trip Function ensures that protection is provided against an uncontrolled RCCA bank rod withdrawal accident from a subcritical condition during startup. This trip Function provides redundant protection to the Power Range Neutron Flux-Low Setpoint trip Function. The NIS intermediate range detectors are located external to the reactor vessel and measure neutrons leaking from the core. The NIS intermediate range detectors do not provide any input to control systems.

The LCO requires two channels of Intermediate Range Neutron Flux to be OPERABLE. Two OPERABLE channels are sufficient to ensure no single random failure will disable this trip Function. D RAI 3.3.1-1 Errata #145

Because this trip Function is important only during startup, there is generally no need to disable channels for testing while the Function is required to be OPERABLE. Therefore, a third channel is unnecessary.

In MODE 1 below the P-10 setpoint, and in MODE 2, when there is a potential for an uncontrolled RCCA bank rod withdrawal accident during reactor startup, the Intermediate Range Neutron Flux trip must be OPERABLE. Above the P-10 setpoint, the Power Range Neutron Flux-High Setpoint trip provides core protection for a rod withdrawal accident. In MODE 3, 4, or 5, Intermediate Range Neutron Flux trip does not have to be OPERABLE because the

BASES

APPLICABLE SAFETY ANALYSES, LCO AND APPLICABILITY (continued)

control rods must be fully inserted and only the shutdown rods may be withdrawn. The reactor cannot be started up in this condition. The core also has the required SDM to mitigate the consequences of a positive reactivity addition accident. In MODE 6, all rods are fully inserted and the core has a required increased SDM. Also, the NIS intermediate range detectors cannot detect neutron levels present in this MODE.

D RAI 3.3.1-1 Errata #145

4. Source Range Neutron Flux

The LCO requirement for the Source Range Neutron Flux trip Function ensures that protection is provided against an uncontrolled RCCA bank rod withdrawal accident from a subcritical condition during startup.

This trip Function provides redundant protection to the Power Range Neutron Flux-Low trip Function. In MODES 3, 4, and 5, administrative controls also prevent the uncontrolled withdrawal of rods. The NIS source range detectors are located external to the reactor vessel and measure neutrons leaking from the core. The NIS source range detectors do not provide any inputs to control systems. The source range trip is the only RPS automatic protection function required in MODES 3, 4, and 5. Therefore, the functional capability at the specified Trip Setpoint is assumed to be available.

D Errata #145

The LCO requires two channels of Source Range Neutron Flux to be OPERABLE. Two operable channels are sufficient to ensure no single random failure will disable this trip function.

D RAI 3.3.1-1

The Source Range Neutron Flux Function provides protection for control rod withdrawal from subcritical and control rod ejection events.

In MODE 2 when below the P-6 setpoint, and in MODES 3, 4 and 5 when there is a potential for an uncontrolled RCAA bank rod withdrawal accident, the Source Range Neutron Flux trip must be OPERABLE. Above the P-6 setpoint, the Intermediate Range Neutron Flux trip and the Power Range Neutron Flux-Low Setpoint trip will provide core protection for reactivity accidents. Above the P-6 setpoint, the NIS source range detectors are de-energized.

D RAI 3.3.1-12

In MODES 3, 4 and 5 with the Rod Control System not capable of rod withdrawal, and in MODE 6, this Function is not required to be OPERABLE. The requirements for the NIS source range detectors to monitor core neutron levels and provide indication of reactivity

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APPLICABLE SAFETY ANALYSES, LCO AND APPLICABILITY (continued)

changes that may occur as a result of events like a boron dilution are addressed in LCO 3.9.2, "Nuclear Instrumentation," for MODE 6.

5. Overtemperature ΔT

The Overtemperature ΔT trip Function is provided to ensure that the design limit DNBR is met. This trip Function also limits the range over which the Overpower ΔT trip Function must provide protection. The inputs to the Overtemperature ΔT trip include all pressure, coolant temperature, axial power distribution, and reactor power as indicated by loop ΔT assuming full reactor coolant flow. Protection from violating the DNBR limit is assured for those transients that are slow with respect to delays from the core to the measurement system. The Function monitors both variation in power and flow since a decrease in flow has the same effect on ΔT as a power increase. The Overtemperature ΔT trip Function uses each loop's ΔT as a measure of reactor power and is compared with a setpoint that is automatically varied with the following parameters:

- reactor coolant average temperature-the Trip Setpoint is varied to correct for changes in coolant density and specific heat capacity with changes in coolant temperature;
- pressurizer pressure-the Trip Setpoint is varied to correct for changes in system pressure; and
- axial power distribution — $f(\Delta I)$, the Trip Setpoint is varied to account for imbalances in the axial power distribution as detected by the NIS upper and lower power range detectors. If axial peaks are greater than the design limit, as indicated by the difference between the upper and lower NIS power range detectors, the Trip Setpoint is reduced in accordance with Note 1 of Table 3.3.1-1.

The Overtemperature ΔT trip Function is calculated for each channel as described in Note 1 of Table 3.3.1-1. Reactor Trip occurs if Overtemperature ΔT is indicated in two channels. Because the pressure and temperature signals are used for other control functions, the actuation logic must be able to withstand an input failure to the control system, which may then require the protection function actuation, and a single failure in the other channels providing the protection function actuation. Note that this Function also provides a signal to generate a turbine runback prior to reaching the Trip Setpoint. A turbine runback will reduce turbine power and reactor power. A reduction in power will normally alleviate the Overtemperature ΔT condition and may prevent a reactor trip.

BASES

APPLICABLE SAFETY ANALYSES, LCO AND APPLICABILITY (continued)

The LCO requires all four channels of the Overtemperature ΔT trip Function to be OPERABLE. Note that the Overtemperature ΔT Function receives input from channels shared with other RPS Functions. Failures that affect multiple Functions require entry into the Conditions applicable to all affected Functions.

In MODE 1 or 2, the Overtemperature ΔT trip must be OPERABLE to prevent DNB. In MODE 3, 4, 5, or 6, this trip Function does not have to be OPERABLE because the reactor is not operating and there is insufficient heat production to be concerned about DNB.

6. <u>Overpower ΔT</u>

The Overpower ΔT trip Function ensures that protection is provided to ensure the integrity of the fuel (i.e., no fuel pellet melting and less than 1% cladding strain) under all possible overpower conditions. This trip Function also limits the required range of the Overtemperature ΔT trip Function and provides a backup to the Power Range Neutron Flux-High Setpoint trip. The Overpower ΔT trip Function ensures that the allowable heat generation rate (kW/ft) of the fuel is not exceeded. It uses the ΔT of each loop as a measure of reactor power with a setpoint that is automatically varied with the following parameters:

- reactor coolant average temperature — the Trip Setpoint is varied to correct for changes in coolant density and specific heat capacity with changes in coolant temperature; and

- rate of change of reactor coolant average temperature.

The Overpower ΔT trip Function is calculated for each channel as per Note 2 of Table 3.3.1-1. Trip occurs if Overpower ΔT is indicated in two channels. The temperature signals are used for other control functions. The actuation logic must be able to withstand an input failure to the control system, which may then require the protection function actuation and a single failure in the remaining channels providing the protection function actuation. Note that this Function also provides a signal to generate a turbine runback prior to reaching the Allowable Value. A turbine runback will reduce turbine power and reactor power. A reduction in power will normally alleviate the Overpower ΔT condition and may prevent a reactor trip.

BASES

APPLICABLE SAFETY ANALYSES, LCO AND APPLICABILITY (continued)

The LCO requires four channels of the Overpower ΔT trip Function to be OPERABLE. Note that the Overpower ΔT trip Function receives input from channels shared with other RPS Functions. Failures that affect multiple Functions require entry into the Conditions applicable to all affected Functions.

In MODE 1 or 2, the Overpower ΔT trip Function must be OPERABLE. These are the only times that enough heat is generated in the fuel to be concerned about the heat generation rates and overheating of the fuel. In MODE 3, 4, 5, or 6, this trip Function does not have to be OPERABLE because the reactor is not operating and there is insufficient heat production to be concerned about fuel overheating and fuel damage.

7. Pressurizer Pressure

The same sensors provide input to the Pressurizer Pressure-High and -Low trips and the Overtemperature ΔT trip. The Pressurizer Pressure channels are also used to provide input to the Pressurizer Pressure Control System. The actuation logic must be able to withstand an input failure to the control system, which may then require the protection function actuation, and a single failure in the other channels providing the protection function actuation.

a. Pressurizer Pressure-Low

The Pressurizer Pressure-Low trip Function ensures that protection is provided against violating the DNBR limit due to low pressure.

The LCO requires four channels of Pressurizer Pressure-Low to be OPERABLE.

In MODE 1, when DNB is a major concern, the Pressurizer Pressure-Low trip must be OPERABLE. This trip Function is automatically enabled on increasing power by the P-7 interlock (NIS power range P-10 or turbine impulse pressure greater than approximately 10% of full power equivalent). On decreasing power, this trip Function is automatically blocked below P-7. Below the P-7 interlock, no conceivable power distributions can occur that would cause DNB concerns. D RAI 3.3.1-4

BASES

APPLICABLE SAFETY ANALYSES, LCO AND APPLICABILITY (continued)

b. Pressurizer Pressure-High

The Pressurizer Pressure-High trip Function ensures that protection is provided against overpressurizing the RCS. This trip Function operates in conjunction with the pressurizer relief and safety valves to prevent RCS overpressure conditions.

The LCO requires three channels of the Pressurizer Pressure-High to be OPERABLE.

For operation at 2250 psia, the Pressurizer Pressure-High LSSS is selected to be below the pressurizer safety valve actuation pressure and above the power operated relief valve (PORV) setting. This setting minimizes challenges to safety valves while avoiding unnecessary reactor trip for those pressure increases that can be controlled by the PORVs.

For operation at 2000 psia, a 50% load rejection with steam dump results in a peak pressure below the Pressurizer Pressure-High LSSS. Therefore, even though the PORV setting is above the reactor trip, the transient will not result in PORV actuation or a reactor trip on high Pressurizer Pressure.

In MODE 1 or 2, the Pressurizer Pressure-High trip must be OPERABLE to help prevent RCS overpressurization and minimize challenges to the relief and safety valves. In MODE 3, 4, 5, or 6, the Pressurizer Pressure-High trip Function does not have to be OPERABLE because transients that could cause an overpressure condition will be slow to occur. Therefore, the operator will have sufficient time to evaluate unit conditions and take corrective actions. Additionally, low temperature overpressure protection systems provide overpressure protection when below MODE 4.

8. Pressurizer Water Level—High

The Pressurizer Water Level-High trip Function provides a backup signal for the Pressurizer Pressure-High trip and also provides protection against water relief through the pressurizer safety valves. These valves are designed to pass steam in order to achieve their design energy removal rate. A reactor trip is actuated prior to the pressurizer becoming water solid. The LCO requires three channels of Pressurizer Water Level-High to be OPERABLE. The pressurizer level channels are used as input to the Pressurizer Level Control System. A fourth channel is not required to address control/protection interaction concerns. The level channels do not actuate the safety valves, and the high pressure reactor trip is set

BASES

APPLICABLE SAFETY ANALYSES, LCO AND APPLICABILITY (continued)

below the safety valve setting. Therefore, with the slow rate of charging available, pressure overshoot due to level channel failure cannot cause the safety valve to lift before reactor high pressure trip.

In MODE 1, when there is a potential for overfilling the pressurizer, the Pressurizer Water Level-High trip must be OPERABLE. This trip Function is automatically enabled on increasing power by the P-7 interlock. On decreasing power, this trip Function is automatically blocked below P-7. Below the P-7 interlock, transients that could raise the pressurizer water level will be slow and the operator will have sufficient time to evaluate unit conditions and take corrective actions. D RAI 3.3.1-4

9. <u>Reactor Coolant Flow-Low</u>

a. <u>Reactor Coolant Flow-Low (Single Loop)</u>

The Reactor Coolant Flow—Low (Single Loop) trip Function ensures that protection is provided against violating the DNBR limit due to low flow in one or more RCS loops, while avoiding reactor trips due to normal variations in loop flow. Above the P-8 setpoint, which is approximately 50% RTP, a loss of flow in any RCS loop will actuate a reactor trip. Each RCS loop has three flow detectors to monitor flow. The flow signals are not used for any control system input. D RAI 3.3.1-5

The LCO requires three Reactor Coolant Flow-Low channels per loop to be OPERABLE in MODE 1 above P-8.

In MODE 1 above the P-8 setpoint, a loss of flow in one RCS loop could result in DNB conditions in the core. In MODE 1 below the P-8 setpoint, a loss of flow in two loops is required to actuate a reactor trip (Function 9.b) because of the lower power level and the greater margin to the design limit DNBR.

b. <u>Reactor Coolant Flow-Low (Two Loops)</u>

The Reactor Coolant Flow-Low (Two Loops) trip Function ensures that protection is provided against violating the DNBR limit due to low flow in two or more RCS loops while avoiding reactor trips due to normal variations in loop flow.

BASES

APPLICABLE SAFETY ANALYSES, LCO AND APPLICABILITY (continued)

Above the P-7 interlock and below the P-8 setpoint, a loss of flow in two loops will initiate a reactor trip. Each loop has three flow detectors to monitor flow. The flow signals are not used for any control system input. D RAI 3.3.1-4

The LCO requires three Reactor Coolant Flow-Low channels per loop to be OPERABLE.

In MODE 1 above the P-7 interlock and below the P-8 setpoint, the Reactor Coolant Flow-Low (Two Loops) trip must be OPERABLE. Below the P-7 interlock, all reactor trips on low flow are automatically blocked since no conceivable power distributions could occur that would cause a DNB concern at this low power level. Above the P-7 interlock, the reactor trip on low flow in two RCS loops is automatically enabled. Above the P-8 setpoint, a loss of flow in any one loop will actuate a reactor trip because of the higher power level and the reduced margin to the design limit DNBR. D RAI 3.3.1-4

10. Reactor Coolant Pump (RCP) Breaker Position

Both RCP Breaker Position trip Functions operate together on two sets of auxiliary contacts, with one set on each RCP breaker. These Functions anticipate the Reactor Coolant Flow-Low trips to avoid RCS heatup that would occur before the low flow trip actuates.

a. Reactor Coolant Pump Breaker Position (Single Loop)

The RCP Breaker Position (Single Loop) trip Function ensures that protection is provided against violating the DNBR limit due to a loss of flow in one RCS loop. The position of each RCP breaker is monitored. If one RCP breaker is open above the P-8 setpoint, a reactor trip is initiated. This trip Function will generate a reactor trip before the Reactor Coolant Flow-Low (Single Loop) Trip Setpoint is reached. D RAI 3.3.1-10

The LCO requires one RCP Breaker Position channel per RCP to be OPERABLE. A channel consists of the RCP Breaker auxiliary contact and the associated RCP Loss of Power Trip Matrix Relay. One OPERABLE channel is sufficient for this trip Function because the RCS Flow-Low trip alone provides sufficient protection of unit SLs for loss of flow events. The RCP Breaker Position trip serves only to anticipate the low flow trip, minimizing the thermal transient associated with loss of a pump.

BASES

APPLICABLE SAFETY ANALYSES, LCO AND APPLICABILITY (continued)

This Function measures only the discrete position (open or closed) of the RCP breaker, using a position switch. Therefore, the Function has no adjustable trip setpoint with which to associate an LSSS.

In MODE 1 above the P-8 setpoint, when a loss of flow in any RCS loop could result in DNB conditions in the core, the RCP Breaker Position (Single Loop) trip must be OPERABLE. In MODE 1 below the P-8 setpoint, a loss of flow in two loops is required to actuate a reactor trip because of the lower power level and the greater margin to the design limit DNBR.

b. Reactor Coolant Pump Breaker Position (Two Loops)

The RCP Breaker Position (Two Loops) trip Function ensures that protection is provided against violating the DNBR limit due to a loss of flow in two RCS loops. The position of each RCP breaker is monitored. Above the P-7 interlock and below the P-8 setpoint, a loss of flow in two loops will initiate a reactor trip. This trip Function will generate a reactor trip before the Reactor Coolant Flow—Low (Two Loops) Trip Setpoint is reached.

D
RAI 3.3.1-10

The LCO requires one RCP Breaker Position channel per RCP to be OPERABLE. A channel consists of the RCP Breaker auxiliary contact and the associated RCP Loss of Power Trip Matrix Relay. One OPERABLE channel is sufficient for this Function because the RCS Flow—Low trip alone provides sufficient protection of unit SLs for loss of flow events. The RCP Breaker Position trip serves only to anticipate the low flow trip, minimizing the thermal transient associated with loss of an RCP.

This Function measures only the discrete position (open or closed) of the RCP breaker, using a position switch. Therefore, the Function has no adjustable trip setpoint with which to associate an LSSS.

In MODE 1 above the P-7 interlock and below the P-8 setpoint, the RCP Breaker Position (Two Loops) trip must be OPERABLE. Below the P-7 interlock, all reactor trips on loss of flow are automatically blocked since no conceivable power distributions could occur that would cause a DNB concern at this low power level. Above the P-7 interlock, the reactor trip on loss of flow in two RCS loops is automatically enabled. Above the P-8 setpoint, a loss of flow in any one loop will actuate a reactor trip because of the higher power level and the reduced margin to the design limit DNBR.

D
RAI 3.3.1-10

BASES

APPLICABLE SAFETY ANALYSES, LCO AND APPLICABILITY (continued)

11. Undervoltage Bus A01 and A02

The Undervoltage Bus A01 and A02 reactor trip Function ensures that protection is provided against violating the DNBR limit due to a loss of flow in both RCS loops. The voltage to Bus A01 and A02 is monitored. Above the P-7 interlock, a loss of voltage detected on both buses will initiate a reactor trip. This trip Function will generate a reactor trip before the Reactor Coolant Flow—Low (Two Loops) Trip Setpoint is reached. Time delays are incorporated into the Undervoltage Bus A01 and A02 channels to prevent reactor trips due to momentary electrical power transients. D RAI 3.3.1-8

The LCO requires two Undervoltage channels per bus to be OPERABLE. An Undervoltage channel consists of the A01/A02 Bus Undervoltage Relay and the associated Bus Undervoltage Matrix Relay.

In MODE 1 above the P-7 interlock, the Undervoltage Bus A01 and A02 trip must be OPERABLE. Below the P-7 interlock, all reactor trips on loss of flow are automatically blocked since no conceivable power distributions could occur that would cause a DNB concern at this low power level. Above the P-7 interlock, the reactor trip on loss of flow in both RCS loops is automatically enabled. D RAI 3.3.1-8

12. Underfrequency Bus A01 and A02

The Underfrequency Bus A01 and A02 RCP breaker trip Function ensures that protection is provided against violating the DNBR limit due to a loss of flow in two RCS loops from a major network frequency disturbance. An underfrequency condition will slow down the pumps, thereby reducing their coastdown time following a pump trip. The proper coastdown time is required so that reactor heat can be removed immediately after reactor trip. The frequency of each RCP bus is monitored. Above the P-7 interlock, a loss of frequency detected on two RCP buses will trip both RCP breakers. Tripping both RCP breakers will generate a reactor trip before the Reactor Coolant Flow—Low (Two Loops) Trip Setpoint is reached. Time delays are incorporated into the Underfrequency Bus A01 and A02 channels to prevent reactor trips due to momentary electrical power transients. D RAI 3.3.1-8

The LCO requires two Underfrequency Bus A01 channels and two Underfrequency Bus A02 channels to be OPERABLE. D RAI 3.3.1-1

In MODE 1 above the P-7 interlock, the Underfrequency Bus A01 and A02 RCP breaker trip must be OPERABLE. Below the P-7 interlock, this trip and all reactor trips on loss of flow are

BASES

APPLICABLE SAFETY ANALYSES, LCO AND APPLICABILITY (continued)

automatically blocked, because no conceivable power distributions could occur that would cause a DNB concern at this low power level. Above the P-7 interlock, the Underfrequency Bus A01 and A02 RCP breaker trip is automatically enabled.

13. Steam Generator Water Level—Low Low

The SG Water Level—Low Low trip Function ensures that protection is provided against a loss of heat sink and actuates the AFW System prior to uncovering the SG tubes. The SGs are the heat sink for the reactor. In order to act as a heat sink, the SGs must contain a minimum amount of water. A narrow range low low level in any SG is indicative of a loss of heat sink for the reactor. The level transmitters provide input to the SG Level Control System. Therefore, the actuation logic must be able to withstand an input failure to the control system, which may then require the protection function actuation, and a single failure in the other channels providing the protection function actuation. This Function also performs the ESFAS function of starting the AFW pumps on low low SG level.

The LCO requires three channels of SG Water Level—Low Low per SG to be OPERABLE.

In MODE 1 or 2, when the reactor requires a heat sink, the SG Water Level—Low Low trip must be OPERABLE. The normal source of water for the SGs is the Main Feedwater (MFW) System (not safety related). The MFW System is only in operation in MODE 1 or 2. The AFW System is the safety related backup source of water to ensure that the SGs remain the heat sink for the reactor. During normal startups and shutdowns, the AFW System provides feedwater to maintain SG level. In MODE 3, 4, 5, or 6, the SG Water Level—Low Low Function does not have to be OPERABLE because the MFW System is not in operation and the reactor is not operating or even critical. Decay heat removal is accomplished by the AFW System in MODE 3 and by the Residual Heat Removal (RHR) System in MODE 4, 5, or 6.

14. Steam Generator Water Level—Low, Coincident With Steam Flow/Feedwater Flow Mismatch

SG Water Level-Low, in conjunction with the Steam Flow/Feedwater Flow Mismatch, ensures that protection is provided against a loss of heat sink. In addition to a decreasing water level in the SG, the difference between feedwater flow and steam flow is evaluated to determine if feedwater flow is significantly less than steam flow.

BASES

APPLICABLE SAFETY ANALYSES, LCO AND APPLICABILITY (continued)

With less feedwater flow than steam flow, SG level will decrease at a rate dependent upon the magnitude of the difference in flow rates. There are two SG level channels and two Steam Flow/Feedwater Flow Mismatch channels per SG. One narrow range level channel sensing a low level coincident with one Steam Flow/ Feedwater Flow Mismatch channel sensing flow mismatch (steam flow greater than feed flow) will actuate a reactor trip.

The field setting for SG Water Level - Low was developed outside the setpoint methodology and has been provided by the NSSS supplier. No analytical value is assumed in the accident analyses for this function. D RAI 3.3.1-26 RAI 3.3.1-1

The LCO requires two channels of SG Water Level-Low coincident with Steam Flow/Feedwater Flow Mismatch per SG.

In MODE 1 or 2, when the reactor requires a heat sink, the SG Water Level-Low coincident with Steam Flow/Feedwater Flow Mismatch trip must be OPERABLE. The normal source of water for the SGs is the MFW System (not safety related). The MFW System is only in operation in MODE 1 or 2. The AFW System is the safety related backup source of water to ensure that the SGs remain the heat sink for the reactor. During normal startups and shutdowns, the AFW System provides feedwater to maintain SG level. In MODE 3, 4, 5, or 6, the SG Water Level-Low coincident with Steam Flow/Feedwater Flow Mismatch Function does not have to be OPERABLE because the MFW System is not in operation and the reactor is not operating or even critical. Decay heat removal is accomplished by the AFW System in MODE 3 and by the RHR System in MODE 4, 5, or 6. The MFW System is in operation only in MODE 1 or 2 and, therefore, this trip Function need only be OPERABLE in these MODES.

15. Turbine Trip

a. Turbine Trip-Low Autostop Oil Pressure

The Turbine Trip-Low Autostop Oil Pressure trip Function anticipates the loss of heat removal capabilities of the secondary system following a turbine trip. This trip Function acts to minimize the pressure/temperature transient on the reactor. Any turbine trip from a power level below the P-9 setpoint (approximately 50% power, with at least one circulating water pump breaker closed, and condenser vacuum not high, will not actuate a reactor trip. Three pressure switches monitor the control oil pressure in the Turbine Electrohydraulic Control System. A low pressure condition sensed by two-out-of-three D RAI 3.3.1-27

BASES

APPLICABLE SAFETY ANALYSES, LCO AND APPLICABILITY (continued)

pressure switches will actuate a reactor trip. These pressure switches do not provide any input to the control system. The unit is designed to withstand a complete loss of load and not sustain core damage or challenge the RCS pressure limitations. Core protection is provided by the Pressurizer Pressure-High trip Function and RCS integrity is ensured by the pressurizer safety valves.

The field setting for the Turbine Trip - Low Autostop Oil Pressure trip function was developed outside of the setpoint methodology and has been provided by the NSSS supplier. No analytical value is assumed in the accident analyses for this function. D RAI 3.3.1-27

The LCO requires three channels of Turbine Trip-Low Autostop Oil Pressure to be OPERABLE in MODE 1 above P-9. D RAI 3.3.1-7

Below the P-9 setpoint, a turbine trip does not actuate a reactor trip. In MODE 2, 3, 4, 5, or 6, there is no potential for a turbine trip, and the Turbine Trip-Low Autostop Oil Pressure trip Function does not need to be OPERABLE.

b. Turbine Trip-Turbine Stop Valve Closure

The Turbine Trip-Turbine Stop Valve Closure trip Function anticipates the loss of heat removal capabilities of the secondary system following a turbine trip. Any turbine trip with from a power level below the P-9 setpoint, approximately 50% power, with at least one circulating water pump breaker closed, and condenser vacuum not high, will not actuate a reactor trip. The trip Function anticipates the loss of secondary heat removal capability that occurs when the stop valves close. Tripping the reactor in anticipation of loss of secondary heat removal acts to minimize the pressure and temperature transient on the reactor. This trip Function will not and is not required to operate in the presence of a single channel failure. The unit is designed to withstand a complete loss of load and not sustain core damage or challenge the RCS pressure limitations. Core protection is provided by the Pressurizer Pressure-High trip Function, and RCS integrity is ensured by the pressurizer safety valves. This trip Function is diverse to the Turbine Trip-Low Autostop Oil Pressure trip Function. Each turbine stop valve is equipped with one limit switch that inputs to the RPS. If both limit switches indicate that the stop valves are all closed, a reactor trip is initiated. D RAI 3.3.1-27

No analytical value is assumed in the accident analyses for this function.

BASES

APPLICABLE SAFETY ANALYSES, LCO AND APPLICABILITY (continued)

The LCO requires two Turbine Trip-Turbine Stop Valve Closure channels, one per valve, to be OPERABLE in MODE 1above P-9. Both channels must trip to cause reactor trip. D

RAI 3.3.1-7

Below the P-9 setpoint, a load rejection can be accommodated by the Steam Dump System. In MODE 2, 3, 4, 5, or 6, there is no potential for a load rejection, and the Turbine Trip-Stop Valve Closure trip Function does not need to be OPERABLE.

16. Safety Injection Input from Engineered Safety Feature Actuation System

The SI Input from ESFAS ensures that if a reactor trip has not already been generated by the RPS, the ESFAS automatic actuation logic will initiate a reactor trip upon any signal that initiates SI. This is a condition of acceptability for the LOCA. However, other transients and accidents take credit for varying levels of ESF performance and rely upon rod insertion, except for the most reactive rod that is assumed to be fully withdrawn, to ensure reactor shutdown. Therefore, a reactor trip is initiated every time an SI signal is present.

Allowable Values are not applicable to this Function. The SI Input is provided by relay in the ESFAS. Therefore, there is no measurement signal with which to associate an LSSS.

The LCO requires two trains of SI Input from ESFAS to be OPERABLE in MODE 1 or 2.

A reactor trip is initiated every time an SI signal is present. Therefore, this trip Function must be OPERABLE in MODE 1 or 2, when the reactor is critical, and must be shut down in the event of an accident. In MODE 3, 4, 5, or 6, the reactor is not critical, and this trip Function does not need to be OPERABLE.

17. Reactor Protection System Interlocks

Reactor protection interlocks are provided to ensure reactor trips are in the correct configuration for the current unit status. They back up operator actions to ensure protection system Functions are not bypassed during unit conditions under which the safety analysis assumes the Functions are not bypassed. Therefore, the interlock Functions do not need to be OPERABLE when the associated reactor trip functions are outside the applicable MODES. These are:

BASES

APPLICABLE SAFETY ANALYSES, LCO AND APPLICABILITY (continued)

a. Intermediate Range Neutron Flux, P-6

The Intermediate Range Neutron Flux, P-6 interlock is actuated when any NIS intermediate range channel goes approximately one decade above the minimum channel reading. If both channels drop below the setpoint, the permissive will automatically be defeated. The LCO requirement for the P-6 interlock ensures that the following Functions are performed:

- on increasing power, the P-6 interlock allows the manual block of the NIS Source Range, Neutron Flux reactor trip. This prevents a premature block of the source range trip and allows the operator to ensure that the intermediate range is OPERABLE prior to leaving the source range. When the source range trip is blocked, the high voltage to the detectors is also removed; and

- on decreasing power, the P-6 interlock automatically energizes the NIS source range detectors and enables the NIS Source Range Neutron Flux reactor trip.

The LCO requires two channels of Intermediate Range Neutron Flux, P-6 interlock to be OPERABLE in MODE 2 when below the P-6 interlock setpoint. D RAI 3.3.1-12

Above the P-6 interlock setpoint, the NIS Source Range Neutron Flux reactor trip will be blocked, and this Function will no longer be necessary.

b. Low Power Reactor Trips Block, P-7

The Low Power Reactor Trips Block, P-7 interlock is actuated by input from either Power Range Neutron Flux or Turbine Impulse Pressure. The LCO requirement for the P-7 interlock ensures that the following Functions are performed: D RAI 3.3.1-14

(1) on increasing power, the P-7 interlock automatically enables reactor trips on the following Functions:
   - Pressurizer Pressure - Low;
   - Pressurizer Water Level - High;
   - Reactor Coolant Flow - Low (Two Loops);
   - RCP Breaker Open (Two Loops);
   - Undervoltage Bus A01 and A02; and
   - Underfrequency Bus A01 and A02.

BASES

APPLICABLE SAFETY ANALYSES, LCO AND APPLICABILITY (continued)

These reactor trips are only required when operating above the P-7 setpoint (approximately 10% power). The reactor trips provide protection against violating the DNBR limit. Below the P-7 setpoint, the RCS is capable of providing sufficient natural circulation without any RCP running.

(2) on decreasing power, the P-7 interlock automatically blocks reactor trips on the following Functions:
- Pressurizer Pressure - Low;
- Pressurizer Water Level - High;
- Reactor Coolant Flow - Low (Two Loops);
- RCP Breaker Position (Two Loops);
- Undervoltage Bus A01 and A02; and
- Underfrequency Bus A01 and A02.

The low power trips are blocked below the P-7 setpoint and unblocked above the P-7 setpoint. In MODE 2, 3, 4, 5 or 6, this Function does not have to be OPERABLE because the interlock performs its Function when power level drops below 10% power, which is in MODE 1.

D
RAI 3.3.1-14

Power Range Neutron Flux

Power Range Neutron Flux is actuated by two-out-of-four NIS power range channels. The LCO requirement for this Function ensures that this input to the P-7 interlock is available.

The LCO requires four channels of Power Range Neutron Flux to be OPERABLE in MODE 1.

OPERABILITY in MODE 1 ensures the Function is available to perform its increasing power Functions.

Turbine Impulse Pressure

The Turbine Impulse Pressure interlock is actuated when the pressure in the first stage of the high pressure turbine is greater than approximately 10% of the rated full power pressure. This is determined by one-out-of-two pressure detectors. The LCO requirement for this Function ensures that one of the inputs to the P-7 interlock is available.

The LCO requires two channels of Turbine Impulse Pressure interlock to be OPERABLE in MODE 1.

The Turbine Impulse Chamber Pressure interlock must be OPERABLE when the turbine generator is operating. The

BASES

APPLICABLE SAFETY ANALYSES, LCO AND APPLICABILITY (continued)

interlock Function is not required OPERABLE in MODE 2, 3, 4, 5, or 6 because the turbine generator is not operating. D RAI 3.3.1-14

c. Power Range Neutron Flux, P-8

The Power Range Neutron Flux, P-8 interlock is actuated at approximately 50% power as determined by two-out-of-four NIS power range detectors.

The P-8 interlock automatically enables the Reactor Coolant Flow-Low (Single Loop) and RCP Breaker Position (Single Loop) reactor trips on increasing power. The LCO requirement for this trip Function ensures that protection is provided against a loss of flow in any RCS loop that could result in DNB conditions in the core when greater than approximately 50% power. On decreasing power, the reactor trip on low flow in any loop is automatically blocked.

The LCO requires four channels of Power Range Neutron Flux, P-8 interlock to be OPERABLE in MODE 1.

In MODE 1, a loss of flow in one RCS loop could result in DNB conditions, so the Power Range Neutron Flux, P-8 interlock must be OPERABLE. In MODE 2, 3, 4, 5, or 6, this Function does not have to be OPERABLE because the core is not producing sufficient power to be concerned about DNB conditions.

d. Power Range Neutron Flux, P-9 D RAI 3.3.1-14 RAI 3.3.1-27

The Power Range Neutron Flux, P-9 interlock, is actuated at approximately 50% power, as determined by two-out-of-four NIS power range detectors, if the Steam Dump System is available. The LCO requirement for this Function ensures that the Turbine Trip-Low Autostop Oil Pressure and Turbine Trip-Turbine Stop Valve Closure reactor trips are enabled above the P-9 setpoint. Above the P-9 setpoint, a turbine trip will cause a load rejection beyond the capacity of the Steam Dump System. A reactor trip is automatically initiated on a turbine trip when it is above the P-9 setpoint to minimize the transient on the reactor.

The LCO requires four channels of Power Range Neutron Flux, P-9 interlock, to be OPERABLE in MODE 1 with one of two circulating water pump breakers closed and condenser vacuum greater than or equal to 22 "Hg.

BASES

APPLICABLE SAFETY ANALYSES, LCO AND APPLICABILITY (continued)

In MODE 1, a turbine trip could cause a load rejection beyond the capacity of the Steam Dump System, so the Power Range Neutron Flux interlock must be OPERABLE. In MODE 2, 3, 4, 5, or 6, this Function does not have to be OPERABLE because the reactor is not at a power level sufficient to have a load rejection beyond the capacity of the Steam Dump System.

e. Power Range Neutron Flux, P-10

D RAI 3.3.1-14

The Power Range Neutron Flux, P-10 interlock is actuated at approximately 10% power, as determined by two-out-of-four NIS power range detectors. If power level falls below 10% RTP on 3 of 4 channels, the nuclear instrument trips will be automatically unblocked. The LCO requirement for the P-10 interlock ensures that the following Functions are performed:

- on increasing power, the P-10 interlock allows the operator to manually block the Intermediate Range Neutron Flux reactor trip;
- on increasing power, the P-10 interlock allows the operator to manually block the Power Range Neutron Flux-Low reactor trip;
- on increasing power, the P-10 interlock automatically provides a backup signal to block the Source Range Neutron Flux reactor trip, and also to de-energize the NIS source range detectors;

D RAI 3.3.1-14

- on decreasing power, the P-10 interlock automatically enables the Power Range Neutron Flux-Low reactor trip and the Intermediate Range Neutron Flux reactor trip.

The LCO requires four channels of Power Range Neutron Flux, P-10 interlock to be OPERABLE in MODE 1 or 2.

OPERABILITY in MODE 1 ensures the Function is available to perform its decreasing power Functions in the event of a reactor shutdown. This Function must be OPERABLE in MODE 2 to ensure that core protection is provided during a startup or shutdown by the Power Range Neutron Flux-Low and Intermediate Range Neutron Flux reactor trips. In MODE 3, 4, 5, or 6, this Function does not have to be OPERABLE because the reactor is not at power and the Source Range Neutron Flux reactor trip provides core protection.

APPLICABLE SAFETY ANALYSES, LCO AND APPLICABILITY (continued)

18. Reactor Trip Breakers

This trip Function applies to the RTBs exclusive of individual trip mechanisms. The LCO requires two OPERABLE RTBs. Two OPERABLE RTBs ensure no single random failure can disable the RPS trip capability. These trip Functions must be OPERABLE in MODE 1 or 2 when the reactor is critical. In MODE 3, 4, or 5, these RPS trip Functions must be OPERABLE when the RTBs are closed and the Rod Control System is capable of rod withdrawal.

D
RAI 3.3.1-14

19. Reactor Trip Breaker Undervoltage and Shunt Trip Mechanisms

The LCO requires both the Undervoltage and Shunt Trip Mechanisms to be OPERABLE for each RTB that is in service. The trip mechanisms are not required to be OPERABLE for trip breakers that are open, racked out, incapable of supplying power to the Rod Control System, or declared inoperable under Function 18 above. OPERABILITY of both trip mechanisms on each breaker ensures that no single trip mechanism failure will prevent opening any breaker on a valid signal.

These trip Functions must be OPERABLE in MODE 1 or 2 when the reactor is critical. In MODE 3, 4, or 5, these RPS trip Functions must be OPERABLE when the RTBs are closed and the Rod Control System is capable of rod withdrawal.

20. Reactor Trip Bypass Breaker and associated Undervoltage Trip Mechanism

The LCO requires the Reactor Trip Bypass Breaker and its associated Undervoltage Trip Mechanism to be OPERABLE when the Reactor Trip Bypass Breaker is racked in and closed. The bypass breaker and its associated trip mechanism are not required to be OPERABLE when the bypass breaker is open or racked out.

These trip Functions must be OPERABLE in MODE 1 or 2 when a Reactor Trip Bypass Breaker is racked in and closed. In MODE 3, 4, or 5, this RPS trip Function must be OPERABLE when a Reactor Trip Bypass Breaker is racked in and closed and the Rod Control System is capable of rod withdrawal.

BASES

APPLICABLE SAFETY ANALYSES, LCO AND APPLICABILITY (continued)

21. Automatic Trip Logic

The LCO requirement for the RTBs (Functions 18 and 19) and Automatic Trip Logic (Function 21) ensures that means are provided to interrupt the power to allow the rods to fall into the reactor core. Each RTB is equipped with an undervoltage coil and a shunt trip coil to trip the breaker open when needed. Each RTB is equipped with a bypass breaker to allow testing of the trip breaker while the unit is at power. The reactor trip signals generated by the RPS Automatic Trip Logic cause the RTBs and associated bypass breakers to open and shut down the reactor.

The LCO requires two trains of RPS Automatic Trip Logic to be OPERABLE. Having two OPERABLE channels ensures that random failure of a single logic channel will not prevent reactor trip. These trip Functions must be OPERABLE in MODE 1 or 2 when the reactor is critical. In MODE 3, 4, or 5, these RPS trip Functions must be OPERABLE when the RTBs are closed and the Rod Control System is capable of rod withdrawal.

The RPS instrumentation satisfies Criterion 3 of the NRC Policy Statement.

ACTIONS

A Note has been added to the ACTIONS to clarify the application of Completion Time rules. The Conditions of this Specification may be entered independently for each Function listed in Table 3.3.1-1.

In the event a channel's Trip Setpoint is found nonconservative with respect to the Allowable Value, or the transmitter, instrument loop, signal processing electronics, or bistable is found inoperable, then all affected Functions provided by that channel must be declared inoperable and the LCO Condition(s) entered for the protection Function(s) affected.

When the number of inoperable channels in a trip Function exceed those specified in one or other related Conditions associated with a trip Function, then the unit is outside the safety analysis. Therefore, LCO 3.0.3 must be immediately entered if applicable in the current MODE of operation.

A.1

Condition A applies to all RPS protection Functions. Condition A addresses the situation where one or more required channels or trains for one or more Functions are inoperable at the same time. The Required Action is to refer to Table 3.3.1-1 and to take the Required

BASES

ACTIONS (continued)

Actions for the protection functions affected. The Completion Times are those from the referenced Conditions and Required Actions.

B.1 and B.2

Condition B applies to the Manual Reactor Trip in MODE 1 or 2. With one channel inoperable, the inoperable channel must be restored to OPERABLE status within 48 hours. In this condition, the remaining OPERABLE channel is adequate to perform the safety function.

The Completion Time of 48 hours is reasonable considering that there are two automatic actuation trains and another manual initiation channel OPERABLE, and the low probability of an event occurring during this interval. D RAI 3.3.1-1

If the Manual Reactor Trip Function cannot be restored to OPERABLE status within the allowed 48 hour Completion Time, the unit must be brought to a MODE in which the requirement does not apply. To achieve this status, the unit must be brought to at least MODE 3 within 6 additional hours. The 6 additional hours to reach MODE 3 is reasonable, based on operating experience, to reach MODE 3 from full power operation in an orderly manner and without challenging unit systems. With the unit in MODE 3, this trip Function is no longer required to be OPERABLE.

C.1 and C.2

Condition C applies to the Manual Reactor Trip Function in MODE 3, 4, or 5 with the RTBs closed and the Rod Control System capable of rod withdrawal.

With one channel inoperable, the inoperable channel must be restored to OPERABLE status within 48 hours. If the Reactor Manual Trip channel cannot be restored to OPERABLE status within the allowed 48 hour Completion Time, the unit must be placed in a MODE in which the requirement does not apply. To achieve this status, the RTBs must be opened within the next hour. The additional hour provides sufficient time to accomplish the action in an orderly manner. With the RTBs open, the Manual Reactor Trip Function is no longer required. D RAI 3.3.1-1

D.1 and D.2

Condition D applies to the following reactor trip Functions:

- Power Range Neutron Flux-High;

BASES

ACTIONS (continued)

- Power Range Neutron Flux-Low;
- Overtemperature ΔT;
- Overpower ΔT;
- Pressurizer Pressure-High;
- SG Water Level-Low Low; and
- SG Water Level - Low coincident with Steam Flow/Feedwater Flow Mismatch.

D RAI 3.3.1-1

A known inoperable channel must be placed in the tripped condition within 6 hours. Placing the channel in the tripped condition results in a partial trip condition requiring only one-out-of-two logic for actuation of the two-out-of-three trips and one-out-of-three logic for actuation of the two-out-of-four trips.

D RAI 3.3.1-2

If the inoperable channel cannot be placed in the tripped condition within the specified Completion Time, the unit must be placed in a MODE where these Functions are not required OPERABLE. An additional 6 hours is allowed to place the unit in MODE 3. Six hours is a reasonable time, based on operating experience, to place the unit in MODE 3 from full power in an orderly manner and without challenging unit systems.

D RAI 3.3.1-30

E.1 and E.2

D RAI 3.3.1-1 Errata #145

Condition E applies to the Underfrequency Bus A01 and A02 trip function. With one channel inoperable, the inoperable channel must be placed in the tripped condition within 6 hours. Placing the channel in the tripped condition results in a partial trip condition requiring only one additional channel to initiate a reactor trip above the P-7 setpoint. The 6 hours to place the channel in the tripped condition is necessary due to plant design requiring maintenance personnel to effect the trip of the channel outside of the Control Room. An additional 6 hours is allowed to reduce THERMAL POWER to below P-7 if the inoperable channel cannot be restored to OPERABLE status or placed in trip within the specified Completion Time.

Allowance of this time interval takes into consideration the redundant capability provided by the remaining redundant OPERABLE channel and the low probability of occurrence of an event during this period that may require the protection afforded by this trip function.

BASES

ACTIONS (continued) F.1 and F.2

D
Errata #145

Condition F applies to the Intermediate Range Neutron Flux trip when THERMAL POWER is above the P-6 setpoint and below the P-10 setpoint and one channel is inoperable. Above the P-6 setpoint and below the P-10 setpoint, the NIS intermediate range detector performs the monitoring Functions. If THERMAL POWER is greater than the P-6 setpoint but less than the P-10 setpoint, 24 hours is allowed to reduce THERMAL POWER below the P-6 setpoint or increase to THERMAL POWER above the P-10 setpoint. The NIS Intermediate Range Neutron Flux channels must be OPERABLE when the power level is above the capability of the source range, P-6, and below the capability of the power range, P-10. If THERMAL POWER is greater than the P-10 setpoint, the NIS power range detectors perform the monitoring and protection functions and the intermediate range is not required. The Completion Times allow for a slow and controlled power adjustment above P-10 or below P-6 and take into account the redundant capability afforded by the redundant OPERABLE channel, and the low probability of its failure during this period. This action does not require the inoperable channel to be tripped because the Function uses one-out-of-two logic. Tripping one channel would trip the reactor. Thus, the Required Actions specified in this Condition are only applicable when channel failure does not result in reactor trip.

G.1 and G.2

Condition G applies to two inoperable Intermediate Range Neutron Flux trip channels in MODE 2 when THERMAL POWER is above the P-6 setpoint and below the P-10 setpoint. Required Actions specified in this Condition are only applicable when channel failures do not result in reactor trip. Above the P-6 setpoint and below the P-10 setpoint, the NIS intermediate range detector performs the monitoring Functions. With no intermediate range channels OPERABLE, the Required Actions are to suspend operations involving positive reactivity additions immediately. This will preclude any power level increase since there are no OPERABLE Intermediate Range Neutron Flux channels. The operator must also reduce THERMAL POWER below the P-6 setpoint within two hours. Below P-6, the Source Range Neutron Flux channels will be able to monitor the core power level. The Completion Time of 2 hours will allow a slow and controlled power reduction to less than the P-6 setpoint and takes into account the low probability of occurrence of an event during this period that may require the protection afforded by the NIS Intermediate Range Neutron Flux trip.

BASES

ACTIONS (continued)

H.1

D
RAI 3.3.1-1
Errata #145

Condition H applies to one inoperable Source Range Neutron Flux trip channel when in MODE 2, below the P-6 setpoint, and performing a reactor startup. With the unit in this Condition, below P-6, the NIS source range performs the monitoring and protection functions. With one of the two channels inoperable, operations involving positive reactivity additions shall be suspended immediately.

This will preclude any power escalation. With only one source range channel OPERABLE, core protection is severely reduced and any actions that add positive reactivity to the core must be suspended immediately.

I.1

Condition I applies to two inoperable Source Range Neutron Flux trip channels when in MODE 2, below the P-6 setpoint and performing a reactor startup, or in MODE 3, 4, or 5 with the RTBs closed and the Rod Control System capable of rod withdrawal. With the unit in this Condition, below P-6, the NIS source range perform the monitoring and protection functions. With both source range channels inoperable, the RTBs must be opened immediately. With the RTBs open, the core is in a more stable condition.

J.1 and J.2

Condition J applies to one inoperable source range channel in MODE 3, 4, or 5 with the RTBs closed and the Rod Control System capable of rod withdrawal. With the unit in this Condition, below P-6, the NIS source range performs the monitoring and protection functions. With one of the source range channels inoperable, 48 hours is allowed to restore it to an OPERABLE status. If the channel cannot be returned to an OPERABLE status, 1 additional hour is allowed to open the RTBs. Once the RTBs are open, the core is in a more stable condition.

K.1 and K.2

Condition K applies to the following reactor trip Functions:

- Pressurizer Pressure-Low;
- Pressurizer Water Level-High;
- Reactor Coolant Flow-Low (Two Loops);

BASES

ACTIONS (continued)

- Undervoltage Bus A01 and A02; and
- Underfrequency Bus A01 and A02.

RAI 3.3.1-1

With one channel inoperable, the inoperable channel must be placed in the tripped condition within 6 hours. Placing the channel in the tripped condition results in a partial trip condition requiring only one additional channel to initiate a reactor trip above the P-7 interlock and below the P-8 setpoint. These Functions do not have to be OPERABLE below the P-7 interlock because there are no loss of flow trips below the P-7 interlock. An additional 6 hours is allowed to reduce THERMAL POWER to below P-7 if the inoperable channel cannot be restored to OPERABLE status or placed in trip within the specified Completion Time.

RAI 3.3.1-4
RAI 3.3.1-2

RAI 3.3.1-4

Allowance of this time interval takes into consideration the redundant capability provided by the remaining redundant OPERABLE channel, and the low probability of occurrence of an event during this period that may require the protection afforded by the Functions associated with Condition K.

RAI 3.3.1-1
RAI 3.3.1-2
RAI 3.3.1-5
RAI 3.3.1-10
RAI 3.3.1-30
Errata #145

L.1 and L.2

Condition L applies to the Reactor Coolant Flow-Low (Single Loop) reactor trip Function. With one channel inoperable, the inoperable channel must be placed in the tripped condition within 6 hours. If the channel cannot be restored to OPERABLE status or the channel placed in trip within the 6 hours, then THERMAL POWER must be reduced below the P-8 setpoint within the next 4 hours. This places the unit in a MODE where the LCO is no longer applicable. This trip Function does not have to be OPERABLE below the P-8 setpoint because other RPS trip Functions provide core protection below the P-8 setpoint.

M.1 and M.2

Condition M applies to the RCP Breaker Position (Single Loop) reactor trip Function. There is one breaker position device per RCP breaker. With one channel inoperable, the inoperable channel(s) must be restored to OPERABLE status within 1 hour. If the channel cannot be restored to OPERABLE status within the 1 hour, then THERMAL POWER must be reduced below the P-8 setpoint within the next 4 hours.

This places the unit in a MODE where the LCO is no longer applicable. This Function does not have to be OPERABLE below the P-8 setpoint because other RPS Functions provide core protection below the P-8 setpoint.

BASES

ACTIONS (continued)

N.1 and N.2

D
RAI 3.3.1-1
Errata #145

Condition N applies to the RCP Breaker Position (Two Loop) reactor trip Function. With one channel inoperable, the inoperable channel must be restored to OPERABLE status within 1 hour. If the channel cannot be restored to OPERABLE status in 1 hour, then THERMAL POWER must be reduced below the P-7 interlock within the next 6 hours. This places the unit in a MODE where the LCO is no longer applicable. This function does not have to be OPERABLE below the P-7 interlock because there are no loss of flow trips below the P-7 interlock. The Completion Time of 6 hours is reasonable, based on operating experience, to reduce THERMAL POWER to below the P-7 interlock from full power in an orderly manner without challenging unit systems.

D
RAI 3.3.1-1
RAI 3.3.1-10
RAI 3.3.1-8
Errata #145

O.1 and O.2

D
RAI 3.3.1-1
RAI 3.3.1-2
RAI 3.3.1-7
RAI 3.3.1-30
Errata #145

Condition O applies to Turbine Trip on Low Autostop Oil Pressure or on Turbine Stop Valve Closure. With one channel inoperable, the inoperable channel must be placed in the trip condition within 6 hours. If placed in the tripped condition, this results in a partial trip condition requiring only one additional channel to initiate a reactor trip. If the channel cannot be restored to OPERABLE status or placed in the trip condition, then power must be reduced below the P-9 setpoint within the next 4 hours.

P.1 and P.2

Condition P applies to the SI Input from ESFAS reactor trip and the RPS Automatic Trip Logic in MODES 1 and 2. These actions address the train orientation of the RPS for these Functions. With one train inoperable, 6 hours are allowed to restore the train to OPERABLE status (Required Action P.1) or the unit must be placed in MODE 3 within the next 6 hours. The Completion Time of 6 hours (Required Action P.1) is reasonable considering that in this Condition, the remaining OPERABLE train is adequate to perform the safety function and given the low probability of an event during this interval. The Completion Time of 6 hours (Required Action P.2) is reasonable, based on operating experience, to reach MODE 3 from full power in an orderly manner and without challenging unit systems.

The Required Actions have been modified by a Note that allows bypassing one train for up to 8 hours for surveillance testing, provided the other train is OPERABLE.

BASES

ACTIONS (continued)

Q.1 and Q.2

RAI 3.3.1-1 Errata #145

Condition Q applies to the RTBs in MODES 1 and 2. With one RTB inoperable, 1 hour is allowed to restore the RTB to OPERABLE status or the unit must be placed in MODE 3 within the next 6 hours. The Completion Time of 6 hours is reasonable, based on operating experience, to reach MODE 3 from full power in an orderly manner and without challenging unit systems. The 1 hour and 6 hour Completion Times are equal to the time allowed by LCO 3.0.3 for shutdown actions in the event of a complete loss of RPS Function. Placing the unit in MODE 3 removes the requirement for this particular Function.

The Required Actions have been modified by a Note allowing one channel to be bypassed for up to 8 hours provided the other channel is OPERABLE.

R.1 and R.2

RAI 3.3.1-1 Errata #145

Condition R applies to the P-6 interlock (in MODE 2) and the P-10 interlock. With one or more channels inoperable for one-out-of-two or two-out-of-four coincidence logic, the associated interlock must be verified to be in its required state for the existing unit condition within 1 hour or the unit must be placed in MODE 3 within the next 6 hours. Verifying the interlock status manually accomplishes the interlock's Function. The Completion Time of 1 hour is based on operating experience and the minimum amount of time allowed for manual operator actions. The Completion Time of 6 hours is reasonable, based on operating experience, to reach MODE 3 from full power in an orderly manner and without challenging unit systems. The 1 hour and 6 hour Completion Times are equal to the time allowed by LCO 3.0.3 for shutdown actions in the event of a complete loss of RPS Function.

S.1 and S.2

RAI 3.3.1-1 Errata #145

Condition S applies to the P-7, P-8, and P-9 interlocks. With one or more channels inoperable for one-out-of-two or two-out-of-four coincidence logic, the associated interlock must be verified to be in its required state for the existing unit condition within 1 hour or the unit must be placed in MODE 2 within the next 6 hours. These actions are conservative for the case where power level is being raised. Verifying the interlock status manually accomplishes the interlock's Function. The Completion Time of 1 hour is based on operating experience and the minimum amount of time allowed for manual operator actions. The Completion Time of 6 hours is reasonable, based on operating experience, to reach MODE 2 from full power in an orderly manner and without challenging unit systems.

BASES

ACTIONS (continued)

T.1 and T.2

Condition T applies to the RTBs and the RTB Undervoltage and Shunt Trip Mechanisms in MODES 3, 4, or 5 with the RTBs closed and the Rod Control System capable of rod withdrawal.

With one trip mechanism or RTB inoperable, the inoperable trip mechanism or RTB must be restored to OPERABLE status within 48 hours. The Completion Time is reasonable considering that the remaining OPERABLE trip mechanism or RTB is adequate to perform the safety function, and given the low probability of an event occurring during this interval.

If the RTB or trip mechanism cannot be restored to OPERABLE status within 48 hours, the unit must be placed in a MODE in which the requirement does not apply. This is accomplished by opening the RTBs within the next hour (49 hours total time). The Completion Time of 1 hour provides sufficient time to accomplish this action in an orderly manner and takes into account the low probability of an event occurring in this interval.

RAI 3.3.1-1
Errata #145

U.1 and U.2

Condition U applies to the RTB Undervoltage and Shunt Trip Mechanisms, or diverse trip features, in MODES 1 and 2. With one of the diverse trip features inoperable, it must be restored to an OPERABLE status within 48 hours or the unit must be placed in a MODE where the requirement does not apply. This is accomplished by placing the unit in MODE 3 within the next 6 hours (54 hours total time). The Completion Time of 6 hours is a reasonable time, based on operating experience, to reach MODE 3 from full power in an orderly manner and without challenging unit systems.

RAI 3.3.1-1
Errata #145

With the unit in MODE 3, Condition T would apply to any inoperable RTB trip mechanisms. The affected RTB shall not be bypassed while one of the diverse features is inoperable except for the time required to perform maintenance to one of the diverse features. The allowable time for performing maintenance of the diverse features is 8 hours for the reasons stated under Condition Q.

Errata #145

The Completion Time of 48 hours is reasonable considering that in this Condition there is one remaining diverse feature for the affected RTB, and one OPERABLE RTB capable of performing the safety function and given the low probability of an event occurring during this interval.

BASES

ACTIONS (continued)

V.1 and V.2

Condition V applies to the Reactor Trip Bypass Breaker (RTBB) and associated Undervoltage Trip Mechanism in MODE 1 or 2, when the RTBB is racked in and closed. With the required RTBB inoperable, 1 hour is allowed to restore the RTBB to OPERABLE status or the unit must be placed in MODE 3 within the next 6 hours. The Completion Time of 6 hours is reasonable, based on operating experience, to reach MODE 3 from full power in an orderly manner and without challenging unit systems. The 1 hour and 6 hour completion times are equal to the time allowed by LCO 3.0.3 for shutdown action in the event of a complete loss of RPS Function. Placing the unit in MODE 3 removes the requirement for this particular Function.

D
RAI 3.3.1-1
Errata #145

W.1 and W.2

Condition W applies to the Reactor Trip Bypass Breaker (RTBB) and associated Undervoltage Trip Mechanism in MODES 3, 4, or 5, when an RTBB is racked in and closed and the Rod Control System is capable of rod withdrawal. With the required RTBB inoperable, 48 hours is allowed to restore the RTBB to OPERABLE status or the unit must be placed in a MODE in which the requirement does not apply. To achieve this status, the RTBs and RTBBs must be opened within the next 1 hour (49 hours total time). The Completion Time of 1 hour provides sufficient time to accomplish the action in an orderly manner. With the RTBs and RTBBs open, this Function is no longer required.

D
RAI 3.3.1-1
Errata #145

X.1 and X.2

Condition X applies to the RPS Automatic Trip Logic in MODES 3, 4 or 5 with the RTBs closed and the Rod Control System capable of rod withdrawal. With one train inoperable, 48 hours are allowed to restore the train to an OPERABLE status. The Completion Time of 48 hours is reasonable considering that in this condition, the remaining OPERABLE train is adequate to perform the safety function, and given the low probability of an event occurring in this interval.

D
RAI 3.3.1-1
Errata #145

If the RPS Automatic Trip Logic cannot be restored to OPERABLE status within 48 hours, the unit must be placed in a MODE where this Function is not required to be OPERABLE. To achieve this status, the RTBs must be opened within the next 1 hour (49 hours total time). The additional hour provides sufficient time to accomplish the action in an orderly manner. With the RTBs open, the Automatic Trip Logic is no longer required.

BASES

SURVEILLANCE REQUIREMENTS

The SRs for each RPS Function are identified by the SRs column of Table 3.3.1-1 for that Function.

A Note has been added to the SR Table stating that Table 3.3.1-1 determines which SRs apply to which RPS Functions.

Note that each channel of process protection supplies both trains of the RPS. When testing Channel I, Train A and Train B must be examined. Similarly, Train A and Train B must be examined when testing Channel II, Channel III, and Channel IV (if applicable). The CHANNEL CALIBRATION and COTs are performed in a manner that is consistent with the assumptions used in analytically calculating the required channel accuracies.

SR 3.3.1.1

Performance of the CHANNEL CHECK once every 12 hours ensures that gross failure of instrumentation has not occurred. A CHANNEL CHECK is normally a comparison of the parameter indicated on one channel to a similar parameter on other channels. It is based on the assumption that instrument channels monitoring the same parameter should read approximately the same value. Significant deviations between the two instrument channels could be an indication of excessive instrument drift in one of the channels or of something even more serious. A CHANNEL CHECK will detect gross channel failure; thus, it is key to verifying that the instrumentation continues to operate properly between each CHANNEL CALIBRATION.

Agreement criteria are determined by the unit staff based on a combination of the channel instrument uncertainties, including indication and readability. If a channel is outside the criteria, it may be an indication that the sensor or the signal processing equipment has drifted outside its limit.

The Frequency is based on operating experience that demonstrates channel failure is rare. 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.

SR 3.3.1.2

SR 3.3.1.2 compares the calorimetric heat balance calculation to the NIS channel output every 24 hours. If the calorimetric exceeds the NIS channel output by > 2% RTP, the NIS is not declared inoperable, but must be adjusted. If the NIS channel output cannot be properly adjusted, the channel is declared inoperable.

BASES

SURVEILLANCE REQUIREMENTS (continued)

Two Notes modify SR 3.3.1.2. The first Note indicates that the NIS channel output shall be adjusted consistent with the calorimetric results if the absolute difference between the NIS channel output and the calorimetric is > 2% RTP. The second Note clarifies that this Surveillance is required only if reactor power is ≥ 15% RTP and that 12 hour is allowed for performing the first Surveillance after reaching 15% RTP. At lower power levels, calorimetric data are inaccurate. The Frequency of every 24 hours is adequate. It is based on unit operating experience, considering instrument reliability and operating history data for instrument drift. Together these factors demonstrate the change in the absolute difference between NIS and heat balance calculated powers rarely exceeds 2% in any 24 hour period.

In addition, control room operators periodically monitor redundant indications and alarms to detect deviations in channel outputs.

<u>SR 3.3.1.3</u>

SR 3.3.1.3 compares the incore system to the NIS channel output every 31 EFPD. SR 3.3.1.3 is performed by means of the moveable incore detection system. If the absolute difference is ≥ 3%, the NIS channel is still OPERABLE, but must be readjusted. D RAI 3.3.1-18

If the NIS channel cannot be properly readjusted, the channel is declared inoperable. This Surveillance is performed to verify the f(ΔI) input to the overtemperature ΔT Function.

Two Notes modify SR 3.3.1.3. Note 1 indicates that the excore NIS channel shall be adjusted if the absolute difference between the incore and excore AFD is ≥ 3%.

Note 2 clarifies that the Surveillance is required only if reactor power is ≥ 50% RTP and that 24 hours is allowed for performing the first Surveillance after reaching 50% RTP.

The Frequency of every 31 EFPD is adequate. It is based on unit operating experience, considering instrument reliability and operating history data for instrument drift. Also, the slow changes in neutron flux during the fuel cycle can be detected during this interval.

<u>SR 3.3.1.4</u>

SR 3.3.1.4 is the performance of a TADOT every 31 days on a STAGGERED TEST BASIS. This test shall verify OPERABILITY by actuation of the end devices.

The RTB test shall include separate verification of the undervoltage and

BASES

SURVEILLANCE REQUIREMENTS (continued)

shunt trip mechanisms. The independent test for bypass breakers is included in SR 3.3.1.13. The bypass breaker test shall include an undervoltage trip. A Note has been added to SR 3.3.1.4 to indicate that this test must be performed on the bypass breaker prior to placing it in service.

The Frequency of every 31 days on a STAGGERED TEST BASIS is adequate. It is based on industry operating experience, considering instrument reliability and operating history data.

SR 3.3.1.5

SR 3.3.1.5 is the performance of an ACTUATION LOGIC TEST, every 31 days on a STAGGERED TEST BASIS. The train being tested is placed in the bypass condition, thus preventing inadvertent actuation. All possible logic combinations, with and without applicable permissives, are tested for each protection function. The Frequency of every 31 days on a STAGGERED TEST BASIS is adequate. It is based on industry operating experience, considering instrument reliability and operating history data.

SR 3.3.1.5 is modified by two Notes. Note 1 provides an 8 hour delay in the requirement to perform this Surveillance for the Source Range Neutron Flux trip function instrumentation when power is reduced to below P-6. This Note allows a normal shutdown to proceed without a delay for testing in MODE 2 and for a short time in MODE 3 until the RTBs are open and SR 3.3.1.5 is no longer required to be performed. If the unit is to be in MODE 2 below P-6 for > 8 hours, this Surveillance must be performed prior to 8 hours after reducing power below P-6. D RAI 3.3.1-22 RAI TR-2

Note 2 excludes the RCP Breaker Position (Two Loop), Reactor Coolant Flow-Low (Two Loop) and Underfrequency Bus A01 and A02 Trip Functions, and the P-6, P-7, P-8, P-9 and P-10 Interlocks. These functions/interlocks are tested at an 18 month frequency via SR 3.3.1.15.

SR 3.3.1.6

SR 3.3.1.6 is a calibration of the excore channels to the incore channels. If the measurements do not agree, the excore channels are not declared inoperable but must be calibrated to agree with the incore detector measurements. If the excore channels cannot be adjusted, the channels are declared inoperable. This Surveillance is performed to verify the f(ΔI) input to the overtemperature ΔT Function.

A Note modifies SR 3.3.1.6. The Note states that this Surveillance is required only if reactor power is > 50% RTP and that 24 hours is

Insert N

N. One channel inoperable.	N.1	Restore channel to OPERABLE status.	1 hour	D RAI 3.3.1-1 RAI 3.3.1-8 RAI 3.3.1-10 Errata #145
	OR			
	N.2	Reduce THERMAL POWER to < P-7.	7 hours	

SURVEILLANCE REQUIREMENTS (continued)

allowed for performing the first surveillance after reaching 50% RTP.

The Frequency of 92 EFPD is adequate. It is based on industry operating experience, considering instrument reliability and operating history data for instrument drift.

SR 3.3.1.7

SR 3.3.1.7 is the performance of a COT every 92 days.

A COT is performed on each required channel to ensure the entire channel will perform the intended Function.

Setpoints must be within the Allowable Values specified in Table 3.3.1-1.

The difference between the current "as found" values and the previous test "as left" values must be consistent with the drift allowance used in the setpoint methodology. The setpoint shall be left set consistent with the assumptions of the current unit specific setpoint methodology.

The "as found" and "as left" values must also be recorded and verified to be within the required limits.

SR 3.3.1.7 is modified by a Note that provides a 4 hour delay in the requirement to perform this Surveillance for source range instrumentation when entering MODE 3 from MODE 2. This Note allows a normal shutdown to proceed without a delay for testing in MODE 2 and for a short time in MODE 3 until the RTBs are open and SR 3.3.1.7 is no longer required to be performed. If the unit is to be in MODE 3 with the RTBs closed for > 4 hours this Surveillance must be performed prior to 4 hours after entry into MODE 3.

D Errata #155

SR 3.3.1.8

SR 3.3.1.8 is the performance of a COT as described in SR 3.3.1.7, except it is modified by a Note that this test shall include verification that the P-6 and P-10 interlocks are in their required state for the existing unit condition. The Frequency is modified by a Note that allows this surveillance to be satisfied if it has been performed within 92 days of the Frequencies prior to reactor startup and four hours after reducing power below P-10 and P-6. The Frequency of "prior to startup" ensures this surveillance is performed prior to critical operations and applies to the source, intermediate and power range low instrument channels.

D RAI 3.3.1-17

BASES

SURVEILLANCE REQUIREMENTS (continued)

The Frequency of "4 hours after reducing power below P-10" (applicable to intermediate and power range low channels) and "4 hours after reducing power below P-6" (applicable to source range channels) allows a normal shutdown to be completed and the unit removed from the MODE of Applicability for this surveillance without a delay to perform the testing required by this surveillance. The Frequency of every 92 days thereafter applies if the plant remains in the MODE of Applicability after the initial performances of prior to reactor startup and four hours after reducing power below P-10 or P-6. The MODE of Applicability for this surveillance is < P-10 for the power range low and intermediate range channels and < P-6 for the source range channels. Once the unit is in MODE 3, this surveillance is no longer required. If power is to be maintained < P-10 or < P-6 for more than 4 hours, then the testing required by this surveillance must be performed prior to the expiration of the 4 hour limit. Four hours is a reasonable time to complete the required testing or place the unit in a MODE where this surveillance is no longer required. This test ensures that the NIS source, intermediate, and power range low channels are OPERABLE prior to taking the reactor critical and after reducing power into the applicable MODE (< P-10 or < P-6) for periods > 4 hours. D RAI 3.3.1-17

SR 3.3.1.9

SR 3.3.1.9 is the performance of a TADOT and is performed every 31 days. D Errata #155

SR 3.3.1.10

A CHANNEL CALIBRATION is performed every 18 months, or approximately at every refueling. CHANNEL CALIBRATION is a complete check of the instrument loop, including the sensor. The test verifies that the channel responds to a measured parameter within the necessary range and accuracy.

CHANNEL CALIBRATIONS must be performed consistent with the assumptions of the unit specific setpoint methodology. The difference between the current "as found" values and the previous test "as left" values must be consistent with the drift allowance used in the setpoint methodology.

The Frequency of 18 months is based on the assumption of an 18 month calibration interval in the determination of the magnitude of equipment drift in the setpoint methodology.

SR 3.3.1.10 is modified by a Note stating that this test shall include verification that the time delays are adjusted to the prescribed values where applicable.

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.3.1.11

SR 3.3.1.11 is the performance of a CHANNEL CALIBRATION, as described in SR 3.3.1.10, every 18 months. This SR is modified by a Note stating that neutron detectors are excluded from the CHANNEL CALIBRATION. The CHANNEL CALIBRATION for the power range neutron detectors consists of a normalization of the detectors based on a power calorimetric and flux map performed above 15% RTP. The CHANNEL CALIBRATION for the source range and intermediate range neutron detectors consists of obtaining the detector plateau or preamp discriminator curves, evaluating those curves, and comparing the curves to the manufacturer's data. This Surveillance is not required for the NIS power range detectors for entry into MODE 2 or 1, and is not required for the NIS intermediate range detectors for entry into MODE 2, because the unit must be in at least MODE 2 to perform the test for the intermediate range detectors and MODE 1 for the power range detectors. 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 an unplanned transient if the Surveillance were performed with the reactor at power. Operating experience has shown these components usually pass the Surveillance when performed on the 18 month Frequency.

SR 3.3.1.12

SR 3.3.1.12 is the performance of a COT of RPS interlocks every 18 months.

The Frequency is based on the known reliability of the interlocks and the multichannel redundancy available, and has been shown to be acceptable through operating experience.

SR 3.3.1.13

SR 3.3.1.13 is the performance of a TADOT of the Manual Reactor Trip, RCP Breaker Position, SI Input from ESFAS, and the Condenser Pressure-High and Circulating Water Pump Breaker Position inputs to the P-9 Interlock. This TADOT is performed every 18 months. The test shall independently verify the OPERABILITY of the undervoltage and/or shunt trip circuits for the Manual Reactor Trip Function for the Reactor Trip Breakers and Reactor Trip Bypass Breakers. D Errata #101

The Frequency is based on the known reliability of the Functions and the multichannel redundancy available, and has been shown to be acceptable through operating experience.

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.3.1.14

SR 3.3.1.14 is the performance of a TADOT of Turbine Trip Functions. This TADOT is as described in SR 3.3.1.4, except that this test is performed prior to exceeding the P-9 interlock whenever the unit has been in MODE 3. This Surveillance is not required if it has been performed within the previous 31 days. Performance of this test will ensure that the turbine trip Function is OPERABLE prior to exceeding the P-9 interlock.

D RAI 3.3.1-7

D RAI 3.3.1-26 RAI 3.3.1-22

SR 3.3.1.15

SR 3.3.1.15 is the performance of an ACTUATION LOGIC TEST on the RCP Breaker Position (Two Loop), Reactor Coolant Flow-Low (Two Loop) and Underfrequency Bus A01 and A02 Trip Functions, and P-6, P-7, P-8, P-9 and P-10 Interlocks every 18 months.

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 an unplanned transient if the surveillance were performed with the reactor at power.

REFERENCES

1. FSAR, Chapter 7.
2. FSAR, Chapter 14.
3. IEEE-279-1968.
4. 10 CFR 50.49.
5. DG-I01, Instrument Setpoint Methodology.

D RAI 3.3.1-2 RAI 3.3.1-25 RAI 3.3.1-30

JFD Number	JFD Text	
06 Rev. D	Not used.	
	ITS: SPEC 5.02.01.C	**NUREG:** SPEC 5.02.01.C

5.0 ADMINISTRATIVE CONTROLS

5.2 Organization

5.2.1 Onsite and Offsite Organizations

Approved TSTF-65, R1

Onsite and offsite organizations shall be established for unit operation and corporate management, respectively. The onsite and offsite organizations shall include the positions for activities affecting safety of the nuclear power plant.

, including the plant-specific titles of those personnel fulfilling the responsibilities of the positions delineated in these Technical Specifications,

a. Lines of authority, responsibility, and communication shall be defined and established throughout highest management levels, intermediate levels, and all operating organization positions. These relationships shall be documented and updated, as appropriate, in organization charts, functional descriptions of departmental responsibilities and relationships, and job descriptions for key personnel positions, or in equivalent forms of documentation. These requirements shall be documented in the [FSAR]; /QA ~~Plan~~

Plant Manager

b. The ~~[Plant Superintendent]~~ shall be responsible for overall safe operation of the plant and shall have control over those onsite activities necessary for safe operation and maintenance of the plant;

FSAR

A specified corporate officer

c. The ~~[a specified corporate executive position]~~ shall have corporate responsibility for overall plant nuclear safety and shall take any measures needed to ensure acceptable performance of the staff in operating, maintaining, and providing technical support to the plant to ensure nuclear safety; and

01

D

RAI 5.2-1

d. The individuals who train the operating staff, carry out health physics, or perform quality assurance functions may report to the appropriate onsite manager; however, these individuals shall have sufficient organizational freedom to ensure their independence from operating pressures.

Facility

05

5.2.2 ~~Unit~~ Staff

The ~~unit~~ staff organization shall include the following:

a. A non-licensed operator shall be assigned to each reactor containing fuel and an additional non-licensed operator

5.0 ADMINISTRATIVE CONTROLS

5.2 Organization

5.2.1 Onsite and Offsite Organizations

Onsite and offsite organizations shall be established for unit operation and corporate management, respectively. The onsite and offsite organizations shall include the positions for activities affecting safety of the nuclear power plant.

a. Lines of authority, responsibility, and communication shall be defined and established throughout highest management levels, intermediate levels, and all operating organization positions. These relationships shall be documented and updated, as appropriate, in organization charts, functional descriptions of departmental responsibilities and relationships, and job descriptions for key personnel positions, or in equivalent forms of documentation. These requirements, including the plant-specific titles of those personnel fulfilling the responsibilities of the positions delineated in these Technical Specifications, shall be documented in the FSAR;

b. The Plant Manager shall be responsible for overall safe operation of the plant and shall have control over those onsite activities necessary for safe operation and maintenance of the plant;

c. A specified corporate officer shall have corporate responsibility for overall plant nuclear safety and shall take any measures needed to ensure acceptable performance of the staff in operating, maintaining, and providing technical support to the plant to ensure nuclear safety; and D RAI 5.2-1

d. The individuals who train the operating staff, carry out health physics, or perform quality assurance functions may report to the appropriate onsite manager; however, these individuals shall have sufficient organizational freedom to ensure their independence from operating pressures.

Justification For Deviations - NUREG-1431 Section 5.04

30-Jan-01

JFD Number	JFD Text
01 Rev. D	The CTS 15.6.8.1 requirements for plant procedures is being retained in the proposed ITS 5.4.1. The NUREG-1431 item 5.4.1.a requirements for plant procedures (procedures recommended in Reg Guide 1.33) is not being adopted in the proposed ITS. Point Beach is not committed to Regulatory Guide 1.33, Revision 2, Appendix A, February 1978; therefore, it is not part of the current licensing basis. The NUREG-1431 item 5.4.1.b requirements for plant procedures (emergency operating procedures (EOPs) required to implement the requirements of NUREG-0737 and Supplement 1, as stated in Generic Letter 82-33) is being adopted in the proposed ITS. The NUREG-1431 item 5.4.1.c requirements for plant procedures (quality assurance for effluent and environmental monitoring) will be adopted in the proposed ITS. The CTS already contains this requirement in CTS 15.7.8.3. The NUREG-1431 item 5.4.1.d requirements for plant procedures (fire protection program implementation) will be adopted in the proposed ITS and is consistent with CTS 15.6.8.1.8. The NUREG-1431 item 5.4.1.e requirements for plant procedures (all programs specified in specification 5.5) will be adopted in the proposed ITS. **ITS:** SPEC 5.04.01 **NUREG:** SPEC 5.04.01

5.0 ADMINISTRATIVE CONTROLS

5.4 Procedures

5.4.1 Written procedures shall be established, implemented, and maintained covering the following activities:

a. ~~The applicable procedures recommended in Regulatory Guide 1.33, Revision 2, Appendix A, February 1978;~~

b. ~~The emergency operating procedures required to implement the requirements of NUREG-0737 and to NUREG-0737, Supplement 1, as stated in [Generic Letter 82-33];~~

c. ~~Quality assurance for effluent and environmental monitoring;~~

d. ~~Fire Protection Program implementation; and~~

e. ~~All programs specified in Specification 5.5.~~

a. Normal sequences of startup, operation and shutdown of components, systems and overall plant;
b. Refueling;
c. Specific and foreseen potential malfunctions of systems or components including abnormal reactivity changes;
d. Security Plan Implementation;
e. The emergency operating procedures required to implement the requirements of NUREG-0737 and to NUREG-0737, Supplement 1, as stated in Generic Letter 82-33;
f. Nuclear core testing;
g. Surveillance and Testing of safety related equipment;
h. Fire Protection Implementation;
i. Quality Assurance for effluent and environmental monitoring;

D RAI 5.4-1

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5.0 ADMINISTRATIVE CONTROLS

5.4 Procedures

5.4.1 Written procedures shall be established, implemented, and maintained covering the following activities:

a. Normal sequences of startup, operation and shutdown of components, systems and overall plant;
b. Refueling;
c. Specific and foreseen potential malfunctions of systems or components including abnormal reactivity changes;
d. Security Plan Implementation;
e. The emergency operating procedures required to implement the requirements of NUREG-0737 and to NUREG-0737, Supplement 1, as stated in Generic Letter 82-33; D RAI 5.4-1
f. Nuclear core testing;
g. Surveillance and Testing of safety related equipment;
h. Fire Protection Implementation;
i. Quality Assurance for effluent and environmental monitoring;
j. All programs specified in Specification 5.5.

DOC Number	DOC Text
A.01 Rev. A	The information contained in CTS sections 15.3.9, 15.4.10, 15.7.3, 15.7.4, 15.7.5, 15.7.6 and 15.7.7 is not being retained in ITS. This information does not provide any regulatory requirements necessary to protect the public health and safety, but rather states that the requirements previously contained in the above CTS sections were relocated to the Radiological Effluents and Materials Control and Accountability Program Manual (REMCAP). Therefore, deletion of this information is administrative. **CTS:** / **ITS:** 15.03.09 — N/A 15.04.10 — N/A 15.07.03 — N/A 15.07.04 — N/A 15.07.05 — N/A 15.07.06 — N/A 15.07.07 — N/A
A.02 Rev. A	The information contained in CTS 15.7 is not being retained in ITS. This information does not provide any regulatory requirements necessary to protect the public health and safety, but rather states that the RETS do not expand the responsibilities of the licensed operators, and the material contained therein will not be the subject of SRO/RO licensing examinations. Therefore, deletion of this information is administrative. **CTS:** / **ITS:** 15.07 — N/A
A.03 Rev. A	CTS 15.7.8.3.a is revised to reflect the format of the ISTS. The Environmental Manual (EM) will become the ODCM, which will contain the methodology and parameters used in the conduct of the radiological environmental monitoring program. The ODCM will also contain the radiological effluent controls and radiological environmental monitoring activities and descriptions of the information that should be included in the Annual Monitoring Report. **CTS:** / **ITS:** 15.07.08.03.A — SPEC 5.05.01.A SPEC 5.05.01.B

DOC Number	DOC Text
A.04 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:
15.04.02	SPEC 5.05.07
15.04.02 T 15.04.02-01	SPEC 5.05.08 T 5.05.08-01
15.04.02.A	SPEC 5.05.08
15.04.02.A.01	SPEC 5.05.08.a.01
15.04.02.A.02	SPEC 5.05.08.b
15.04.02.A.02.A	SPEC 5.05.08.b.01
15.04.02.A.02.A.01	SPEC 5.05.08.b.01.i
15.04.02.A.02.A.02	SPEC 5.05.08.b.01.ii
15.04.02.A.02.B	SPEC 5.05.08.b.02 SPEC 5.05.08.b.02.i SPEC 5.05.08.b.02.ii SPEC 5.05.08.b.02.iii
15.04.02.A.02.C	SPEC 5.05.08.b.03
15.04.02.A.02.D	SPEC 5.05.08.b.04
15.04.02.A.02.F	SPEC 5.05.08.b.05
15.04.02.A.04	SPEC 5.05.08.d
15.04.02.A.04.A	SPEC 5.05.08.d.01
15.04.02.A.04.B	SPEC 5.05.08.d.02
15.04.02.A.04.C	SPEC 5.05.08.d.03
15.04.02.A.04.D	SPEC 5.05.08.d.04
15.04.02.A.04.E	SPEC 5.05.08.d.05
15.04.02.A.05.A	SPEC 5.05.08.a SPEC 5.05.08.a.02 SPEC 5.05.08.a.03 SPEC 5.05.08.a.04 SPEC 5.05.08.a.05 SPEC 5.05.08.a.06
15.04.02.B	SPEC 5.05.07
15.04.02.B.03	SPEC 5.05.07
15.04.02.B.03.a	SPEC 5.05.07.d
15.04.11.04.a	SPEC 5.05.10.a

Description of Changes - NUREG-1431 Section 5.05

30-Jan-01

DOC Number	DOC Text	
	15.04.11.04.b	SPEC 5.05.10.a
	15.04.11.04.d	SPEC 5.05.10.c
	15.04.16 T 15.04.16-01 FOOTNOTE (a).01	SPEC 5.05.16.01
	15.04.16 T 15.04.16-01 FOOTNOTE (a).02	SPEC 5.05.16.02
	15.04.16 T 15.04.16-01 FOOTNOTE (a).03	SPEC 5.05.16.03
	15.04.16 T 15.04.16-01 FOOTNOTE (a).04	SPEC 5.05.16.04
	15.06.08.04.A.I	SPEC 5.05.03.A
	15.06.08.04.A.II	SPEC 5.05.03.B
	15.06.08.04.A.III	SPEC 5.05.03.C
	15.06.12	SPEC 5.05.15
	15.06.12.A	SPEC 5.05.15.A
	15.06.12.B	SPEC 5.05.15.B
	15.06.12.C	SPEC 5.05.15.C
	15.06.12.D	SPEC 5.05.15.D
	15.06.12.D.01	SPEC 5.05.15.D.01
	15.06.12.D.02	SPEC 5.05.15.D.02
	15.06.12.E	SPEC 5.05.15.E
	15.06.12.F	SPEC 5.05.15.F
	15.07.08.03.A	SPEC 5.05.01.B
	15.07.08.03.B.02	SPEC 5.05.04.C
	15.07.08.03.B.03	SPEC 5.05.04.B
	15.07.08.03.B.04	SPEC 5.05.04.E
	15.07.08.03.B.06	SPEC 5.05.04.G
	15.07.08.03.B.06.a	SPEC 5.05.04.G
	15.07.08.03.B.06.b	SPEC 5.05.04.G
	15.07.08.03.B.06.c	SPEC 5.05.04.G
	15.07.08.03.B.07	SPEC 5.05.04.I
	15.07.08.03.B.08	SPEC 5.05.01.B
	15.07.08.03.C	SPEC 5.05.01.A SPEC 5.05.04.D
	15.07.08.07.B.01.a	SPEC 5.05.01.C.01.i
	15.07.08.07.B.01.b	SPEC 5.05.01.C.01.ii
	15.07.08.07.B.02	SPEC 5.05.01.C.02
	BASES	SPEC 5.05.10.c
	DPR-24 OL 3.I	SPEC 5.05.09
	DPR-24 OL 3.I.01	SPEC 5.05.09.A
	DPR-24 OL 3.I.02	SPEC 5.05.09.B

DOC Number	DOC Text	
	DPR-24 OL 3.I.03	SPEC 5.05.09.C
	DPR-24 OL 3.I.04	SPEC 5.05.09.D
	DPR-24 OL 3.I.05	SPEC 5.05.09.E
	DPR-24 OL 3.I.06	SPEC 5.05.09.F
	DPR-27 OL 3.I	SPEC 5.05.09
	DPR-27 OL 3.I.01	SPEC 5.05.09.A
	DPR-27 OL 3.I.02	SPEC 5.05.09.B
	DPR-27 OL 3.I.03	SPEC 5.05.09.C
	DPR-27 OL 3.I.04	SPEC 5.05.09.D
	DPR-27 OL 3.I.05	SPEC 5.05.09.E
	DPR-27 OL 3.I.06	SPEC 5.05.09.F
	NEW	SPEC 5.05.10

A.05
Rev. A

15.7.8.3, 15.7.8.3.b, 15.7.8.3.c and 15.7.8.7.B have been revised to reflect the concurrent reorganization of the Radiological Effluents and Materials Control and Accountability Program Manual (REMCAP), Environmental Manual (EM), Radiological Environmental Monitoring Program (REMP) and Radiological Effluent Control Program (RECP) into the Offsite Dose Calculation Manual (ODCM), consistent with the recommendation of GL 89-01. The revisions to the CTS are necessary to adopt certain wording preferences or conventions which do not result in technical changes.

CTS:	ITS:
15.07.08.03	SPEC 5.05.04
15.07.08.03.B	SPEC 5.05.04
	SPEC 5.05.04.C
15.07.08.03.B.02	SPEC 5.05.04.C
15.07.08.03.C	SPEC 5.05.01.A
	SPEC 5.05.04.D
15.07.08.07.B	SPEC 5.05.01.C
15.07.08.07.B.01	SPEC 5.05.01.C.01
15.07.08.07.B.03	SPEC 5.05.01.C.03

A.06
Rev. D

CTS 15.7.8.7.B.4 requires all changes regarding explosive gas to be made via the 50.59 process. The Explosive Gas Monitoring Program is contained in the TRM and requires that all changes regarding explosive gas must be made via the 10 CFR 50.59 process. It is unnecessary to state this requirement in Technical Specifications. Therefore, deletion of this statement is administrative in nature.

CTS:	ITS:
15.07.08.07.B.04	N/A

DOC Number	DOC Text	
A.07 Rev. A	CTS 15.6.8.4.A is modified by foot note *, "Post-Accident Coolant Sampling and Post-Accident Containment Atmospheric Sampling Systems" and foot note **, "It is acceptable if the licensee maintains details of the program in plant operation manuals." These footnotes do not establish or relax any requirement and these details are not required in ITS to provide adequate protection of the public health and safety.	
	CTS:	**ITS:**
	15.06.08.04.A	SPEC 5.05.03
	15.06.08.04.A FOOT NOTE *	N/A
	15.06.08.04.A FOOT NOTE **	N/A
A.08 Rev. A	CTS 15.4.16, Table 15.4.16-1, footnotes (a) and (b) are retained in ITS as the requirements of the RCS PIV Leakage Program. These footnotes are being preceded by a statement that the program shall be established to verify the leakage from each RCS PIV is within the limits specified, in accordance with the Event V Order, issued April 20, 1981. This statement does not impose any additional requirements, but rather provides information necessary to apply the specified limits to the RCS PIVs.	
	CTS:	**ITS:**
	NEW	SPEC 5.05.16
A.09 Rev. A	CTS 15.4.2.A.2(e) and associated footnote 1, and 15.4.2.A.5(a) Definitions for F* Distance and F* Tube and associated footnote 2, have not been retained in ITS. These items were applicable only to Westinghouse Model 44 steam generators in Unit 2. According to the footnotes, these requirements, definitions, and repair options are null and void following Unit 2 steam generator replacement. Due to the replacement of the Unit 2 steam generators, these requirements, definitions, and repair options are no longer required to be in the Technical Specifications, and are therefore deleted.	
	CTS:	**ITS:**
	15.04.02.A.02.E	N/A
	15.04.02.A.05.A	N/A
	15.04.02.A.06	SPEC 5.05.08.e
A.10 Rev. A	CTS 15.4.2.A.3 has been modified by replacing reference to CTS 15.4.2.B.1 with a reference to 10 CFR 50.55a(g). CTS 15.4.2.B.1 provided Inservice Inspection requirements, which have been removed from the Technical Specifications, because they are duplicative of the 10 CFR 50.55a(g) requirements.	
	CTS:	**ITS:**
	15.04.02.A.03	SPEC 5.05.08.c

DOC Number	DOC Text
A.11 Rev. A	CTS 15.3.12.2.a states the results of the in-place cold DOP and halogenated hydrocarbon tests on the HEPA and charcoal adsorber banks shall show a "minimum of 99% DOP removal and 99% halogenated hydrocarbon removal." CTS 15.3.12.2.b states the laboratory charcoal adsorbent tests shall show a "minimum of 99% removal of methyl iodide." The requirements of CTS 15.3.12.2.a have been changed to "penetration and system bypass </= 1.0%." The requirement of CTS 15.3.12.2.b has been changed to "methyl iodide penetration </= 1.0%." These revisions do not change the requirements, but rather restate the same requirement in different terms. Therefore, this change is administrative. **CTS:** 15.03.12.02.a — **ITS:** SPEC 5.05.10.a, SPEC 5.05.10.b **CTS:** 15.03.12.02.b — **ITS:** SPEC 5.05.10.c
A.12 Rev. A	CTS 15.4.2 and 15.7.5 provide introductory statements (Applicability / Objectives) which simply state which systems/components are addressed within each section and provide a brief summary of the purpose for each Section. 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:** 15.04.02 APPL — **ITS:** N/A **CTS:** 15.04.02 OBJ — **ITS:** N/A **CTS:** 15.07.05 APPL — **ITS:** N/A **CTS:** 15.07.05 OBJ — **ITS:** N/A
A.13 Rev. A	Editorial changes to CTS 15.4.6.A.6 have been made to clarify the diesel fuel oil testing program. The program will include sampling and testing requirements and acceptance criteria in accordance with applicable ASTM standards. **CTS:** 15.04.06.A.06 — **ITS:** SPEC 5.05.12

DOC Number	DOC Text
LA.01 Rev. A	The information contained in CTS sections 15.7.1 is not being retained in ITS. This information does not provide any regulatory requirements necessary to protect the public health and safety, but provides definitions for frequently used terms in the RETS. The requirements of the RETS were removed from the CTS in Amendments 184/188 and placed in the Radiological Effluents and Materials Control and Accountability Program (REMCAP). In conjunction with the ITS project, the REMCAP is being reorganized to reflect the recommendations of GL 89-01, and will become the Offsite Dose Calculation Manual (ODCM). The information contained in CTS 15.7.1 will be moved to the ODCM. This information is not necessary to adequately describe the actual regulatory requirement and can be moved to other documents without impact on safety. Changes to the ODCM will be controlled by the ODCM process in Section 5 of the proposed ITS. **CTS:** — **ITS:** 15.07.01.A — ODCM 15.07.01.B — ODCM 15.07.01.C — ODCM 15.07.01.D — ODCM
LA.02 Rev. A	The information contained in CTS sections 15.7.8.3.a regarding an annual milk survey is not being retained in ITS. This information will be located in the ODCM. This information is not necessary to adequately protect the public and can be moved to other documents without impact on safety. Changes to the ODCM will be controlled by the ODCM process in Section 5 of the proposed ITS. **CTS:** — **ITS:** 15.07.08.03.A — N/A
LA.03 Rev. A	The information contained in CTS 15.7.8.5 regarding major changes to radioactive liquid, gaseous and solid waste treatment systems is not being retained in ITS. This information will be located in the ODCM. This information is not necessary to adequately protect the public and can be moved to other documents without impact on safety. Changes to the ODCM will be controlled by the ODCM process in Section 5 of the proposed ITS. **CTS:** — **ITS:** 15.07.08.05 — N/A 15.07.08.05.A — N/A 15.07.08.05.B — N/A 15.07.08.05.C — N/A 15.07.08.05.D — N/A 15.07.08.05.E — N/A 15.07.08.05.F — N/A

DOC Number	DOC Text	
LA.04 Rev. A	The information contained in CTS 15.7.8.2 regarding audits of the activities encompassed by the Radioactive Effluent and Materials and Accountability Program (REMCAP) is not being retained in ITS. In conjunction with the ITS project, the REMCAP is being reorganized to reflect the recommendations of GL 89-01, and will become the Offsite Dose Calculation Manual (ODCM). The information contained in CTS 15.7.8.2 will be moved to the ODCM. This information is not necessary to adequately protect the public and can be moved to other documents without impact on safety. Changes to the ODCM will be controlled by the ODCM process in Section 5 of the proposed ITS.	
	CTS:	**ITS:**
	15.07.08.02	N/A
	15.07.08.02.A	N/A
	15.07.08.02.B	N/A
LA.05 Rev. A	The Bases associated with CTS 15.4.2 is not being retained in ITS, but is moved to the FSAR. This information provides details which are not directly pertinent to the actual requirements. Since these details are not necessary to adequately describe the actual regulatory requirement, they can be moved to other documents without impact on safety. Changes to the FSAR are controlled in accordance with the 10 CFR 50.59 process.	
	CTS:	**ITS:**
	BASES	N/A
LA.06 Rev. B	CTS 15.3.12.A, Control Room Emergency Filtration, has been modified by removing the testing requirements of the Control Room Emergency Filtration (CREF) system. The CREF testing requirements will instead be in accordance with the frequencies specified in Regulatory Guide (RG) 1.52, Revision 2, and in accordance with ASTM D3803-1989 and the methodology of ASME N510-1980, Sections 10, 12 and 13, excluding subsections 10.3 and 12.3. Although this change will result in less restrictive testing requirements for the HEPA filters and charcoal adsorbers, Regulatory Guide 1.52 contains methods acceptable to the NRC for implementing the regulations in 10 CFR 50, Appendix A, with regard to the testing criteria for air filtration and adsorption units of ESF atmospheric cleanup systems designed to mitigate the consequences of a postulated accident.	
	CTS:	**ITS:**
	15.03.12.02.a	N/A
	15.03.12.02.b	N/A
	15.04.11.01	SPEC 5.05.10.d
	15.04.11.04.a	N/A
	15.04.11.04.b	N/A
	15.04.11.04.c	N/A
	15.04.11.04.d	N/A
		SPEC 5.05.10.c

Description of Changes - NUREG-1431 Section 5.05

30-Jan-01

DOC Number	DOC Text	
LA.07 Rev. A	The Gas Decay Tank oxygen concentration limit and the required actions if the limit is exceeded are not being retained in ITS. This information will be contained in the Explosive Gas Monitoring Program. This information is not necessary to adequately protect the public and can be moved to other documents without impact on safety. Changes to the Explosive Gas Monitoring Program will be controlled via the 10 CFR 50.59 process.	
	CTS:	**ITS:**
	15.07.05.A	N/A
	15.07.05.A.01	N/A
	15.07.05.A.02	N/A
LA.08 Rev. D	CTS 15.7.8.3 lists regulations and PBNP GDC regarding control of radioactive effluents, control of the release of and processing of waste materials, and the assessment of radioactivity in the environs of PBNP. This list includes PBNP GDC 17, PBNP GDC 70, and GDC 60 of Appendix A to 10 CFR 50. This information (PBNP GDCs) is duplicated in the PBNP FSAR (Section 1.3). PBNP GDC 70 restates GDC 60 of Appendix A to 10 CFR 50. These criteria contain broad standards regarding the associated requirements and may be moved the the FSAR without impact on safety. The FSAR is controlled via the 10CFR 50.59 process. DOC LB.6 contains additional information.	
	CTS:	**ITS:**
	15.07.08.03	N/A
	15.07.08.03.A	N/A

DOC Number	DOC Text
LA.09 Rev. D	The Tendon Surveillance Program of CTS 15.4.4.II is being replaced by the Tendon Surveillance Program of STS 5.5.6. 10 CFR 50.55.a requires facilities to adopt the ASME Section XI, Subsection IWE and IWL programs by September 2001. The details currently contained in CTS 5.4.4.11 will be moved to the Tendon Surveillance Program. These details are also specified by ASME Section XI, as endorsed and required by 10 CFR 50.55.a. Since these regulations apply to PBNP, this change is an administrative relocation of information.

CTS:	ITS:
15.04.04.II	N/A
15.04.04.II.A	N/A
15.04.04.II.B	N/A
15.04.04.II.C	N/A
15.04.04.II.C.01	N/A
15.04.04.II.C.01.A	N/A
15.04.04.II.C.02	N/A
15.04.04.II.C.02.A	N/A
15.04.04.II.C.02.B	N/A
15.04.04.II.C.02.B.I	N/A
15.04.04.II.C.02.B.II	N/A
15.04.04.II.C.02.B.III	N/A
15.04.04.II.C.02.B.IV	N/A
15.04.04.II.C.02.C	N/A
15.04.04.II.C.02.D	N/A
15.04.04.II.C.02.E	N/A
15.04.04.II.C.02.E.01	N/A
15.04.04.II.C.02.E.02	N/A
15.04.04.II.C.02.E.03	N/A
15.04.04.II.D	N/A

DOC Number	DOC Text	
LB.01 Rev. A	CTS 15.7.8.3.d and 15.7.8.7 contain requirements to establish and maintain a Process Control Program (PCP) to assure compliance with 10 CFR Parts 20, 61 and 71. These requirements duplicate current regulations which provide sufficient and appropriate control of these requirements. Therefore, these details are not required to be in the ITS to provide adequate protection of public health and safety. Since this information is contained in 10 CFR Parts 20, 61 and 71, the requirements will continue to be applicable to Point Beach. Therefore, this change is an administrative relocation of information.	
	CTS:	**ITS:**
	15.07.08.03.D	N/A
	15.07.08.07.A	N/A
	15.07.08.07.A.01	N/A
	15.07.08.07.A.02	N/A
	15.07.08.07.A.03	N/A
LB.02 Rev. D	Not used.	
	CTS:	**ITS:**
	N/A	N/A
LB.03 Rev. A	The End Anchorage Concrete Surveillance requirements of CTS 15.4.4.III are not being retained in the ITS. The Inservice Inspection of ASME Code Class 1, Class 2, and Class 3 components are required to be performed in accordance with Section XI of the ASME Boiler and Pressure Vessel Code and applicable Addenda as required by 10 CFR 50, Section 50.55.a(g) modified by Section 50.55.a(b), except where specific relief is granted by the NRC, pursuant to 10 CFR 50, Section 50.55a(g)(6)(i). Therefore, the Inservice Inspection requirements in the CTS are duplicative of the above ASME Section XI requirements and removing these requirements from CTS is an administrative relocation of the information.	
	CTS:	**ITS:**
	15.04.04.III	N/A
	15.04.04.III.A	N/A
	15.04.04.III.B	N/A
	15.04.04.III.C	N/A
	15.04.04.III.C.01	N/A
	15.04.04.III.C.02	N/A
	15.04.04.III.C.03	N/A
	15.04.04.III.C.04	N/A
	15.04.04.III.C.05	N/A
	15.04.04.III.C.06	N/A
	15.04.04.III.D	N/A
	15.04.04.III.E	N/A

DOC Number	DOC Text
LB.04 Rev. A	The Liner Plate examination requirements of CTS 15.4.4.IV are not being retained in the ITS. The Inservice Inspection of ASME Code Class 1, Class 2, and Class 3 components are required to be performed in accordance with Section XI of the ASME Boiler and Pressure Vessel Code and applicable Addenda as required by 10 CFR 50, Section 50.55.a(g) modified by Section 50.55.a(b), except where specific relief is granted by the NRC, pursuant to 10 CFR 50, Section 50.55a(g)(6)(i). Therefore, the Inservice Inspection requirements in the CTS are duplicative of the above ASME Section XI requirements and removing these requirements from CTS is an administrative relocation of the information.

CTS:	ITS:
15.04.04.IV	N/A
15.04.04.IV.A	N/A
15.04.04.IV.A.01	N/A
15.04.04.IV.A.02	N/A
15.04.04.IV.B	N/A
15.04.04.IV.C	N/A
15.04.04.IV.D	N/A
15.04.04.IV.E	N/A

DOC Number	DOC Text
LB.05 Rev. A	The Inservice Inspection requirements of CTS 15.4.2.B, 15.4.2.B.1 and 15.4.2.B.3 are not being retained in the ITS. The Inservice Inspection of ASME Code Class 1, Class 2, and Class 3 components are required to be performed in accordance with Section XI of the ASME Boiler and Pressure Vessel Code and applicable Addenda as required by 10 CFR 50, Section 50.55.a(g) modified by Section 50.55.a(b), except where specific relief is granted by the NRC, pursuant to 10 CFR 50, Section 50.55a(g)(6)(i). Therefore, the Inservice Inspection requirements in the CTS are duplicative of the above ASME Section XI requirements and removing these requirements from CTS is an administrative relocation of the information.

CTS:	ITS:
15.04.02.B	N/A
15.04.02.B.01	N/A
15.04.02.B.01.a	N/A
15.04.02.B.03	N/A

DOC Number	DOC Text
LB.06 Rev. D	CTS 15.7.8.3 lists regulations regarding control of radioactive effluents, control of the release of and processing of waste materials, and the assessment of radioactivity in the environs of PBNP. This list includes 10 CFR 50.34a and 10 CFR 50.36a. This duplicates current regulations, which provide sufficient and appropriate control of these requirements. Therefore, these details are not required to be in ITS to provide adequate protection of public health and safety. Since these requirements continue to apply to PBNP, this change is an administrative relocation of information.

CTS:	ITS:
15.07.08.03	N/A

DOC Number	DOC Text
M.01 Rev. A	CTS 15.6.8.4.A is proposed to be revised by the addition of "radioactive gases, and particulates in" before the words "containment atmosphere and in plant gaseous effluent samples . . " The addition of this text imposes additional requirements on unit operation and is more restrictive. **CTS:** 15.06.08.04.A **ITS:** SPEC 5.05.03
M.02 Rev. A	The CTS has been revised by the addition of a requirement to establish, implement and maintain a Primary Coolant Sources Outside Containment Program. This program is required to provide controls to minimize leakage from those portions of systems outside containment that could contain highly radioactive fluids during a serious transient or accident to levels as low as practical. The program will be required to include preventive maintenance and periodic visual inspection requirements, and integrated leak test requirements for each system. This change imposes additional requirements for unit operation and is more restrictive. **CTS:** NEW **ITS:** SPEC 5.05.02 SPEC 5.05.02.a SPEC 5.05.02.b
M.03 Rev. A	CTS 15.4.11.1 has been revised from requiring the pressure drop test across the combined HEPA filters and charcoal adsorber banks be demonstrated to be < 6 inches of water at "design Flow rate" to "4950 cfm +/- 10%." Stipulating the value of the design flow in the Technical Specifications imposes additional requirements and is therefore more restrictive. **CTS:** 15.04.11.01 **ITS:** SPEC 5.05.10.d
M.04 Rev. A	CTS 15.7.8.3.b.4) has been modified by the addition of a requirement in the Radiological Effluent Program to provide limitations on the functional capability and use of the appropriate portions of the of the liquid and gaseous effluent treatment system. This revision imposes additional requirements on unit operation and is more restrictive. **CTS:** 15.07.08.03.B.05 **ITS:** SPEC 5.05.04.F
M.05 Rev. A	CTS 15.7.8.3.c has been modified by the addition of the following requirements. In addition to the requirements to specify the annual doses to a member of the public from radioactive materials in liquid effluents and radioactivity and radiation from uranium fuel cycle sources released from the facility to unrestricted areas, the ODCM will be required to specify quarterly doses and dose commitments. This revision imposes additional requirements and is more restrictive. **CTS:** 15.07.08.03.C **ITS:** SPEC 5.05.04.D SPEC 5.05.04.J

DOC Number	DOC Text	
M.06 Rev. A	The CTS has been modified by the addition of the requirement to provide limitations on the annual and quarterly air doses resulting from noble gases released in gaseous effluents from the facility to areas beyond the site boundary, conforming to 10 CFR 50, Appendix I. This revision imposes additional requirements and is more restrictive.	
	CTS:	**ITS:**
	NEW	SPEC 5.05.04.H
M.07 Rev. A	The CTS has been revised by the addition of a requirement to establish, implement and maintain a Component Cyclic or Transient Limit Program. This program is required to provide controls to track the FSAR Section 4.1, cyclic and transient occurrences to ensure that components are maintained within design limits. The requirement to establish, implement and maintain a Component Cyclic or Transient Limit Program imposes additional requirements for unit operation and is more restrictive.	
	CTS:	**ITS:**
	NEW	SPEC 5.05.05
M.08 Rev. A	The CTS has been revised by the addition of a requirement to establish, implement and maintain a Reactor Coolant Pump Flywheel Inspection Program. This program is required to provide for the inspection of each reactor coolant pump flywheel per the recommendations of Regulatory Position c.4.b of Regulatory Guide 1.14, Revision 1. However, in lieu of position c.4.b(1) and c.4.b(2), a qualified in-place UT examination over the volume from the inner bore of the flywheel to the circle one-half of the outer radius or a surface examination (MT and PT) of exposed surfaces of the removed flywheels may be conducted at approximately 10 year intervals coinciding with the Inservice Inspection schedule as required by ASME Section XI. The requirement to establish, implement and maintain a Reactor Coolant Pump Flywheel Inspection Program imposes additional requirements for unit operation and is more restrictive.	
	CTS:	**ITS:**
	NEW	SPEC 5.05.06
M.09 Rev. A	CTS 15.4.2.B.3 has been modified by the adoption of a table that indicates the required frequencies for performing inservice testing activities as they relate to the testing frequencies specified in Section XI of the ASME Boiler and Pressure Vessel Code and applicable Addenda. Also, statements requiring the provisions of SR 3.0.2 and SR 3.0.3 to be applicable to the inservice testing activities frequencies have been added to CTS 15.4.2.B.3. These changes impose additional requirements and are therefore more restrictive.	
	CTS:	**ITS:**
	NEW	SPEC 5.05.07.a SPEC 5.05.07.b SPEC 5.05.07.c

Description of Changes - NUREG-1431 Section 5.05

30-Jan-01

DOC Number	DOC Text	
M.10 Rev. A	A statement requiring the provisions of SR 3.0.2 to be applicable to the SG Tube Surveillance Testing Program test frequencies has been added to CTS 15.4.2.A. This change imposes additional requirements and is therefore more restrictive.	
	CTS:	**ITS:**
	NEW	SPEC 5.05.08
M.11 Rev. A	CTS 15.4.11.4.b and 15.4.11.4.c have been revised from requiring the DOP and the halogenated hydrocarbon testing at "design velocity +/- 20%" to "4950 cfm +/- 10%," to stipulate the actual design flowrate of the Control Room Emergency ventilation system. This change imposes additional requirements and is therefore more restrictive.	
	CTS:	**ITS:**
	15.04.11.04.b	SPEC 5.05.10.b
	15.04.11.04.c	SPEC 5.05.10.b
M.12 Rev. A	A statement requiring the provisions of SR 3.0.2 and SR 3.0.3 to be applicable to the Ventilation Filter Test Program test frequencies has been added to CTS 15.4.11. This change imposes additional requirements and is therefore more restrictive.	
	CTS:	**ITS:**
	NEW	SPEC 5.05.10
M.13 Rev. A	CTS 15.7.5 has been modified by the addition of a requirement to establish, implement and maintain an Explosive Gas Monitoring Program. This program is required to provide controls for potentially explosive gas mixtures contained in the on-service Gas Decay Tank. The program will include a limit for oxygen concentration in the on-service Gas Decay Tank and a surveillance program to ensure the limit is maintained. Additionally, the provisions of SR 3.0.2 and SR 3.0.3 will be applicable to the program surveillance frequencies. The requirement to establish, implement and maintain an Explosive Gas Monitoring Program imposes additional requirements and is therefore more restrictive.	
	CTS:	**ITS:**
	NEW	SPEC 5.05.11 SPEC 5.05.11.A

DOC Number	DOC Text
M.14 Rev. A	CTS 15.4.6.A.6 has been modified by specifying the diesel fuel oil program will establish acceptability of new fuel for use by: determining that the fuel has an API gravity or an absolute specific gravity within limits, a flash point and kinematic viscosity within limits for ASTM 2D fuel oil, and by determining the fuel has a clear and bright appearance with proper color; within 31 days of addition of the new fuel oil to storage tanks, the properties of the new fuel oil (other than API or absolute specific gravity, appearance, and flash point and kinematic viscosity) will be verified to be within limits for ASTM 2D fuel oil; and total particulate concentration of the fuel oil shall be < 10 mg/l, when tested every 92 days in accordance with the applicable ASTM standards. Adopting these requirements imposes additional requirements on unit operation and is therefore more restrictive.

CTS:	ITS:
NEW	SPEC 5.05.12.A
	SPEC 5.05.12.A.1
	SPEC 5.05.12.A.2
	SPEC 5.05.12.A.3
	SPEC 5.05.12.B
	SPEC 5.05.12.C

DOC Number	DOC Text
M.15 Rev. A	Two new programs are added in the ITS. These programs are: ITS 5.5.13 Technical Specification (TS) Bases Control ITS 5.5.14 Safety Function Determination Program (SFDP) The TS Bases Control Program is provided to specifically delineate the appropriate methods and reviews necessary for a change to the Technical Specification Bases. The Safety Function Determination Program is included to support implementation of the support system OPERABILITY characteristics of the Technical Specifications. Adopting these programs imposes additional requirements and is therefore more restrictive.

CTS:	**ITS:**
NEW	SPEC 5.05.13
	SPEC 5.05.13.A
	SPEC 5.05.13.B.1
	SPEC 5.05.13.B.2
	SPEC 5.05.13.C
	SPEC 5.05.13.D
	SPEC 5.05.14
	SPEC 5.05.14.01.A
	SPEC 5.05.14.01.B
	SPEC 5.05.14.01.C
	SPEC 5.05.14.01.D
	SPEC 5.05.14.02.A
	SPEC 5.05.14.02.B
	SPEC 5.05.14.02.C

DOC Number	DOC Text
M.16 Rev. A	Included in CTS 15.6.12 are the requirements for the Containment Leakage Rate Testing Program (CLRTP). These requirements will be retained in the proposed ITS in new section 5.5.15, with additional requirements for air lock testing being added. NUREG-1431 SR 3.6.1.1 includes CLRTP acceptance criteria, which mirror those contained in CTS 15.6.12.D. However, these requirements were not adopted in proposed ITS SR 3.6.1.1. Proposed ITS SR 3.6.1.1 simply states "in accordance with the Containment Leakage Rate Testing Program" when describing the CLRTP acceptance criteria. Therefore, the PBNP CLRTP requirements are being added to section 5.5, "Programs and Manuals," of the proposed ITS so that the CLRTP requirements are included in the Technical Specifications. NUREG-1431 SR 3.6.2.1 includes air lock leakage rate acceptance criteria. However, these requirements were not adopted in proposed ITS SR 3.6.2.1. Proposed ITS SR 3.6.2.1 simply states "in accordance with the Containment Leakage Rate Testing Program" when describing the air lock leakage rate acceptance criteria. Therefore, the PBNP air lock leakage rate acceptance criteria is being added to section 5.5.15 (CLRTP requirements) of the proposed ITS so that the requirements are included in the Technical Specifications. This change is more restrictive, since it adds an additional section on CLRTP requirements to proposed ITS section 5.5.

CTS:	**ITS:**
NEW	SPEC 5.05.15.D.03 SPEC 5.05.15.D.03.a SPEC 5.05.15.D.03.b

15.4.4 CONTAINMENT TESTS

Applicability

Applies to containment leakage and structural integrity.

Objective

To verify that potential leakage from the containment and the pre-stressing tendon loads are maintained within acceptable values.

< See 3.6.1 >

Specification

I. Perform required visual examinations and leakage rate testing in accordance with the Containment Leakage Rate Testing Program.

II. TENDON SURVEILLANCE

A. Object

In order to insure containment structural integrity, selected tendons shall be periodically inspected for symptoms of material deterioration or lift-off force reduction. The tendons for inspection shall be randomly but representatively selected from each group for each inspection; however, to develop a history and to correlate the observed data, one tendon from each group shall be physically unchanged. Tendons selected for inspection will consist of five hoop tendons, three vertical tendons located approximately 120° apart, and three dome tendons, one from each of the three dome tendon groups.

B. Frequency

Tendon surveillance shall be conducted at five-year intervals in accordance with the following schedule:*

Unit	Year	Surveillance Required
1	1984	Physical
2	1984	Visual
1	1989	Visual
2	1989	Physical

LA.9

D

RAI 5.5-4

~~*Subsequent five-year interval inspections repeat this pattern.~~

C. Inspections

Tendon surveillance in accordance with 15.4.4.II.B shall consist of either a visual or physical inspection.

(1) Visual Inspection

a. Tendon anchorage assembly hardware of the randomly selected tendons shall be visually examined to the extent practicable without dismantling load bearing components of the anchorage. The immediate concrete area shall be checked visually for indications of abnormal material behavior.

(2) Physical Inspection

a. Tendons which are physically inspected shall first be visually inspected in accordance with C.(1).

b. All tendons which are physically inspected shall be subjected to a lift-off test to monitor their prestressing force.

(i) If the prestressing force of a selected tendon in a group lies above the predicted lower limit, the tendon is considered to be acceptable.

(ii) If the prestressing force of a selected tendon lies between the predicted lower limit and 90% of the predicted lower limit, two tendons, one on each side of the test tendon, shall be checked for their prestressing forces. If the prestressing forces for these tendons are above the predicted lower limit for the tendons, all three tendons shall be restored to the required level of integrity. A single deficiency shall be considered unique and acceptable. If the prestressing force of either of the adjacent tendons falls below the predicted lower limit of the tendon, additional lift-off testing should be done if necessary, so that the cause and extent of such occurrence can be determined and the condition shall be considered an abnormal degradation of the containment structure and the provisions of Specification 15.3.6.E are applicable.

(iii) If the prestressing force of the selected test tendon falls below 90% of the predicted lower limit, the tendon shall be completely detensioned and a determination shall be made as to the cause of the condition. Such a condition shall be considered an abnormal degradation of the containment structure and the provisions of Specification 15.3.6.E are applicable.

(iv) If the average of all measured tendon forces for each group (corrected for average condition) is found to be less than the minimum required prestress level at Anchorage location for that group, the condition should be considered as abnormal degradation of the containment structure and the provisions of 15.3.6.E are applicable. The average minimum design values adjusted for elastic losses are as follows:[6]

Hoop	134.5 ksi
Vertical	140.6 ksi
Dome	137.4 ksi

c. One randomly selected tendon from each group of tendons shall be subjected to complete detensioning in order to identify broken or damaged wires. During the retensioning of the detensioned tendon, simultaneous measurements of elongation and jacking force shall be made at a minimum of two levels of force between the required seating force and zero. During the detensioning and retensioning of the tendons tested, if the elongation corresponding to a specific load differs by more than 5% from that recorded during installation of the tendons, an investigation shall be made to ensure that such discrepancies are not related to wire failures or slippage of wires in anchorages.

d. A tendon wire shall be removed from the one tendon from each group which has been completely detensioned. The wire shall be inspected over its entire length to determine if evidence of corrosion or other deleterious effects are present. Tensile tests shall be made on three samples cut from each removed wire. The samples will be cut from the midsection and each end of the removed wire. Failure of the material to demonstrate the minimum required tensile strength of 240,000 psi shall be considered an abnormal condition of the containment structure and the engineering evaluation provisions of Specification 15.3.6.E.1 are applicable. If an acceptable justification for continued operation cannot be concluded from this evaluation, then the shutdown requirements of Specification 15.3.6.E.1 are applicable.

e. The sheathing filler grease will be sampled and inspected on each physically inspected tendon. The operability of the sheathing filler grease shall be verified by assuring:

1) There are no voids in the filler material in excess of 5% of net duct volume.

2) Complete grease coverage exists for the different parts of the Anchorage system, and
3) The chemical properties of the filler material are within the tolerance limits specified by the manufacturer.

D. Reports

A final report documenting the results of each tendon surveillance will be prepared and maintained as a permanent plant record.

Abnormal conditions observed during testing will be evaluated to determine the effect of such conditions on containment structural integrity. This evaluation should be completed within 30 days of the identification of the condition. Any condition which is determined in this evaluation to have a significant adverse effect on containment structural integrity will be considered an abnormal degradation of the containment structure.

Any abnormal degradation of the containment structure identified during the engineering evaluation of abnormal conditions shall be reported to the Nuclear Regulatory Commission pursuant to the requirements of 10 CFR 50.4 within thirty days of that determination. Other conditions that indicate possible effects on the integrity of two or more tendons shall be reportable in the same manner. Such reports shall include a description of the tendon condition, the condition of the concrete (especially at tendon anchorages), the inspection procedure and the corrective action taken.

III. End Anchorage Concrete Surveillance

A. Specific locations for surveillance will be determined by information obtained from design calculations, as-built end anchorage concrete and prestressing records, observations of the end anchorage concrete during and after prestressing, and results of deformation measurements made during prestressing and the initial structural test.

B. The inspection intervals will be approximately one-half year and one year after the initial structural test and shall be chosen such that the inspection occurs during the warmest and coldest part of the year following the initial structural test.

LB.3

15.7.8 ADMINISTRATIVE CONTROLS

15.7.8.1 DELETED

LA.4

15.7.8.2 ~~Audits~~

A. ~~An audit of the activities encompassed by the Radioactive Effluent and Materials Control and Accountability Program (REMCAP), [the Offsite Dose Calculation Manual (ODCM), the Radiological Effluent Control Program (RECP), the Radiological Environmental Monitoring Program (REMP), and the Process Control Program (PCP)] and its implementing procedures shall be performed utilizing either offsite licensee personnel or a consulting firm.~~

B. ~~The results of the audit above shall be transmitted to the Chief Nuclear Officer and the Chairman of the Offsite Review Committee.~~

15.7.8.3 Plant Operating Procedures and Programs

A.5

The Radioactive Effluent ~~and Materials~~ Control ~~and Accountability~~ Program ~~(REMCAP)~~ shall be established, implemented, and maintained ~~in accordance with the provisions of Technical Specification 15.6.8. REMCAP shall assure that radioactive effluent and waste material from PBNP complies with applicable Federal, State, and burial ground regulations while keeping all exposures to members of the public as low as reasonably achievable (ALARA). This program shall conform to and implement the requirements of PBNP GDC 70 and of 10 CFR 50.34a and 10 CFR 50.36a for the control of radioactive effluents while maintaining doses from these effluents ALARA, shall implement the requirements of 10 CFR 50.34a and General Design Criterion 60 of Appendix A to 10 CFR 50 to suitably control the release and the processing of waste materials, shall conform to the guidance of Appendix I to 10 CFR 50 and PBNP GDC 17 for the assessment of radioactivity in the environs of PBNP~~ and shall include remedial actions to be taken whenever the program limits are exceeded. REMCAP shall be implemented and maintained under the procedures ~~and methodologies specified in the four (4) manuals/programs listed below and supported, as required, by other procedures.~~ Effluent and environmental monitoring shall be addressed in the Quality Assurance Program.

LA.8

LB.6

A.5

The Radiological Effluent Control Program shall be contained in the ODCM.

by

A.5

D

RAI 5.5-1

< See 5.4.1 >

D
RAI 5.5-1

LA.8
A.3 Replace with Insert 5.5-10

a. Environmental Manual (EM) - In order to assure conformance with PBNP GDC 17 and with the guidance of Appendix I to 10 CFR 50, this manual shall contain the Radiological Environmental Monitoring Program (REMP) methodology, parameters, and administrative functions required for monitoring, sampling, analyzing, and reporting of radiation levels and radionuclide concentrations in the environment around PBNP. ~~The REMP shall require an annual milk survey and shall require that radioanalyses to be conducted by a laboratory participating in an Interlaboratory Comparison Program.~~ LA.2

b. Radiological Effluent Control Program ~~Manual (RECM) - The RECM~~ shall contain the ~~Radiological Effluent Control Program (RECP)~~ parameters and methodologies needed in order to assure conformance with 10 CFR 50.36a for the control of effluents and for maintaining doses to the public ALARA and for compliance with 10 CFR 20.1302. The following ~~RECP~~ items are contained in the RECM: A.5

A.5 Radiological Effluent Control Program

A.4 liquid and gaseous / Monitoring / liquid and gaseous

1) radioactive effluent monitoring instrumentation operability, with actions to be taken if operability requirements are not met, and surveillance requirements.
2) radioactive effluent sampling and analyses requirements;
3) limitations on the concentrations of radioactive materials in liquid effluents to the unrestricted area to 10 times the concentration value in Appendix B, Table 2, Column 2 to 10 CFR 20.1001-20.2402; A.5 and with the methodology and parameters in the ODCM
4) determination of the cumulative and projected dose contributions from radioactive effluents for the current calendar quarter and current calendar year in accordance with ODCM methodology and parameters at least every 31 days;
5) assurance that the appropriate portions of the liquid and gaseous effluent treatment system ~~are used~~ whenever the projected doses in a period of 31 days would exceed 2 percent of the 10 CFR 50, Appendix I dose guidelines;

M.4 Limitations on the functional capability and use of

6) limitations to the release rate of radioactive effluent to the atmosphere in order to assure that the following dose rates would not be exceeded at or beyond the side boundary:
   a. 500 mrem/yr., total body, from noble gases,
   b. 3000 mrem/yr., skin, from noble gases,
   c. 1500 mrem/yr., any organ, from I-131, I-133, H-3, and all radioactive particulates with a half-life greater than 8 days;
7) limitations on the annual and quarterly doses to a member of the public at or beyond the site boundary from I-131, I-133, H-3, and radioactive particulates with a half life

Justification For Deviations - NUREG-1431 Section 5.05

30-Jan-01

JFD Number	JFD Text	
04 Rev. a	NUREG-1431, Inservice Testing (IST) Program, has been modified to state that the IST Program provides controls for ASME Code Class 1, 2 and 3 "pumps and valves," in place of "components." 10 CFR 50.55.a(f) provides the regulatory requirements for an IST Program and specifies that ASME Code Class 1, 2 and 3 pumps and valves are the only components covered by an IST Program. 10 CFR 50.55a(g) provides regulatory requirements for an Inservice Inspection (ISI) Program and specifies that ASME Code Class 1, 2 and 3 components are covered by the ISI Program, and that pumps and valves are covered by the IST Program per 10 CFR 50.55.a(f). NUREG-1431 does not include ISI Program requirements as these requirements have been relocated to plant specific documents. Therefore, the components of the IST Program have been clarified to only include pumps and valves.	
	ITS:	**NUREG:**
	SPEC 5.05.07	SPEC 5.05.08
05 Rev. D	NUREG-1431, specification 5.5.6, Pre-Stressed Concrete Containment Tendon Surveillance Program, is being retained as ITS Specification 5.5.17. Succeeding ITS program requirements have been renumbered accordingly. The brackets have been removed and the proper plant specific information has been provided.	
	ITS:	**NUREG:**
	N/A	SPEC 5.05.06
	SPEC 5.05.06	SPEC 5.05.07
	SPEC 5.05.07	SPEC 5.05.08
	SPEC 5.05.08	SPEC 5.05.09
	SPEC 5.05.09	SPEC 5.05.10
	SPEC 5.05.10	SPEC 5.05.11
	SPEC 5.05.11	SPEC 5.05.12
	SPEC 5.05.12	SPEC 5.05.13
	SPEC 5.05.13	SPEC 5.05.14
	SPEC 5.05.13.D	SPEC 5.05.14.D
	SPEC 5.05.14	SPEC 5.05.15

JFD Number	JFD Text	
07 Rev. A	NUREG-1431, specification 5.5.10, Secondary Water Chemistry Program, has been modified by the deletion of the requirement for the program to provide controls to monitor secondary water chemistry to inhibit low pressure turbine disc stress corrosion cracking. This requirement is not a part of the Point Beach current licensing basis.	
	ITS:	**NUREG:**
	SPEC 5.05.09	SPEC 5.05.10
08 Rev. B	NUREG-1431, specification 5.5.11, Ventilation Filter Testing Program (VFTP), has been modified. References to "Engineered Safety Feature (ESF) filter ventilation systems" have been replaced with "Control Room Emergency Filtration System (F-16)," because this is the only filter ventilation system at Point Beach which requires the testing delineated in this specification. This distinction has also resulted in reorganization of the specification.	
	ITS:	**NUREG:**
	SPEC 5.05.10	SPEC 5.05.11
	SPEC 5.05.10.a	SPEC 5.05.11.a
	SPEC 5.05.10.b	SPEC 5.05.11.b
	SPEC 5.05.10.c	SPEC 5.05.11.c
	SPEC 5.05.10.d	SPEC 5.05.11.d
09 Rev. A	NUREG-1431, specification 5.5.11.d requirement to demonstrate the pressure drop across the combined HEPA filters, the prefilters, and the charcoal adsorbers, has been modified by the deletion of the requirement to include "prefilters." The Point Beach current licensing basis (CTS 15.4.11.1) does not require "prefilters" to be included in the overall pressure drop surveillance.	
	ITS:	**NUREG:**
	SPEC 5.05.10.d	SPEC 5.05.11.d

JFD Number	JFD Text	
10 Rev. D	The liquid radwaste requirements of NUREG-1431, specification 5.5.12, Explosive Gas and Storage Tank Radioactivity Monitoring Program, have not been retained in ITS 5.5.11. This reflects the current licensing basis (for PBNP). As currently stated in CTS 15.7.5, "Radioactive effluent release limits have been removed from Technical Specifications and placed in the REMCAP Manual." The requirements associated with Radiological Effluent Technical Specifications were relocated from the Point Beach CTS to the Radiological Effluents and Materials Control and Accountability Program (REMCAP) using the guidance of Generic Letters 89-01 and 95-10. Removal of these requirements was approved in the NRC Safety Evaluation for Amendments 184 (Unit 1) and 188 (Unit 2) dated July 13, 1998. As stated in the NRC SER regarding removal of these requirements, "Acceptable concentrations of explosive gases are actually controlled by other limiting conditions for operation." The specification has been reformatted as appropriate due to this relocation of the radwaste related requirements.	
	ITS:	**NUREG:**
	N/A	SPEC 5.05.12.B
		SPEC 5.05.12.C
	SPEC 5.05.11	SPEC 5.05.12
11 Rev. A	NUREG-1431, specification 5.5.12.a has been modified by the deletion of the requirement for a hydrogen concentration limit in the Gas Decay Tank. Point Beach current licensing basis (CTS 15.7.5) only requires a limit on the concentration of oxygen in the Gas Decay Tank.	
	ITS:	**NUREG:**
	SPEC 5.05.11.A	SPEC 5.05.12.A
12 Rev. A	NUREG-1431, specification 5.5.13, Diesel Fuel Oil Testing Program, has been modified. The requirement to verify the other properties of ASTM 2D fuel oil within 31 days "following" addition of new fuel oil to the storage tanks, has been modified to within 31 days "of" addition of new fuel oil to the storage tanks. This change is necessary due to the configuration of the diesel fuel oil storage system at Point Beach. The Point Beach diesel fuel oil system includes a fill tank in addition to the storage tanks. The new fuel oil is received in the fill tank, where the new fuel oil is sampled and stored until the test results are obtained. Once satisfactory test results are obtained, the fuel oil is transferred to the storage tanks. Therefore, the requirement to verify "the other properties" of ASTM 2D fuel within 31 days "of" addition to the storage tanks is necessary to prevent redundant testing of new fuel oil (tested upon receipt in the fill tank), upon transfer to the storage tanks.	
	ITS:	**NUREG:**
	SPEC 5.05.12.B	SPEC 5.05.13.B

JFD Number	JFD Text
13 Rev. A	NUREG-1431, specification 5.5.13, Diesel Fuel Oil Testing Program, has been revised from requiring the total particulate concentration of the fuel oil to be tested "every 31 days" to "every 92 days", consistent with CTS 15.4.6.A.6. **ITS:** SPEC 5.05.12.C — **NUREG:** SPEC 5.05.13.C
14 Rev. A	NUREG-1431, specification 5.5.13, Diesel Fuel Oil Testing Program, has been revised from requiring the total particulate concentration of the fuel oil to be tested in accordance with "ASTM D-2276, Method A-2 or A-3" to "the applicable ASTM standards." This change is necessary because ASTM D-2276 provides testing requirements for a field monitor system. Point Beach does not utilize a field monitor, but rather uses the laboratory analysis method. Specifying the total particulate concentration of the fuel will be tested in accordance with the applicable ASTM standards is consistent with CTS 15.4.6.A.6. **ITS:** SPEC 5.05.12.C — **NUREG:** SPEC 5.05.13.C
15 Rev. A	NUREG-1431 has been modified by the addition of a Containment Leakage Rate Testing Program, based on the current licensing basis. This additional program is based on CTS 15.6.12, approved exemptions to 10 CFR 50, Appendix J, and the requirements of 10 CFR 50 Appendix J, Option B.

ITS:	NUREG:
SPEC 5.05.15	N/A
SPEC 5.05.15.A	N/A
SPEC 5.05.15.B	N/A
SPEC 5.05.15.C	N/A
SPEC 5.05.15.D	N/A
SPEC 5.05.15.D.01	N/A
SPEC 5.05.15.D.02	N/A
SPEC 5.05.15.D.03	N/A
SPEC 5.05.15.D.03.a	N/A
SPEC 5.05.15.D.03.b	N/A
SPEC 5.05.15.E	N/A
SPEC 5.05.15.F	N/A

Justification For Deviations - NUREG-1431 Section 5.05

30-Jan-01

JFD Number	JFD Text	
16 Rev. A	NUREG-1431 has been modified by the addition of a Reactor Coolant System Pressure Isolation Valve Leakage Program, based on CTS 15.4.16.	
	ITS:	**NUREG:**
	SPEC 5.05.16	N/A
	SPEC 5.05.16.01	N/A
	SPEC 5.05.16.02	N/A
	SPEC 5.05.16.03	N/A
	SPEC 5.05.16.04	N/A
17 Rev. B	NUREG-1431, 5.5.11 has been modified to explicitly state that testing of the Control Room Emergency Filtration System will be in accordance with the methodologies of ANSI N510-1980, Sections 10, 12 and 13, excluding subsections 10.3 and 12.3. As stated in ANSI N510-1980, "It is the intent of this standard (N510) that it be rigorously applied only to systems designated and built to ANSI N509, however, sections of this standard may be used for technical guidance for testing of non-N509 systems." Point Beach's design and construction pre-dated the first issuance of ANSI N509; therefore, PBNP was not built or designated as a N509 system.	
	ITS:	**NUREG:**
	SPEC 5.05.10	SPEC 5.05.11
	SPEC 5.05.10.a	SPEC 5.05.11.a
	SPEC 5.05.10.b	SPEC 5.05.11.b
	SPEC 5.05.10.c	SPEC 5.05.11.c
	SPEC 5.05.10.d	SPEC 5.05.11.d
18 Rev. B	NUREG-1431, 5.5.11.c has been modified to more closely reflect the requirements of ASTM D3803-1989. The inequalities preceding the temperature and relative humidity values under which the charcoal adsorbent laboratory sample analysis is required to be performed have been deleted. Also, the phrase "applying the tolerances of ASTM D3803-1989" has been added to the end of the last sentence of the paragraph.	
	ITS:	**NUREG:**
	SPEC 5.05.10.c	SPEC 5.05.11.c

5.5 Programs and Manuals

5.5.4 Radioactive Effluent Controls Program (continued)

Approved TSTF-258

i. Limitations on the annual and quarterly doses to a member of the public from iodine-131, iodine-133, tritium, and all radionuclides in particulate form with half lives > 8 days in gaseous effluents released from each unit to areas beyond the site boundary, conforming to 10 CFR 50, Appendix I; and

, beyond the site boundary,

j. Limitations on the annual dose or dose commitment to any member of the public due to releases of radioactivity and to radiation from uranium fuel cycle sources, conforming to 40 CFR 190.

The provisions of SR 3.0.2 and SR 3.0.3 are applicable to the Radioactive Effluent Controls Program surveillance frequency.

5.5.5 Component Cyclic or Transient Limit

2 → 4.1

This program provides controls to track the FSAR, Section [ ], cyclic and transient occurrences to ensure that components are maintained within the design limits.

5.5.6 Pre-Stressed Concrete Containment Tendon Surveillance Program

17

This program provides controls for monitoring any tendon degradation in pre-stressed concrete containments, including effectiveness of its corrosion protection medium, to ensure containment structural integrity. The program shall include baseline measurements prior to initial operations. The Tendon Surveillance Program, inspection frequencies, and acceptance criteria shall be in accordance with [Regulatory Guide 1.35, Revision 3, 1989].

5

The provisions of SR 3.0.2 and SR 3.0.3 are applicable to the Tendon Surveillance Program inspection frequencies.

D

RAI 5.5-4

5.5.7 Reactor Coolant Pump Flywheel Inspection Program

5 → 6

This program shall provide for the inspection of each reactor coolant pump flywheel per the recommendations of Regulatory Position c.4.b of Regulatory Guide 1.14, Revision 1, August 1975.

Insert 5.0-04

Approved TSTF-237

Approved TSTF-118, R0

The provisions of SR 3.0.2 and SR 3.0.3 are applicable to the Diesel Fuel Oil Testing Program test frequencies.

5.5 Programs and Manuals

12

of

5.5.13 12 5 Diesel Fuel Oil Testing Program (continued)

2. a flash point and kinematic viscosity within limits for ASTM 2D fuel oil, and

3. a clear and bright appearance with p roper color;

Within 31 days following addition of the new fuel oil to storage tanks verify that the properties of the new fuel oil, other than those addressed in a. above, are within limits for ASTM 2D fuel oil

b. ~~Other properties for ASTM 2D fuel oil are within limits within 31 days following sampling and addition to storage tanks~~; and

c. Total particulate concentration of the fuel oil is ≤ 10 mg/l when tested every ~~31~~ days in accordance with ~~ASTM D-2276~~, ~~Method A-2 or A-3~~.

Approved TSTF-106, R1

14

92 13 14 the applicable ASTM standard

5.5.14 5 13 Technical Specifications (TS) Bases Control Program

This program provides a means for processing changes to the Bases of these Technical Specifications.

a. Changes to the Bases of the TS shall b e made under appropriate administrative controls and reviews.

b. Licensees may make changes to Bases without prior NRC approval provided the changes do not involve either of the following:

1. a change in the TS incorporated in the license; or

2. a change to the updated FSAR or Bases that ~~involves an unreviewed safety question as defined in 10 CFR 50.59~~.

D

c. The Bases Control Program shall contain provisions to ensure that the Bases are maintained consistent with the FSAR.

d. Proposed changes that meet the criteria of Specification 5.5.14b above shall be reviewed and approved by the NRC prior to implementation. Changes to the Bases implemented without prior NRC approval shall be provided to the NRC on a frequency consistent with 10 CFR 50.71(e).

5 13

TSTF-364

requires NRC approval pursuant to 10 CFR 50.59.

5.5 Programs and Manuals

5.5.12 Diesel Fuel Oil Testing Program

A diesel fuel oil testing program to implement required testing of both new fuel oil and stored fuel oil shall be established. The program shall include sampling and testing requirements, and acceptance criteria, all in accordance with applicable ASTM Standards. The purpose of the program is to establish the following:

a. Acceptability of new fuel oil for use prior to addition to storage tanks by determining that the fuel oil has:

   1. an API gravity or an absolute specific gravity within limits,

   2. a flash point and kinematic viscosity within limits for ASTM 2D fuel oil, and

   3. a clear and bright appearance with proper color;

b. Within 31 days of addition of the new fuel oil to storage tanks verify that the properties of the new fuel oil, other than those addressed in a. above, are within limits for ASTM 2D fuel oil; and

c. Total particulate concentration of the fuel oil is ≤ 10 mg/l when tested every 92 days in accordance with the applicable ASTM standard.

d. The provisions of SR 3.0.2 and SR 3.0.3 are applicable to the Diesel Fuel Oil Testing Program test frequencies.

5.5.13 Technical Specifications (TS) Bases Control Program

This program provides a means for processing changes to the Bases of these Technical Specifications.

a. Changes to the Bases of the TS shall be made under appropriate administrative controls and reviews.

b. Licensees may make changes to Bases without prior NRC approval provided the changes do not involve either of the following:

   1. a change in the TS incorporated in the license; or

   2. a change to the updated FSAR or Bases that requires NRC approval pursuant to 10 CFR 50.59. D

5.5 Programs and Manuals

5.5.16 Reactor Coolant System (RCS) Pressure Isolation Valve (PIV) Leakage Program

A program shall be established to verify the leakage from each RCS PIV is within the limits specified below, in accordance with the Event V Order, issued April 20, 1981.

a. Minimum differential test pressure shall not be less than 150 psid.

b. Leakage rate acceptance criteria are:

1. Leakage rates less than or equal to 1.0 gpm are considered acceptable.

2. Leakage rates greater than 1.0 gpm but less than or equal to 5.0 gpm are considered acceptable if the latest measured rate has not exceeded the rate determined by the previous test by an amount that reduces the margin between measured leakage rate and the maximum permissible rate of 5.0 gpm by 50% or greater.

3. Leakage rates greater than 1.0 gpm but less than or equal to 5.0 gpm are considered unacceptable if the latest measured rate exceeded the rate determined by the previous test by an amount that reduces the margin between measured leakage rate and the maximum permissible rate of 5.0 gpm by 50% or greater.

4. Leakage rates greater than 5.0 gpm are considered unacceptable.

5.5.17 Pre-Stressed Concrete Containment Tendon Surveillance Program

This program provides controls for monitoring any tendon degradation in pre-stressed concrete containments, including effectiveness of its corrosion protection medium, to ensure containment structural integrity. The program shall include baseline measurements prior to initial operations. The Tendon Surveillance Program, inspection frequencies, and acceptance criteria shall be in accordance with Regulatory Guide 1.35, Revision 3, 1989. D RAI 5.5-4

The provisions of SR 3.0.2 and SR 3.0.3 are applicable to the Tendon Surveillance Program inspection frequencies.

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:**
	15.04.04.II.D	SPEC 5.06.07
	15.06.09	SPEC 5.06
	15.06.09.01.B	N/A
	15.06.09.01.B.01	SPEC 5.06
	15.06.09.01.B.02.A	SPEC 5.06.08
	15.06.09.02	N/A
	15.07.08.04	SPEC 5.06.02
A.02 Rev. A	Included in CTS 15.4.2.A.7 are the reporting requirements for steam generator tube inservice inspection results. These reporting requirements will be retained in proposed ITS 5.6.8 (NUREG-1431 item 5.6.10). NUREG-1431, item 5.6.10, "Steam Generator Tube Inspector Report," has a bracketed reviewer's note stating "Reports required by the Licensee's current licensing basis regarding steam generator tube surveillance requirements shall be included here. An appropriate administrative controls format should be used." Therefore, the CTS requirements have simply been transferred to proposed ITS 5.6.8, with certain wording preferences changed to reflect the nomenclature and numbering used in the ITS. The term "Annual Results and Data Report" has been changed to "a Report", because the "Annual Results and Data Report" is not being maintained in the proposed ITS. This change is administrative.	
	CTS:	**ITS:**
	15.04.02.A.07	SPEC 5.06.08
A.03 Rev. A	CTS 15.06.09.01.B.02.B describes an attribute of the "Annual Results and Data Report". This attribute's content is being included in the "Occupational Radiation Exposure Report" in proposed ITS section 5.6.1. Minor administrative wording changes to this attribute were necessary to reflect the wording in NUREG-1431. This change is administrative.	
	CTS:	**ITS:**
	15.06.09.01.B.02.B	SPEC 5.06.01

Description of Changes - NUREG-1431 Section 5.06

30-Jan-01

DOC Number	DOC Text	
A.04 Rev. A	Included in CTS 15.7.8.4.A are the requirements for the contents of the "Annual Monitoring Report (AMR)." CTS 15.7.8.4.A.1 contains the requirements for a subsection of the AMR referred to as the "Radiological Effluent Control Program (RECP)." The required contents of this subsection are consistent with the required contents of NUREG-1431, item 5.6.3 "Radioactive Effluent Release Report," which is proposed for adoption in whole as ITS 5.6.2 "Annual Monitoring Report." This change is administrative.	
	CTS:	**ITS:**
	15.07.08.04.A.01	SPEC 5.06.02
A.05 Rev. A	Included in CTS 15.7.8.4.A are the requirements for the contents of the "Annual Monitoring Report. (AMR)." CTS 15.7.8.4.A.3 contains the requirements for a subsection of the AMR referred to as the "Radiological Effluent Monitoring Program (REMP)." The required contents of this subsection are consistent with the required contents of NUREG-1431, item 5.6.2 "Annual Radiological Environmental Operating Report," which is proposed for adoption in whole as ITS 5.6.2 "Annual Monitoring Report." This change is administrative.	
	CTS:	**ITS:**
	15.07.08.04.A.03	SPEC 5.06.02
A.06 Rev. D	Not used.	
	CTS:	**ITS:**
	N/A	N/A
A.07 Rev. D	Not used.	
	CTS:	**ITS:**
	N/A	N/A

DOC Number	DOC Text	
L.01 Rev. A	CTS 15.6.9.2.E states "if a main steam line radiation monitor (SA-11) fails and cannot be restored to operability in seven days, prepare a special report within thirty days of the event, outlining the action taken, the cause of the inoperability and the plans and schedule for restoring the channel to operable status." This requirement is not contained in NUREG-1431 and will not be retained in the proposed ITS. During the conversion of NUREG-1431 section 3.3.3 (Post Accident Monitoring Instrumentation) to the proposed ITS for PBNP, operability requirements for the steam line radiation monitor (SA-11) were not retained. This was based on the fact that this monitor is not identified as Type A or Category I in the PBNP Regulatory Guide 1.97 analyses, and therefore does not need to be included in the ITS. Not retaining this variable was less restrictive, but is acceptable because it does not result in a reduction in the margin of safety. Therefore, because this monitor was not retained in proposed ITS 3.3.3, the report requirements for its inoperability will not be retained in proposed ITS 5.6. This change is less restrictive. **CTS:** 15.06.09.02.E	 **ITS:** N/A
L.02 Rev. A	CTS 15.7.8.4.2 contains new and spent fuel receipts and shipment reporting requirements as a subsection of the "Annual Monitoring Report (AMR)." This attribute will not be retained in the proposed ITS. This reporting attribute is not contained in NUREG-1431. This reporting attribute was contained in the original Technical Specifications for PBNP (early 1970's). This administrative reporting requirement is not required to be in the ITS to provide adequate protection to the public health and safety. A search of the current licensing basis (CLB) database did not reveal any regulatory commitments associated with this reporting criteria. Therefore, this requirement will not be retained in the proposed ITS. **CTS:** 15.07.08.04.A.02	 **ITS:** N/A
L.03 Rev. A	CTS 15.7.8.4.5 contains administrative requirements for having meteorological data kept on file on site as a subsection of the "Annual Monitoring Report (AMR)." This attribute will not be retained in the proposed ITS. This administrative requirement is not contained in NUREG-1431. This administrative requirement is not required to be in the ITS to provide adequate protection to the public health and safety. A search of the current licensing basis (CLB) database did not reveal any regulatory commitments associated with this administrative requirement. Therefore, this requirement will not be retained in the proposed ITS. **CTS:** 15.07.08.04.A.05	 **ITS:** N/A

DOC Number	DOC Text	
L.04 Rev. A	CTS 15.6.9.2.C states "In the event the low temperature overpressure protection (LTOP) system (power operated relief valves in the low temperature overpressure protection mode) or residual heat removal system relief valves are operated to relieve a pressure transient which, by licensee's evaluation, could have resulted in an overpressurization incident had the system not been operable, a special report shall be prepared and submitted to the Commission within 30 days. The report shall describe the circumstances initiating the transient, the effect of the system on the transient and any corrective action necessary to prevent recurrence." This reporting requirement is a plant specific requirement that is not included in NUREG-1431 and will not be retained in the proposed ITS. This reporting requirement was added in a Wisconsin Electric (WE) license amendment application (TSCR 56) dated 11/02/78, which was an application for adding numerous Technical Specification requirements for "overpressure mitigating system operation." This application was approved by the NRC via License Amendment No. 45 (DPR-24) and No. 50 (DPR-27), and SER dated 5/20/80. This reporting requirement was voluntarily added by WE probably because of the high number of overpressurization events that were occurring in the industry at that time (prior to the industry adding overpressurization protection). The NRC did not mention the addition of this reporting requirement in the SER dated 5/20/80; therefore, it was not used by the NRC as a basis for the approval of the license amendment. This reporting requirement is not required to provide adequate protection to the public health and safety. Any occurrence of a reactor vessel overpressurization event at PBNP will be evaluated in accordance with the applicable regulations, 10 CFR 50.72 and 10 CFR 50.73, and reported to the NRC if appropriate. It should be noted that the NUREG-1431 (STS) requirement in item 5.6.4, to include documentation of all challenges to the pressurizer power operated relief valves or pressurizer safety valves in the monthly operating reports, was deleted from the STS in accordance with NRC approved TSTF-258, revision 4. The basis for this deletion (in TSTF-258) was that these attributes were not included in the guidance contained in NRC Generic Letter 97-02, "Revised Contents of the Monthly Operating Report".	
	CTS:	**ITS:**
	15.06.09.02.C	N/A
L.05 Rev. D	CTS 15.06.09.01.B.02.D states that the Annual Results and Data Report shall include a tabulation of all challenges to the pressurizer power operated relief valves or pressurizer safety valves. This attribute will not be retained in the proposed ITS because the NUREG-1431 (STS) requirement in item 5.6.4, to include documentation of all challenges to the pressurizer power operated relief valves or pressurizer safety valves in the monthly operating reports, was deleted from the STS in accordance with NRC approved TSTF-258, revision 4. The basis for this deletion (in TSTF-258) was that these attributes were not included in the guidance contained in NRC Generic Letter 97-02, "Revised Contents of the Monthly Operating Report". Therefore, this reporting requirement will not be retained in the proposed ITS.	
	CTS:	**ITS:**
	15.06.09.01.B.02.D	N/A

DOC Number	DOC Text
L.06 Rev. D	CTS 15.6.9.1.C contains details that are to be included in the Monthly Operating Report. These unnecessary details are being deleted from the Technical Specifications and will be relocated to licensee control. NUREG-1431 item 5.6.3 contains the necessary details that are to be included in the monthly operating report, and the proposed ITS will adopt these in whole. On May 15, 1997, the NRC issued Generic Letter 97-02, Revised Contents of the Monthly Operating Report. The purpose of this generic letter was to inform licensees that the NRC is requesting the submittal of less information in the monthly operating report. The impetus for the monthly operating report came from the 1973-1974 oil embargo. Draft Regulatory Guide 1.16, Revision 4, "Reporting of Operating Information - Appendix A Technical Specifications," published for comment in August 1975, identifies operating statistics and shutdown experience information then desired in the operating report. The NRC initially compiled this information on a monthly basis and published it in hard copy form as NUREG-0020, "Licensed Operating Reactors - Status Summary Report" (referred to as the "Gray Book"). NUREG-0020 was discontinued after the December 1995 report. GL 97-02 stated that, effective immediately, licensees of operating nuclear power plants submitting monthly operating reports called for in the Technical Specifications may do so in accordance with the guidance provided in Attachment 1 to this generic letter. CTS 15.6.9.1.c does not specify much of the information that GL 97-02 lists as being desired by the NRC to be reported. The proposed ITS, which is consistent with NUREG-1431, would provide the flexibility to report the specific data that the NRC needs to perform its assessment functions. The submittal date is changed to be consistent with NUREG-1431. These additional details that are currently in the CTS, are not required 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. Therefore, the level of safety is unaffected by the change.

CTS:	ITS:
15.06.09.01.C.01	SPEC 5.06.03
15.06.09.01.C.02	SPEC 5.06.03

DOC Number	DOC Text	
L.07 Rev. D	CTS 15.6.9.2.B contains reporting requirements to notify the NRC within 14 days of any plans for removal of any poison assemblies from the spent fuel storage racks. This reporting requirement is being deleted from the Technical Specifications. This requirement was added as a result of License Amendment No. 35 (DPR-24) and No. 41 (DPR-27), approved by the NRC via SER dated April 4, 1979, which allowed for re-racking of the spent fuel pool (to increase available size). This reporting requirement is a plant specific requirement that is not included in NUREG-1431. This reporting requirement included in the CTS is not required to be in the ITS. Any removal of poison assemblies from the spent fuel storage racks would be a change to the facility as described in the FSAR. Therefore, such an activity would be governed by the requirements of 10 CFR 50.59. If this process resulted in the need for a license amendment, the NRC would be notified well in advance via a license amendment request in accordance with 10 CFR 50.90. This reporting requirement is not required to provide adequate protection to the public health and safety. Changes to the plant are subject to controls imposed by 10 CFR 50.59. Therefore, the level of safety is unaffected by the change.	
	CTS:	**ITS:**
	15.06.09.02.B	N/A
LA.01 Rev. D	Not used.	
	CTS:	**ITS:**
	N/A	N/A
LA.02 Rev. D	Not used.	
	CTS:	**ITS:**
	N/A	N/A

DOC Number	DOC Text
LA.03 Rev. A	Wisconsin Electric Power Company has concluded that CTS 15.4.4.II.D, 15.6.10, and 15.7.8.6 can be relocated to licensee control. The basis for this conclusion is as follows: CTS section 15.4.4.II.D, 15.6.10, and 15.7.8.6 describes the administrative requirements for records and records retention. These requirements will not be maintained in the proposed ITS and will be relocated to licensee control. NUREG-1431 does not contain these administrative requirements. PBNP proposes to relocate the requirements in these CTS sections to Section 1.4, "Quality Assurance Program," of the PBNP Final Safety Analysis Report (FSAR). This section of the FSAR describes the PBNP Quality Assurance Program (QAP) in detail. Changes to this section of the FSAR are controlled in accordance with the requirements of 10 CFR 50.54(a). Relocating these requirements out of the CTS and into FSAR §1.4 is consistent with the guidance contained in NRC Administrative Letter (AL) 95-06, "Relocation of Technical Specification Administrative Controls Related to Quality Assurance," dated December 12, 1995. NRC AL 95-06 states that the NRC encourages relocation of the record and record retention requirements out of the licensee's Technical Specifications and into QAP descriptions as long as future revisions to said functions are controlled in accordance with the requirements of 10 CFR 50.54. Conclusion: Based on the above information, the requirements in CTS 15.4.4.II.D, 15.6.10, and 15.7.8.6 can be relocated to licensee control. **CTS:** / **ITS:** 15.04.04.II.D — N/A 15.06.10 — N/A 15.07.08.06 — N/A
LA.04 Rev. A	CTS 15.7.8.4.A.7, and CTS 15.7.8.4.B through D contain numerous details and requirements for the radioactive effluent control program. These requirements will not be maintained in the proposed ITS and will be relocated to licensee control. NUREG-1431 does not contain these administrative requirements. This detail is not required to be in the technical specifications to provide adequate protection to the public health and safety. These CTS requirements are already contained in the Point Beach Offsite Dose Calculation Manual (ODCM) and/or its implementing procedures. Changes to these requirements will be controlled in accordance with the 10 CFR 50.59 process. Therefore, the level of safety is unaffected by the change. **CTS:** / **ITS:** 15.07.08.04.A.07 — N/A 15.07.08.04.B — N/A 15.07.08.04.C — N/A 15.07.08.04.D — N/A

DOC Number	DOC Text
LB.01 Rev. A	CTS 15.6.9.1.A requires submission of a summary report to the NRC of plant startup and power escalation testing under the following conditions: a. Receipt of an operating license, b. Amendment to the license involving a planned increase in power level, c. Installation of fuel that has a different design or has been manufactured by a different fuel supplier, and d. Modifications that may have significantly altered the nuclear, thermal, or hydraulic performance of the plant. These reporting requirements will not be maintained in the proposed ITS, and are not included in NUREG-1431. The conditions describing the above reporting requirements are either no longer applicable or would already require NRC review and approval under the applicable CFRs as follows. Item 15.6.9.1.A.a is no longer applicable since the receipt of the operating licenses for both units occurred in the early 1970s. Item 15.6.9.1.A.b would require submission of a license amendment request package to the NRC, in accordance with the requirements of 10 CFR 50.4 and 10 CFR 50.90, and review and approval from the NRC via a license amendment (the licensed power level is included in the Technical Specifications). Similarly, item 15.6.9.1.A.c would require submission of a license amendment request package to the NRC, in accordance with the requirements of 10 CFR 50.4 and 10 CFR 50.90, and review and approval from the NRC via a license amendment (fuel design is included in the Technical Specifications). Item 15.6.9.1.A.d (modifications to the plant) is covered under the requirements of 10 CFR 50.59, and would require NRC review and approval if a unreviewed safety question (USQ) would exist as a result of the proposed modification. Based on the above, this change removes CTS details that are duplicative of other regulatory requirements, or that are no longer applicable to PBNP. **CTS:** 15.06.09.01.A **ITS:** N/A
LB.02 Rev. A	CTS 15.6.9.1.B.2.C states that the Annual Results and Data Report shall include a description of facility changes, tests or experiments as required pursuant to 10 CFR 50.59(b). This attribute will not be maintained in the proposed ITS. This item is not included in NUREG-1431 because it is duplicative of code requirements. 10 CFR 50.59(b)(2) requires that a report be submitted annually in accordance with 10 CFR 50.4 containing a brief description of any changes, tests, and experiments, including a summary of the safety evaluation of each. Therefore, CTS 15.6.9.1.B.2.C is duplicative of code requirements and will not be retained. **CTS:** 15.06.09.01.B.02.C **ITS:** N/A

DOC Number	DOC Text	
LB.03 Rev. A	Included in CTS 15.7.8.4.A are the requirements for the contents of the "Annual Monitoring Report (AMR)." CTS 15.7.8.4.A.4 contains reporting requirements (as a subsection of the AMR) for leak testing of sealed sources if the tests reveal the presence of 0.005 microcuries or more of removable contamination. This attribute will not be retained in the proposed ITS. This item is not included in NUREG-1431 because it is duplicative of code requirements (reporting requirements contained in 10 CFR 31.5). Therefore, CTS 15.7.8.4.A is duplicative of code requirements and will not be retained.	
	CTS:	**ITS:**
	15.07.08.04.A.04	N/A
LB.04 Rev. D	CTS 15.7.8.4.A.6 states the following, "A description of changes to the REMCAP, ODCM, EM, RECM or PCP which were implemented and became effective during the reporting period shall be submitted pursuant to Specification 15.7.8.7." This requirement is duplicative of the requirements contained in CTS 15.7.8.7, which are being deleted for duplication with regulations (see Section 5.5, DOC LB.01). Therefore, this requirement will not be retained in the proposed ITS. Change requirements for the ODCM are described in proposed ITS 5.5.1.	
	CTS:	**ITS:**
	15.07.08.04.A.06	SPEC 5.05.01
M.01 Rev. A	CTS 15.6.9.1.B.2.E requires the submission (as part of the Annual Results and Data Report) of the results of specific activity analysis only when the primary coolant exceeds the limits of CTS 15.3.1.C "Maximum Coolant Activity", and lists the specific information that is to be included in CTS 15.6.9.1.B.2.E.1 through CTS 15.6.9.1.B.2.E.5. This relaxation is not included in NUREG-1431 and will not be retained in the proposed ITS. This relaxation was added to the PBNP CTS as a result of NRC Generic Letter 85-19. The NRC determined (in GL 85-19) that the reporting requirements for iodine spiking could be reduced from short-term reports (Special Reports or Licensee Event Reports) to a report item that could be included in the annual report and encouraged licensees to amend their Technical Specifications to include this requirement. This was due to the fact that poor fuel reliability in the late 1970's and early 1980's lead to increased primary coolant activity levels, which lead to high volumes of special reports and LERs by licensees. PBNP has concluded that this reporting requirement relaxation is no longer required, due to the extremely low incidence of fuel cladding failures at Point Beach and hence extremely low probability of PBNP exceeding reactor coolant activity limits. Any incidences of high reactor coolant activity at PBNP will be reported to the NRC in accordance with the requirements of 10 CFR 50.72 and 10 CFR 50.73, as appropriate. Because the CTS requirement to report high reactor coolant activity in the annual report is a relaxation of the existing potential reportability requirements in 10 CFR 50.72 and 10 CFR 50.73, elimination of this relaxation is more restrictive.	
	CTS:	**ITS:**
	15.06.09.01.B.02.E	N/A

DOC Number	DOC Text	
M.02 Rev. A	CTS 15.6.9.2.D requires that a special report be submitted to the NRC within 30 days if the minimum number of channels for the containment high-range radiation monitor is not restored within the allowed outage time. Similarly, proposed ITS LCO 3.3.3, Condition B requires submission of a report in accordance with ITS 5.6.6 (NUREG-1431 item 5.6.8) if the minimum number of channels for the containment high-range radiation monitor is not restored within the allowed outage time. The report requirements are similar in that they require outlining the preplanned alternate method of monitoring, the cause of the inoperability, and the plans and schedule for restoring the instrumentation channels of the Function to OPERABLE status. The proposed ITS 5.6.6 will adopt NUREG item 5.6.8 in whole. However, the ITS is more restrictive in that ITS 5.6.6 requires a report to be submitted within 14 days, whereas the CTS requirement is 30 days. Therefore, this change is more restrictive.	
	CTS:	**ITS:**
	15.06.09.02.D	SPEC 5.06.06
M.03 Rev. A	NUREG-1431 item 5.6.5 and proposed ITS 5.6.4 requires that a Core Operating Limits Report (COLR) be established that documents the core operating limits prior to each reload cycle, and specifies the required contents of this report. This COLR report is not required by the CTS, therefore adopting this requirement is more restrictive.	
	CTS:	**ITS:**
	NEW	SPEC 5.06.04
M.04 Rev. A	NUREG-1431 item 5.6.6 and proposed ITS 5.6.5 requires that a Reactor Coolant System (RCS) Pressure and Temperature Limits Report (PTLR) be established that documents the RCS pressure and temperature limits, and specifies the required contents of this report. This PTLR is not required by the CTS, therefore adopting this requirement is more restrictive.	
	CTS:	**ITS:**
	NEW	SPEC 5.06.05
M.05 Rev. A	NUREG-1431 item 5.6.8 and proposed ITS 5.6.6 requires that a Post Accident Monitoring (PAM) Instrumentation report be submitted within 14 days as required by condition B or G of (proposed ITS) LCO 3.3.3. Condition B or G of LCO 3.3.3 basically requires submission of a PAM report when any of the PAM instrumentation listed in TABLE 3.3.3-1 is inoperable for greater than 30 days. This requirement is more restrictive than the CTS, because TABLE 3.3.3-1 includes PAM instrumentation that is not currently required to be reported under the CTS special reporting requirements. Therefore, adopting this requirement is more restrictive.	
	CTS:	**ITS:**
	NEW	SPEC 5.06.06

LB.01

a. 90 days following completion of the startup tests.

b. 90 days following resumption or commencement of commercial power operation.

c. 9 months following initial criticality.

B. Annual Results and Data Report

See ITS 5.6

1. A results and data report covering the period of the previous calendar year shall be submitted prior to March 1 of each year.

2. This report shall include:

See ITS 5.6.8

a. Complete results of steam generator tube inservice inspection completed during the calendar year as required by specification 15.4.2.A.7

b. A tabulation on an annual basis of the number of station, utility, and other personnel receiving exposures greater than 100 mrem/year and their associated man-rem exposure according to work and job functions. The dose assignments to various duty functions may be estimates based on pocket dosimeter, TLD or film badge measurements. Small exposures totalling less than 20 percent of the individual total dose need not be accounted for. In the aggregate, at least 80% of the total whole body dose received from external sources shall be assigned to specific major work functions.

LB.02

c. A description of facility changes, tests or experiments as required pursuant to 10 CFR 50.59(b).

L.05

d. A tabulation of all challenges to the pressurizer power operated relief valves or pressurizer safety valves.

D

RAI 5.6-4

Occupational Radiation Exposure Report

A tabulation on an annual basis of the number of station, utility, and other personnel (including contractors), for whom monitoring was performed, receiving an annual deep dose equivalent > 100 mrems and the associated collective deep dose equivalent (reported in person – rem) according to work and job functions (e.g., reactor operations and surveillance, inservice inspection, routine maintenance, special maintenance, waste processing, and refueling). This tabulation supplements the requirements of 10 CFR 20.2206. The dose assignments to various duty functions may be estimated based on pocket ionization chamber, thermoluminescence dosimeter (TLD), electronic dosimeter, or film badge measurements. Small exposures totaling < 20 percent of the individual total dose need not be accounted for. In the aggregate, at least 80 percent of the total deep dose equivalent received from external sources should be assigned to specific major work functions. The report covering the previous calendar year shall be submitted by April 30 of each year.

A.03

e. Reactor coolant activity

M.01

The results of specific activity analysis in which the primary coolant exceeded the limits of Specification 15.3.1.C. The following information shall be included:

1. Reactor power history starting 48 hours prior to the first sample in which the activity limit was exceeded;
2. Results of the last isotopic analysis for radioiodine analysis prior to exceeding the limit, results of analysis while limit was exceeded and results of one analysis after the radioiodine activity was reduced to less than the limit. Each result should include the date and time of sampling and the radioiodine concentrations;
3. Clean-up flow history starting 48 hours prior to the first sample in which the activity limit was exceeded,
4. Graph of the I-131 concentration and one other radioiodine isotope concentration in microcuries per gram as a function of time for the duration of the specific activity above the steady state level; and
5. The time duration when the specific activity of the primary coolant exceeded 0.8 microcuries per gram DOSE EQUIVALENT I-131.

C. Monthly Operating Reports

L.06

ITS 5.6.3

1. Routine reports of operating statistics and shutdown experience shall be submitted on a monthly basis ~~under the titles "Operating Data Report", "Average Daily Power Levels" and "Unit Shutdowns" and "Power Reduction". In addition, the report shall contain a narrative summary of operating experience that describes the operation of the facility, including major safety-related maintenance for the monthly report period.~~

D

RAI 5.6-5

2. ~~Completed reports shall be sent~~ by the tenth of each month following the calendar month covered by the report.

15th

D

15.6.9.2 Unique Reporting Requirements

The following written reports shall be submitted to the Director, Office of Nuclear Reactor Regulation, USNRC.

A. Deleted

B. Poison Assembly Removal From Spent Fuel Storage Racks

L.07

Plans for removal of any poison assemblies from the spent fuel storage racks shall be reported and described at least 14 days prior to the planned activity. Such report shall describe neutron attenuation testing for any replacement poison assemblies, if applicable, to confirm the presence of boron material.

D

RAI 5.6-5

C. Low Temperature Overpressure Protection System Operation

L.04

In the event the low temperature overpressure protection system (power operated relief valves in the low temperature overpressure protection mode) or residual heat removal system relief valves are operated to relieve a pressure transient which, by licensee's evaluation, could have resulted in an overpressurization incident had the system not been operable, a special report shall be prepared and submitted to the Commission within 30 days. The report shall describe the circumstances initiating the transient, the effect of the system on the transient and any corrective action necessary to prevent recurrence.

LB.04

6. REMCAP Changes

A description of changes to the REMCAP, ODCM, ~~EM,~~ RECM or PCP which were implemented and became effective during the reporting period shall be submitted pursuant to Specification 15.7.8.7.

D

RAI 5.6-3

7. Special Circumstance Reports

a. In accordance with note 7 to RECM Table 3-2, if the Waste Gas Holdup System Explosive Gas Monitor is out of service for greater than 14 days.

b. In accordance with the EM, factors which render the LLDs unachievable.

c. In accordance with the EM, failure of the analytical laboratory to participate in an Interlaboratory Comparison Program.

B. Measured Radioactivity Above Notification Levels

If the confirmed level of radioactivity remains above the notification levels specified in the EM, a written report describing the circumstance shall be prepared and submitted within thirty days of the confirmation that a notification level was exceeded.

LA.04

C. Radioactive Liquid Effluent Treatment

If the radioactive liquid or gaseous effluent treatment system is inoperable and liquid or gaseous effluents are being discharged for 31 days without the treatment required to meet the release limits specified in the RECM, a special report shall be prepared and submitted to the Commission within thirty days which includes the following information:

1. Identification of the inoperable equipment or subsystem and the reason for inoperability.
2. Actions taken to restore the inoperable equipment to operable status.
3. Summary description of actions taken to prevent a recurrence.

D. Radioactive Effluent Releases

If the quantity of radioactive material actually released in liquid or gaseous effluents during any calendar quarter exceeds twice the quarterly limit as specified in the RECM, a special report shall be prepared and submitted to the Commission within thirty days of determination of the release quantity.

Justification For Deviations - NUREG-1431 Section 5.06

30-Jan-01

JFD Number	JFD Text	
08 Rev. A	The reviewer notes in NUREG-1431 item 5.6.6 have not been adopted in the proposed ITS.	
	ITS:	**NUREG:**
	SPEC 5.06.05	SPEC 5.06.06
09 Rev. A	Included in CTS 15.4.4.II.D are the requirements for containment tendon surveillance reports. These current licensing basis requirements will be retained in whole in the proposed ITS 5.6.7, and the requirements in NUREG-1431 item 5.6.9 will not be adopted.	
	ITS:	**NUREG:**
	SPEC 5.06.07	SPEC 5.06.09
10 Rev. D	Not used.	
	ITS:	**NUREG:**
	N/A	N/A

5.6 Reporting Requirements

Approved
TSTF-258, R4

5.6.4 Monthly Operating Reports

Routine reports of operating statistics and shutdown experience~~[including documentation of all challenges to the pressurizer power operated relief valves or pressurizer safety valves.]~~ shall be submitted on a monthly basis no later than the 15th of each month following the calendar month covered by the report. D

5.6.5 CORE OPERATING LIMITS REPORT (COLR)

a. Core operating limits shall be established prior to each reload cycle, or prior to any remaining portion of a reload cycle, and shall be documented in the COLR for the following:

Insert 5.0-08

[The individual specifications that address core operating limits must be referenced here.]

b. The analytical methods used to determine the core operating limits shall be those previously reviewed and approved by the NRC, specifically those described in the following documents:

Insert 5.0-09

[Identify the Topical Report(s) by number, title, date, and NRC staff approval document, or identify the staff Safety Evaluation Report for a plant specific methodology by NRC letter and date.]

c. The core operating limits shall be determined such that all applicable limits (e.g., fuel thermal mechanical limits, core thermal hydraulic limits, Emergency Core Cooling Systems (ECCS) limits, nuclear limits such as SDM, transient analysis limits, and accident analysis limits) of the safety analysis are met.

d. The COLR, including any midcycle revisions or supplements, shall be provided upon issuance for each reload cycle to the NRC.

5.6.6 Reactor Coolant System (RCS) PRESSURE AND TEMPERATURE LIMITS REPORT (PTLR)

a. RCS pressure and temperature limits for heat up, cooldown, low temperature operation, criticality, and hydrostatic

Section 5.0 Inserts

Insert 5.0-03:

A tabulation on an annual basis of the number of station, utility, and other personnel (including contractors), for whom monitoring was performed, receiving an annual deep dose equivalent > 100 mrems and the associated collective deep dose equivalent (reported in person - rem) according to work and job functions (e.g., reactor operations and surveillance, inservice inspection, routine maintenance, special maintenance (describe maintenance), waste processing, and refueling). This tabulation supplements the requirements of 10 CFR 20.2206. The dose assignments to various duty functions may be estimated based on pocket ionization chamber, thermoluminescence dosimeter (TLD), electronic dosimeter, or film badge measurements. Small exposures totaling < 20 percent of the individual total dose need not be accounted for. In the aggregate, at least 80 percent of the total deep dose equivalent received from external sources should be assigned to specific major work functions. The report covering the previous calendar year shall be submitted by April 30 of each year.

D RAI 5.6-1

Insert 5.0-11:

(1) WCAP-14040-NP-A, "Methodology Used to Develop Cold Overpressure Mitigating System Setpoints and RCS Heatup and Cooldown Limit Curves", Revision 2, January 1996

(2) NRC Letter, "Point Beach Nuclear Plant, Units 1 and 2 - Exemption from the Requirements of 10CFR50.60 (TAC NOS. MA9680 and MA 9681)," dated October 6, 2000

(3) USNRC Regulatory Guide 1.99 Rev. 2

D

No Significant Hazards Considerations - NUREG-1431 Section 5.06

30-Jan-01

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 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 an administrative reporting requirement for the main steam line radiation monitor (SA-11). During the conversion of NUREG-1431 section 3.3.3 (Post Accident Monitoring Instrumentation) to the proposed ITS for PBNP, operability requirements for the steam line radiation monitor (SA-11) were not retained. This was based on the fact that this monitor is not identified as Type A or Category I in the PBNP Regulatory Guide 1.97 analyses, and therefore does not need to be included in the ITS. Therefore, because this monitor was not retained in proposed ITS 3.3.3, the report requirements for its inoperability will not be retained in proposed ITS 5.6. Deleting administrative special report requirements 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 of deleting administrative reporting requirements 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? Deleting administrative reporting requirements has no bearing on any margin of safety. 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 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 an administrative reporting requirement for new and spent fuel receipts and shipments. The deletion of an administrative reporting requirement does not involve a significant increase in the probability of an accident previously evaluated because no such accidents are affected by the proposed revision. The proposed change does not introduce any new accident initiators since no accidents previously evaluated have as their initiators anything related to this administrative reporting requirement. All Limiting Conditions of Operation, Limiting Safety System Settings, and Safety Limits specified in the TS remain unchanged. Therefore, this change does not increase the probability of previously evaluated accidents. This change does not involve a significant increase in the consequences of an accident previously evaluated because the source term, containment isolation or radiological releases are not being changed by the proposed change. Existing system and component redundancy and operation is not being changed by this proposed change. The assumptions used in evaluating the radiological consequences in the PBNP Final Safety Analysis Report are not invalidated; therefore, this change does not affect the consequences of previously evaluated accidents. 2. Does the change create the possibility of a new or different kind of accident from any accident previously evaluated? The proposed change of deleting administrative reporting requirements 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. The design and design basis of the facility remain unchanged. The plant safety analyses remain unchanged. 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? Deleting administrative reporting requirements has no bearing on any margin of safety. Accordingly, the proposed change does not involve 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 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 administrative requirements for having meteorological data kept on file on site as a subsection of the "Annual Monitoring Report (AMR)." The deletion of an administrative reporting requirement does not involve a significant increase in the probability of an accident previously evaluated because no such accidents are affected by the proposed revision. The proposed change does not introduce any new accident initiators since no accidents previously evaluated have as their initiators anything related to this administrative requirement. All Limiting Conditions of Operation, Limiting Safety System Settings, and Safety Limits specified in the TS remain unchanged. Therefore, this change does not increase the probability of previously evaluated accidents. This change does not involve a significant increase in the consequences of an accident previously evaluated because the source term, containment isolation or radiological releases are not being changed by the proposed change. Existing system and component redundancy and operation is not being changed by this proposed change. The assumptions used in evaluating the radiological consequences in the PBNP Final Safety Analysis Report are not invalidated; therefore, this change does not affect the consequences of previously evaluated accidents. 2. Does the change create the possibility of a new or different kind of accident from any accident previously evaluated? The proposed change of deleting this administrative requirement 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. The design and design basis of the facility remain unchanged. The plant safety analyses remain unchanged. 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? Deleting this administrative requirement has no bearing on any margin of safety. Accordingly, the proposed change does not involve a significant reduction in a margin of safety.

NSHC Number	NSHC Text
L.04 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 results in the deletion of an administrative reporting requirement for low temperature overpressure protection system operation. The deletion of an administrative reporting requirement does not involve a significant increase in the probability of an accident previously evaluated because no such accidents are affected by the proposed revision. The proposed change does not introduce any new accident initiators since no accidents previously evaluated have as their initiators anything related to this administrative reporting requirement. All Limiting Conditions of Operation, Limiting Safety System Settings, and Safety Limits specified in the TS remain unchanged. Therefore, this change does not increase the probability of previously evaluated accidents. This change does not involve a significant increase in the consequences of an accident previously evaluated because the source term, containment isolation or radiological releases are not being changed by the proposed change. Existing system and component redundancy and operation is not being changed by this proposed change. The assumptions used in evaluating the radiological consequences in the PBNP Final Safety Analysis Report are not invalidated; therefore, this change does not affect the consequences of previously evaluated accidents. 2. Does the change create the possibility of a new or different kind of accident from any accident previously evaluated? The proposed change of deleting administrative reporting requirements 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. The design and design basis of the facility remain unchanged. The plant safety analyses remain unchanged. 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? Deleting administrative reporting requirements has no bearing on any margin of safety. Accordingly, the proposed change does not involve a significant reduction in a margin of safety.

NSHC Number	NSHC Text
L.05 Rev. D	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 an administrative reporting requirement for challenges to pressurizer PORVs and safety valves. The deletion of an administrative reporting requirement does not involve a significant increase in the probability of an accident previously evaluated because no such accidents are affected by the proposed revision. The proposed change does not introduce any new accident initiators since no accidents previously evaluated have as their initiators anything related to this administrative reporting requirement. All Limiting Conditions of Operation, Limiting Safety System Settings, and Safety Limits specified in the TS remain unchanged. Therefore, this change does not increase the probability of previously evaluated accidents. This change does not involve a significant increase in the consequences of an accident previously evaluated because the source term, containment isolation or radiological releases are not being changed by the proposed change. Existing system and component redundancy and operation is not being changed by this proposed change. The assumptions used in evaluating the radiological consequences in the PBNP Final Safety Analysis Report are not invalidated; therefore, this change does not affect the consequences of previously evaluated accidents. 2. Does the change create the possibility of a new or different kind of accident from any accident previously evaluated? The proposed change of deleting administrative reporting requirements 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. The design and design basis of the facility remain unchanged. The plant safety analyses remain unchanged. 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? Deleting administrative reporting requirements has no bearing on any margin of safety. Accordingly, the proposed change does not involve a significant reduction in a margin of safety.

NSHC Number	NSHC Text
L.06 Rev. D	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 specific details in an administrative reporting requirement for the Monthly Operating Report (the requirement for the Monthly Operating Report will remain). The deletion of details from an administrative reporting requirement does not involve a significant increase in the probability of an accident previously evaluated because no such accidents are affected by the proposed revision. The proposed change does not introduce any new accident initiators since no accidents previously evaluated have as their initiators anything related to this administrative reporting requirement. All Limiting Conditions of Operation, Limiting Safety System Settings, and Safety Limits specified in the TS remain unchanged. Therefore, this change does not increase the probability of previously evaluated accidents. This change does not involve a significant increase in the consequences of an accident previously evaluated because the source term, containment isolation or radiological releases are not being changed by the proposed change. Existing system and component redundancy and operation is not being changed by this proposed change. The assumptions used in evaluating the radiological consequences in the PBNP Final Safety Analysis Report are not invalidated; therefore, this change does not affect the consequences of previously evaluated accidents. 2. Does the change create the possibility of a new or different kind of accident from any accident previously evaluated? The proposed change of deleting details from administrative reporting requirements 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. The design and design basis of the facility remain unchanged. The plant safety analyses remain unchanged. 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? Deleting drtails from administrative reporting requirements has no bearing on any margin of safety. Accordingly, the proposed change does not involve a significant reduction in a margin of safety.

NSHC Number	NSHC Text
L.07 Rev. D	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 an administrative reporting requirement for plans for removal of any poison assemblies from the spent fuel storage racks. The deletion of an administrative reporting requirement does not involve a significant increase in the probability of an accident previously evaluated because no such accidents are affected by the proposed revision. The proposed change does not introduce any new accident initiators since no accidents previously evaluated have as their initiators anything related to this administrative reporting requirement. All Limiting Conditions of Operation, Limiting Safety System Settings, and Safety Limits specified in the TS remain unchanged. Therefore, this change does not increase the probability of previously evaluated accidents. This change does not involve a significant increase in the consequences of an accident previously evaluated because the source term, containment isolation or radiological releases are not being changed by the proposed change. Existing system and component redundancy and operation is not being changed by this proposed change. The assumptions used in evaluating the radiological consequences in the PBNP Final Safety Analysis Report are not invalidated; therefore, this change does not affect the consequences of previously evaluated accidents. 2. Does the change create the possibility of a new or different kind of accident from any accident previously evaluated? The proposed change of deleting administrative reporting requirements 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. The design and design basis of the facility remain unchanged. The plant safety analyses remain unchanged. 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? Deleting administrative reporting requirements has no bearing on any margin of safety. Accordingly, the proposed change does not involve a significant 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.

NSHC Number	NSHC Text
LB 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 involves deletion of a Specifications/information which is duplicative of information contained in the Code of Federal Regulations (CFRs). This information is more appropriately addressed by the CFRs and serves no purpose in the Technical Specifications. Deletion of this information will not result in an increase in the probability of an accident. Regulatory requirements do not alter plant design or configuration; therefore, this does not alter any event precursor. Accordingly, there will be no effect on the consequences of any accident. 2. Does the change create the possibility of a new or different kind of accident from any accident previously evaluated? The proposed change does not 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 deletes materials from the Technical Specifications which are adequately addressed in the CFRs. 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 deletes materials from the Technical Specifications which are duplicative of requirements contained in the CFRs. These items are not an input to any accident analysis and, therefore, have no impact on margin of safety.

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 do not affect any 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.

5.0 ADMINISTRATIVE CONTROLS

5.6 Reporting Requirements

The following reports shall be submitted in accordance with 10 CFR 50.4.

5.6.1 Occupational Radiation Exposure Report

--------------------------------------NOTE--
A single submittal may be made that combines sections common to Units 1 and 2.

A tabulation on an annual basis of the number of station, utility, and other personnel (including contractors), for whom monitoring was performed, receiving an annual deep dose equivalent > 100 mrems and the associated collective deep dose equivalent (reported in person – rem) according to work and job functions (e.g., reactor operations and surveillance, inservice inspection, routine maintenance, special maintenance (describe maintenance), waste processing, and refueling). D RAI 5.6-1 This tabulation supplements the requirements of 10 CFR 20.2206. The dose assignments to various duty functions may be estimated based on pocket ionization chamber, thermoluminescence dosimeter (TLD), electronic dosimeter, or film badge measurements. Small exposures totaling < 20 percent of the individual total dose need not be accounted for. In the aggregate, at least 80 percent of the total deep dose equivalent received from external sources should be assigned to specific major work functions. The report covering the previous calendar year shall be submitted by April 30 of each year.

5.6.2 Annual Monitoring Report

--------------------------------------NOTE--
A single submittal may be made that combines sections common to Units 1 and 2.

The Annual Monitoring Report covering the operation of the units during the previous calendar year shall be submitted by April 30 of each year. The report shall include summaries, interpretations, and analyses of trends of the results of the radiological environmental monitoring program for the reporting period. The material provided shall be consistent with the objectives outlined in the Offsite Dose Calculation Manual (ODCM), and in 10 CFR 50, Appendix I, Sections IV.B.2, IV.B.3, and IV.C.

5.6 Reporting Requirements

5.6.4 CORE OPERATING LIMITS REPORT (COLR) (continued)

c. The core operating limits shall be determined such that all applicable limits (e.g., fuel thermal mechanical limits, core thermal hydraulic limits, Emergency Core Cooling Systems (ECCS) limits, nuclear limits such as SDM, transient analysis limits, and accident analysis limits) of the safety analysis are met.

d. The COLR, including any midcycle revisions or supplements, shall be provided upon issuance for each reload cycle to the NRC.

5.6.5 Reactor Coolant System (RCS) PRESSURE AND TEMPERATURE LIMITS REPORT (PTLR)

a. RCS pressure and temperature limits for heat up, cooldown, low temperature operation, criticality, hydrostatic testing, LTOP enabling, and PORV lift settings as well as heatup and cooldown rates shall be established and documented in the PTLR for the following:

(1) LCO 3.4.3, "RCS Pressure and Temperature (P/T) Limits"
(2) LCO 3.4.6, "RCS Loops-MODE 4"
(3) LCO 3.4.7, "RCS Loops-MODE 5, Loops Filled"
(4) LCO 3.4.10, "Pressurizer Safety Valves"
(5) LCO 3.4.12, "Low Temperature Overpressure Protection (LTOP)"

b. The analytical methods used to determine the RCS pressure and temperature limits shall be those previously reviewed and approved by the NRC, specifically those described in the following documents:

(1) WCAP-14040-NP-A, "Methodology Used to Develop Cold Overpressure Mitigating System Setpoints and RCS Heatup and Cooldown Limit Curves", Revision 2, January 1996

(2) NRC Letter, "Point Beach Nuclear Plant, Units 1 and 2 - Exemption from the Requirements of 10CFR50.60 (TAC NOS. MA9680 and MA 9681)," dated October 6, 2000 D

(3) USNRC Regulatory Guide 1.99 Rev. 2

c. The PTLR shall be provided to the NRC upon issuance for each reactor vessel fluence period and for any revision or supplement thereto.