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Nuclear Management Company, LLC

NPL 2000-0371

10 CFR 50.90

August 17, 2000

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

Ladies/Gentlemen:

DOCKETS 50-266 AND 50-301
SUPPLEMENT 5 TO APPLICATION FOR AMENDMENT TO
FACILITY OPERATING LICENSE APPENDIX A:
TECHNICAL SPECIFICATIONS IMPROVEMENT PROJECT
RESPONSE TO RAI ON ITS SECTIONS 3.4 and 3.9
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 a letter dated July 3, 2000, the NRC issued a Request for Additional Information (RAI) to WE on ITS Sections 3.4 and 3.9.

Attachment 1 of this letter includes the Nuclear Management Company (NMC) response to the Staff's questions in the above referenced RAIs. In some instances, the response includes changes that are required to the original submittal, including changes to the Current Technical Specification (CTS) markups, Descriptions of Change (DOC), NUREG markups, proposed ITS and associated Bases, Justifications for Deviation (JFD), and No Significant Hazard Considerations (NSHC). These changes are discussed in the response to each question and are included in the attachment. Pages containing the changes required to the DOC, JFD, and NSHC are identified by "Rev. B".

A001

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

B

RAI 3.4.1-1

The revision bar identifies the section that has been revised; the B in the triangle identifies revision B; and the RAI number identifies which RAI question the revision relates to. 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 section 2.0, 3.1, 3.2, 3.5; reference letter NPL 00-0260).
- Supplement 3 dated June 19, 2000 (ITS section 3.6; reference letter NPL 00-0271).
- Supplement 4 dated July 28, 2000 (ITS section 3.8; reference letter NPL 00-0341).

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 E. Reddemann

Mark Reddemann
Site Vice President
Point Beach Nuclear Plant

Subscribed to and sworn before me
on this 17th day of August, 2000

Christine K. Pozorski Christine K. Pozorski
Notary Public, State of Wisconsin

My Commission expires on 8/25/2002.

JG/tat

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 3.4 and 3.9
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 July 3, 2000.

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

NRC Question 3.4.1-1:

ITS 3.4.1 RCS Pressure, Temperature, and Flow DNB Limits
CTS 15.3.10.G Operational Limitations
JFD-1 RCS Pressurizer Pressure

The improved Technical Specification (ITS) retains the current TS (CTS) Pressurizer Pressure Limits of "≥ 2205 psig during operation at 2250 psia, ≥ 1955 psig at 2000 psia."

Comment: There is not a discussion in the Bases concerning when either of these limits apply, and whether the limits change linearly between these pressures. Provide an explanation.

NMC Response:

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

Proposed ITS section 3.4.1 has been completely revised based on incorporation of the Core Operating Limits Report (COLR), TSTF-339, and Amendments 193/198, approved 2/08/00 (related to incorporation of new fuel design at PBNP). The pressurizer pressure limits and their explanation are now included in the COLR, and are dependent on the type of fuel design used in the reactors (i.e. the upper limit is to be used when there is 422V+ fuel in the reactor). The COLR was submitted to the NRC as TSCR 218, dated 03/02/2000.

As a result of incorporating these submittals, DOC M.02 was changed to "not used" and new DOC LA.01 was created (since this change is now less restrictive), a new "LA" NSHC was created, JFD 01 and JFD 02 were slightly revised, the CTS markup, the NUREG markup, and the proposed ITS and associated Bases were revised as appropriate (see attached markups).

NRC Question 3.4.1-2:

ITS 3.4.1 RCS Pressure, Temperature, and Flow DNB Limits
CTS 15.3.10.G Operational Limitations
DOC M.2 and JFD-2

The ITS adopts a new RCS average temperature limit to correspond with the new expanded Mode 1 Applicability in the ITS.

Comment: There is not a discussion in the Bases concerning the basis for these limits. Provide an explanation.

NMC Response:

A discussion of the RCS average temperature limit that corresponds to the expanded Mode 1 Applicability has been incorporated into the Bases.

NRC Question 3.4.1-3:

ITS 3.4.1 RCS Pressure, Temperature, and Flow DNB Limits
CTS 15.3.10.G Operational Limitations
DOC M.3

The CTS does not provide Required Actions if the DNB parameters are not maintained within limits, while the ITS provides 2 hours to restore the parameters to within limits or be in Mode 2 in 6 hours.

Comment: If the DNB parameters are not within limits, the CTS would require application of CTS 15.3.0.B which requires shutdown. This is a less restrictive change.

NMC Response:

DOC M.3 has been re-written and re-categorized as DOC L.2 to address the less restrictive change of providing 2 hours to restore the DNB parameters to within limits.

NRC Question 3.4.3-1:

ITS 3.4.3 RCS P/T Limits
CTS 15.3.1.B
R.1 and R.2

The Steam Generator P/T Limits and the Pressurizer P/T Limits are being relocated to the FSAR.

Comment: Is Point Beach to have a Technical Requirements Manual (TRM) or equivalent? If so, will these and other relocated limits be located there?

NMC Response:

Point Beach will have a TRM. Current plan is to relocate Steam Generator P/T Limits, Pressurizer P/T Limits, and other identified requirements to the TRM.

NRC Question 3.4.5-1:

ITS 3.4.5 RCS Loops-MODE 3
ITS B 3.4.5 LCO section
STS B 3.4.5 LCO section
JFD-5

Examples of tests that require all RCPs to be de-energized have been deleted from the ITS because they are not applicable to Point Beach. No examples are provided.

Comment: Recommend replacing the invalid examples with plant specific examples.

NMC Response:

After re-evaluation of the examples, validation of the RCP coastdown curve subsequent to changes in the RCS which result in changes to the flow characteristics of the RCS, is a plausible instance where all RCPs might be not in operation and the requirements of the Note would apply. Therefore, this example has been retained in the Bases.

NRC Question 3.4.6-1:

ITS 3.4.6 RCS Loops-MODE 4
CTS 15.3.1.B.2.a
ITS LCO 3.4.6 Note 2
DOC M.2 and DOC M.3

The CTS prohibits starting a RCP if there is not an adequate pressure absorbing volume in either the steam generators or the pressurizer. This limitation is deleted from the ITS because "no method exists to verify the volume ...". Furthermore, this deletion is described as a more restrictive change.

Comment: This justification is inadequate. While the PTLR probably will adequately address this concern (in the PTLR curves), this is not addressed. It appears that this CTS restriction is ignored in the ITS and the change is inaccurately categorized. Provide adequate justification for deleting these requirements.

NMC Response:

The conditions of CTS 15.3.15.B.2.a, whereby a RCP can be started with RCS temperature < 355 F, are alternatives to the conditions of CTS 15.3.15.B.2.b. If the conditions of CTS 15.3.15.2.a (adequate pressure absorbing volume in either the steam generators or the pressurizer) can not be established or verified, the RCP can still be started with RCS temperature < 355 F, if the conditions provided in CTS 15.3.15.B.2.b (secondary water temperature of each steam generator < 50 F above the temperature of the RCS) are met. As stated in DOCs M.2 and M.3, no quantifiable pressurizer water level and no method to verify the volume in the steam generator tubes could be identified to ensure an adequate volume to accommodate the swell resulting from a RCP start. Therefore, the conditions of CTS 15.3.15.2.a are not being retained in ITS. This change is more restrictive, because the only remaining condition in ITS under which a RCP can be started with RCS temperature less than the LTOP enabling temperature specified in the PTLR, will be to verify secondary water temperature of each steam generator < 50 F above the temperature of the RCS.

NRC Question 3.4.6-2:

ITS RCS Loops-MODE 4
ITS B 3.4.6 LCO section
STS B 3.4.6 LCO section
JFD-4

An example of a test that requires all RCPs to be de-energized has been deleted from the ITS because it is not applicable to Point Beach. No examples are provided.

Comment: Recommend replacing the invalid example with plant specific example(s)?

NMC Response:

The no flow rod drop test has been replaced with a plant specific example.

NRC Question 3.4.7-1:

ITS 3.4.7 RCS Loops-MODE 5, Loops Filled
ITS 3.4.7 LCO, Note 2
STS 3.4.7 LCO, Note 2
DOC M.2 and JFD-5

The CTS allows an RHR loop to be temporarily out of service, for an unspecified period, to perform Surveillance Requirements (SRs). The STS limits the time of inoperability to 2 hours. The ITS changes this time to 4 hours because 2 hours would be too limiting.

Comment: What makes Point Beach unique such that 2 hours is insufficient to perform required surveillances? Either provide a plant specific justification, or a TSTF change proposal, or adopt the STS time of 2 hours.

NMC Response:

After re-evaluation of the surveillances requirements for the RHR pumps and the methods under which they are performed, Point Beach will adopt the 2 hour time allowance for an RHR loop to be out of service to perform surveillance requirements. This change also results in the deletion of JFD 5.

NRC Question 3.4.7-2:

ITS 3.4.7 RCS Loops-MODE 5, Loops Filled
ITS B 3.4.7, LCO section
STS B 3.4.7, LCO section
JFD-3

The ITS Bases does not include the discussion about rod drop no-flow tests because they are not performed at Point Beach. The JFD-3 states that the "Bases description of startup testing is revised to reflect the actual testing performed at Point Beach." The ITS does not replace the deleted discussion with an applicable test to which Note 1 would apply.

Comment: Recommend adding appropriate discussion to ITS Bases.

NMC Response:

The no flow rod drop test has been included in the Bases as an example of a test which may be performed with RHR pumps not in operation.

NRC Question 3.4.7-3:

ITS 3.4.7 RCS Loops-MODE 5, Loops Filled
ITS LCO 3.4.7 b, ITS SR 3.4.5.2 and ITS SR 3.4.6.2.
STS LCO 3.4.7.b
JFD 7

ITS 3.4.7 specifies that steam generator level must be ≥ 30% narrow range. JFD 7 indicated that narrow range was added to avoid possible interpretation problems.

Comment: Recommend also adding "narrow range" to ITS SR 3.4.5.2 and ITS SR 3.4.6.2. Request you submit a TSTF change proposal to modify the STS.

NMC Response:

ITS SR 3.4.5.2 and ITS SR 3.4.6.2 have been modified to specify that steam generator level must be ≥ 30% narrow range. Appropriate justifications for these changes have also been provided.

NRC Question 3.4.9-1:

ITS 3.4.9 RCS Pressurizer
ITS 3.4.9 LCO statement
STS 3.4.9 LCO statement
JFD-1

The specific pressurizer operability requirements of water level and heater capacity are not mentioned in the proposed ITS LCO statement and are listed in the surveillance requirements.

Comment: Include the specific pressurizer operability requirements of water level and heater capacity in the LCO statement, as is done in both the CTS and STS.

NMC Response:

The pressurizer operability requirements of water level and heater capacity listed in the surveillance requirements have been duplicated in the LCO statement. This modification has also necessitated changes to JFD 1 and DOC M.2.

NRC Question 3.4.9-2:

ITS 3.4.9 RCS Pressurizer
ITS 3.4.9 Condition A
STS 3.4.9 Condition A
DOC M.3 and JFD-2

The ITS adds a new more restrictive LCO pressurizer level limit for Mode 1, based upon the "loss of normal feedwater accident analyses." If the Mode 1 pressurizer level limit is not met, then 6 hours is provided in proposed ITS Required Action A to restore level.

Comment: Neither the CTS nor the STS provide 6 hours to restore pressurizer level. Justify why the loss of normal feedwater accident analyses for Mode 1 allows the time (6 hours) to restore pressurizer level. Recommend including this discussion in the Bases. Also, is there no mass addition concern in Modes 2 and 3 that would necessitate a lower pressurizer level limit?

NMC Response:

Point Beach will adopt a 1 hour completion time to restore pressurizer water level in MODE 1. If pressurizer water level cannot be restored to within limits in 1 hour, Condition C will be entered, requiring the unit be in MODE 3 in 6 hours and MODE 4 in 12 hours. These completion times are consistent with the requirements of CTS 15.3.0.B, which would be entered for pressurizer water level not within limits.

The magnitude of excursions for a loss of normal feedwater in MODES 2 and 3 are to a much lesser degree, and therefore do not necessitate a lower pressurizer water level limit.

<u>NRC Question 3.4.12-1:</u>

ITS 3.4.12 LTOP
STS 3.4.12 Required Action D.1
JFD-10

When an accumulator's pressure is greater than that allowed in the PTLR and it cannot be isolated, STS Required Action D.1 is to increase RCS cold leg temperature in order to exit the applicability of the LCO. This is proposed to be deleted in the ITS because it "could be easily misinterpreted as an allowance to enter the identified condition...", and such an action to restore compliance is not necessary to state.

Comment: I do not understand the potential misinterpretation; discuss. Also, STS Required Action D.1 is not an action to return conditions to that required by an LCO, rather it is an action to exit an applicability of an LCO, similar to Required Action D.2, and deleting D.1 could possibly be misinterpreted to mean that D.2 is the sole method for responding to the condition.

NMC Response:

STS 3.4.12, Required Action D.1 has been added to ITS 3.4.12, as Required Action C.1, to provide an alternative method for responding to the condition. This change also results in the deletion of JFD 10.

NRC Question 3.4.12-2:

ITS 3.4.12 LTOP
ITS SR 3.4.12.3
STS SR 3.4.12.3
JFD-12

The ITS modifies STS SR 3.4.12.3, which verifies accumulators are isolated, to require its performance only when the accumulator(s) are required to be isolated. The SR is modified by an added phrase to the SR description.

Comment: Recommend adding the modifying statement to ITS SR 3.4.12.3 as a Note, similar to the Note added to ITS SR 3.4.12.2, for consistency in presentation.

NMC Response:

The modifying statement to ITS SR 3.4.12.3 has been changed to a Note similar to the Note added to ITS SR 3.4.12.2.

NRC Question 3.4.12-3:

ITS 3.4.12 LTOP
ITS 3.4.12 LCO statement
STS 3.4.12 LCO statement
JFD-1

The ITS 3.4.12 LCO statement deletes reference to the LTOP "configuration" as a system.

Comment: The CTS refers to the LTOP System, as does the STS. The ITS refers to LTOP Trains of equipment. When LTOP is controlling pressure it is appropriate to refer to the equipment configurations and functioning as the LTOP System; it seems awkward not to refer to it as the LTOP System.

NMC Response:

ITS 3.4.12 LCO statements have been restored to the designations presented in the ISTS, in order to refer to the LTOP "configuration" as a system. This change also results in the deletion of JFD 1.

NRC Question 3.4.13-1:

ITS 3.4.13 RCS Operational Leakage
ITS 3.4.13 Require Action A.1 Completion Time
STS 3.4.13 Require Action A.1 Completion Time
DOC A.2 and JFD-3

The CTS provides 4 hours to conduct an evaluation of the leakage, and to commence a shutdown no later than 24 hours. The STS provides 4 hours to reduce the leakage in Required Action A, and to shutdown in 6 hours per Condition B.

Comment: Adopt the STS time to reduce leakage; Require Action A.1 Completion Time of 4 hours. The CTS time of 24 hours is to commence a shutdown, and not time to reduce leakage.

NMC Response:

CTS 15.3.1.D.1 requires an evaluation of the leakage be initiated as soon as practicable, but no later than 4 hours. The CTS does not specify a required completion time for the evaluation. However, CTS 15.3.1.D.2 requires that if the indicated reactor coolant leakage is substantiated and is not evaluated as safe or is determined to exceed 10 gpm, reactor shutdown shall be initiated as soon as practicable, but no later than 24 hours after the leak was first detected. This would imply that the evaluation is required to be completed within 24 hours, or a reactor shutdown is required to be initiated. Additionally, continued operation with leakage in excess of the limits is allowed, until it is practicable to commence a reactor shutdown, although no later than 24 hours after the leak was first detected.

Secondly, CTS 15.3.1.D.4 requires that if the leakage is determined to be primary to secondary SG leakage in excess of 500 GPD in either SG, the reactor shall be shutdown and placed in the cold shutdown condition within 30 hours after detection. However, ITS would allow 24 hours to commence a reactor shutdown, and an additional 36 hours to cooldown the unit to a cold shutdown condition. This change has been identified as a less restrictive change, and is justified in DOC L.1.

Finally, CTS 15.3.1.D.5 requires that if the coolant leakage exists through a non-isolable fault in a reactor coolant system component, the reactor shall be shutdown, and cooldown to the cold shutdown condition shall be initiated within 24 hours of detection. Therefore, continued operation with leakage in excess of the limits is allowed for up to 24 hours.

Although the CTS does not require RCS leakage to be reduced within 4 hours, good engineering

practice dictates that efforts to reduce the leakage will be ongoing until the leakage is reduced to within limits. Therefore, allowing 24 hours to reduce RCS leakage to within limits before actions are required to commence a reactor shutdown, is consistent with the current licensing basis and safe operation of the plant.

NRC Question 3.4.13-2:

ITS 3.4.13 RCS Operational Leakage
CTS 15.3.1.D Leakage of Reactor Coolant
DOC LA.1, DOC LA.2, and DOC LA.3

Details regarding RCS leakage are being "moved to licensee control."

Comment: Identify the location to which the details are being moved and the change control procedure to be utilized.

NMC Response:

Details of CTS 15.3.1.D.1 identified in DOC LA.1 and DOC LA.2 have been relocated to the Bases of ITS LCO 3.4.13. Changes to these details will be controlled in accordance with the provisions of the Bases Control Program described in Chapter 5 of the Technical Specifications.

CTS 15.3.1.D.3 provides information to be considered in the safety evaluation of a RCS leak and information to be contained in the safety evaluation concerning plant shutdown and exposure to offsite personnel. This information is not being retained in the ITS. DOC L.4 will replace DOC LA.3 to facilitate and justify this deletion.

NRC Question 3.4.13-3:

ITS 3.4.13 RCS Operational Leakage
CTS 15.4.3 Primary System Testing Following Opening
DOC R.1

Comment: The justification to DOC R.1 for relocating CTS 15.4.3 is missing.

NMC Response:

DOC R.1 has been provided.

NRC Question 3.4.13-4:

ITS 3.4.13 RCS Operational Leakage
CTS 15.3.1.D Basis
DOC A.5

DOC A.5 is identified with CTS 15.3.1.D Basis. However, DOC A.5 is not included in the submittal.

Comment: Licensee provide DOC A.5.

NMC Response:

DOC A.5 has been provided.

NRC Question 3.4.14-1:

ITS 3.4.14 RCS PIV Leakage
STS SR 3.4.14.2 and STS SR 3.4.14.3
JFD-1

The ITS does not include the STS surveillances SR 3.4.14.2 and SR 3.4.14.3.

Comment: Why not?

NMC Response:

The RHR System autoclosure interlock is not part of the Point Beach design, and is not credited for mitigation of any accident.

NRC Question 3.4.15-1:

ITS 3.4.15 RCS Leakage Detection Instrumentation
CTS 15.4.1, Table 15.4.1-1, items 36-07 and 43
DOC LA.1

Surveillance Requirements on the Air Ejector Monitor and the Volume Control Tank Level Instrumentation are being "moved to licensee control."

Comment: Identify the location (TRM?) to which the Surveillance Requirements are being moved and the change control procedure to be utilized.

NMC Response:

The Air Ejector Monitor and the Volume Control Tank Level Instrumentation Surveillance Requirements will be located in the Technical Requirements Manual (TRM), which will be maintained using the provisions of 10 CFR 50.59.

NRC Question 3.4.16-1:

ITS RCS Specific Activity
CTS 15.4.1, Table 15.4.1-2
DOC LA.1, DOC LA.2, and DOC LA.3

Details regarding RCS sampling are being "moved to licensee control."

Comment: Identify the location to which the details are being moved and the change control procedure to be utilized.

NMC Response:

Details regarding RCS sampling are being relocated to the Technical Requirements Manual (TRM), which will be maintained using the provisions of 10 CFR 50.59.

<u>NRC Question 3.4.16-2:</u>

CTS 15.3.1.E Maximum Reactor Coolant Oxygen and Chloride and Fluoride Concentration for Power Operation
DOC R.1

CTS 15.3.1.E is being relocated to documents outside the TS.

Comment: Identify the location to which CTS 15.3.1 is being relocated (TRM?) and the change control procedure to be utilized.

NMC Response:

CTS 15.3.1 is being relocated to the Technical Requirements Manual (TRM), which will be maintained using the provisions of 10 CFR 50.59.

Additional Corrections Required to ITS Section 3.4:

Additional corrections to the conversion package for ITS Section 3.4 have been identified as a result of ITS reviews by plant staff.

Sections 3.4.10 and 3.4.12 have been completely revised based on changes to the LTOP enabling temperature, which was submitted as part of TSCR 219 on March 14, 2000 (Adoption of PTLR).

The Bases discussion of ITS LCO 3.4.6 has been modified by the addition of a sentence to clarify that SG secondary water side water temperature can be closely approximated by using the SG metal temperature indicator . This change also results in the addition of JFD 9.

Section 3.4.11, DOC LA.1 discussed the deletion of surveillance requirements on the PORV automatic actuation function, because the function is not credited as a mitigative function for any analyzed accident at Point Beach. DOC LA.1 was mis-categorized and has been re-categorized as an "L" DOC. A specific NSHC for this less restrictive change has also been written.

A typo in ITS 3.4.11, Condition G was revised, such that it correctly references Condition F instead of Condition E.

Section 3.4.13, DOC LA.1 has been revised to more correctly identify which portion of CTS 15.3.1.D.1 is being relocated, and to specify that these details are being relocated to the Bases.

Section 3.4.14 has been revised by updating the location where the PIVs will be listed. Although the list will still be located in the TRM, the TRM will no longer be Appendix T to the FSAR.

Section 3.4.16, DOC LA.01 discussed the deletion of the reactor coolant gross beta-gamma sampling requirements below 500 F, because the LCO limit for gross specific activity when operating in MODES 1 and 2, and in MODE 3 with RCS average temperature greater than or equal to 500 F, is necessary to contain the potential consequences of a steam generator tube rupture (SGTR) to within acceptable site boundary dose values. When the unit is operating in MODE 3 with RCS average temperature less than 500 F, and in MODES 4 and 5, the release of radioactivity in the event of a SGTR is unlikely, because the saturation pressure of the reactor coolant is below the lift pressure settings of the main steam safety valves. DOC LA.1 was mis-categorized and has been re-categorized as an "L" DOC. A specific NSHC for this less restrictive change has also been written.

<u>NRC Question 3.9.3-1:</u>

3.9.3-1 ITS B 3.9.3 LCO section
JFD-2
DOC L.3

The ITS Bases includes words at the end of the LCO section concerning the allowance to leave containment airlock doors open during fuel movement and other core alts.

Comment: The middle paragraph of DOC L.3 seems to be a very appropriate paragraph to include in the Bases, on why it is acceptable for both airlock doors to be open during fuel movement and other core alts.

NMC Response:

ITS 3.9.3 Bases, LCO Section, has been modified by the addition of a statement that provides the basis for allowing containment personnel airlocks to remain open during fuel movements and core alterations.

NRC Question 3.9.3-2:

3.9.3-2 ITS SR 3.9.3.2 Note
STS SR 3.9.4.2
JFD-4

The ITS SR 3.9.3.2 includes a note that the SR is not applicable to valves in isolated penetrations, to avoid confusion over whether a failed surveillance conducted on an isolated is a failed SR that would require a TS condition entry.

Comment: The change is appropriate; request you submit a TSTF change proposal to modify the STS.

NMC Response:

TSTF-284, Rev. 3, approved by the NRC in January, 2000, addresses this change.

NRC Question 3.9.4-1:

3.9.4-1 STS 3.9.5 Required Action A.4
JFD-3

The ITS does not include STS Required Action A.4, to close containment penetrations to the outside atmosphere, because it is not included in the Point Beach current licensing basis.

Comment: Perhaps the CTS, or current licensing basis, should have included this action. Are there any dose calculations, resulting from a core melt accident, to support this exclusion?

NMC Response:

A review of this scenario for Point Beach confirms that the current licensing basis is acceptable. With the reactor in a refueling shutdown, sufficient time will have elapsed since cessation of critical operations such that the decay heat rate will have significantly decreased. Therefore, decay heat removal requirements will be significantly below the maximum design values postulated following a design basis accident. In the unlikely event of a complete loss of all decay heat removal capability under these conditions, the only result initially will be that the water temperature of the refueling cavity will begin to slowly increase. With the refueling cavity filled, a significant amount of time will pass before the water begins to boil. The entire volume of the

refueling cavity above the level of the active fuel must then be boiled off before fuel temperatures will begin to appreciably rise. If the refueling cavity were not completely filled, sufficient volume remains available in the refueling water storage tank to fill it; the capability to do so remains available even with all decay heat removal loops inoperable. Furthermore, existing Point Beach procedures, which are subject to the provisions of 10 CFR 50.59, provide for shutting of containment penetrations under these conditions.

During refueling shutdown, safely maintaining fuel in the refueling cavity is analogous to safely maintaining fuel in the spent fuel pool (SFP). As stated in the Point Beach FSAR, "The calculated values for the bulk water temperature of the SFP are not safety limits, and the nominal conditions assumed in the analysis are not operational limits. As discussed in Reference 3, the design criteria for the SFP thermal and hydraulic analyses are derived from the NRC position papers 'Position for Review and Acceptance of Spent Fuel Storage and Handling Applications,' and include: Decay heat rates are calculated in accordance with Branch Technical Position ASB 9-2 of the Standard Review Plan (NUREG 0800) to assure calculations are conservatively high. Adequate time exists for an alternate cooling method to be implemented in the event of a complete loss of SFP Cooling System capability." "In the event of complete failure of the cooling system for a long period of time, the fuel pool water inventory can be maintained with fire suppression system water." Sufficient time will be available from the time of the loss of all decay heat removal capability until the volume of water in the refueling cavity has boiled away to initiate compensatory actions to restore decay heat removal or provide compensatory means of cooling water inventory addition to the refueling cavity. Since the response to this condition is not time-critical, existing licensee controls to ensure closure of containment penetrations are sufficient. Inclusion of these controls within ITS is not warranted. Therefore, the ITS, as proposed per the current licensing basis, is acceptable.

<u>NRC Question 3.9.5-1:</u>

3.9.5-1 STS 3.9.6 Required Action B.3
JFD-2

The ITS does not include STS Required Action B.3, to close containment penetrations to the outside atmosphere, because it is not included in the Point Beach current licensing basis.

Comment: Perhaps the CTS, or current licensing basis, should have included this action. Are there any dose calculations, resulting from a core melt accident, to support this exclusion?

NMC Response:

See *NMC* Response to NRC Question 3.9.4-1.

Additional Corrections Required to ITS Section 3.9:

Additional corrections to the conversion package for ITS Section 3.9 have been identified as a result of ITS reviews by plant staff.

The ISTS 3.9.1 Bases markup and ITS 3.9.1 Bases, Background discussion of the methods used to fill the refueling cavity and refueling canal have been reverted to the text of the ISTS. Also, the text of JFD 2 has been changed to "not used." The discussion in the ISTS adequately describes the methods used at Point Beach to fill the refueling cavity and refueling canal.

Additionally, the ISTS 3.9.1 Bases markup and ITS 3.9.1 Bases discussion of SR 3.9.1.1 have been modified to specify that a representative sample of the interconnected volumes of the RCS, refueling cavity and/or refueling canal meets the intent of the surveillance requirement.

Lastly, the CTS markup for 3.9.4 has been modified, DOC A.2 has been changed to "not used," and a new DOC (L.4) with associated NSHC have been added. These changes were necessary to reflect the deletion of the requirement for containment penetration closure during the movement of core components other than irradiated fuel inside containment.

ATTACHMENT 2
DISCARD AND INSERTION INSTRUCTIONS

VOLUME 5	
SECTION 3.4.1	
DISCARD	**INSERT**
DOC pages 1 of 4 through 4 of 4	DOC pages 1 of 5 through 5 of 5
CTS markup pages 1 of 3 through 3 of 3	CTS markup pages 1 of 2 and 2 of 2
JFD pages 1 of 2 and 2 of 2	JFD pages 1 of 2 and 2 of 2
ISTS markup pages 3.4-1, 3.4-2 and Insert	ISTS markup pages 3.4-1 and 3.4-2
ISTS Bases markup pages B 3.4-1 through B 3.4-3	ISTS Bases markup pages B 3.4-1 through B 3.4-3
NSHC pages 1 of 3 through 3 of 3	NSHC pages 1 of 5 through 5 of 5
ITS pages 3.4.1-1 through 3.4.1-3	ITS pages 3.4.1-1 and 3.4.1-2
ITS Bases pages B 3.4.1-1 through B 3.4.1-5	ITS Bases pages B 3.4.1-1 through B 3.4.1-4
SECTION 3.4.5	
DISCARD	**INSERT**
JFD pages 2 of 3 and 3 of 3	JFD pages 2 of 3 and 3 of 3
ISTS markup page 3.4-3	ISTS markup page 3.4-3
ISTS Bases markup pages B 3.4-22 and B 3.4-23	ISTS Bases markup pages B 3.4-22 and B 3.4-23
ITS page 3.4.5-2	ITS page 3.4.5-2
ITS Bases pages B 3.4.5-1 through B 3.4.5-5	ITS Bases pages B 3.4.5-1 through B 3.4.5-5
SECTION 3.4.6	
DISCARD	**INSERT**
JFD pages 1 of 2 and 2 of 2	JFD pages 1 of 2 and 2 of 2
ISTS markup page 3.4-12	ISTS markup page 3.4-12
ISTS Bases markup page B 3.4.6-2	ISTS Bases markup page B 3.4.6-2
ITS page 3.4.6-2	ITS page 3.4.6-2
ITS Bases pages B 3.4.6-1 through B 3.4.6-5	ITS Bases pages B 3.4.6-1 through B 3.4.6-4

ATTACHMENT 2
DISCARD AND INSERTION INSTRUCTIONS

SECTION 3.4.7	
DISCARD	**INSERT**
DOC pages 4 of 6 and 5 of 6	DOC pages 4 of 6 and 5 of 6
CTS markup page 4 of 5	CTS markup page 4 of 5
JFD pages 1 of 2 and 2 of 2	JFD pages 1 of 2 and 2 of 2
ISTS markup page 3.4-14	ISTS markup page 3.4-14
ISTS Bases markup pages B 3.4.7-2 and B 3.4.7-3	ISTS Bases markup pages B 3.4.7-2 and B 3.4.7-3
ITS page 3.4.7-1	ITS page 3.4.7-1
ITS Bases pages B 3.4.7-1 through B 3.4.7-5	ITS Bases pages B 3.4.7-1 through B 3.4.7-5
SECTION 3.4.9	
DISCARD	**INSERT**
DOC pages 1 of 3 through 3 of 3	DOC pages 1 of 4 through 4 of 4
CTS markup page 5 of 5	CTS markup page 5 of 5
JFD pages 1 of 4 through 4 of 4	JFD pages 1 of 3 through 3 of 3
ISTS markup pages 3.4-19 and 3.4-20	ISTS markup pages 3.4-19 and 3.4-20
ISTS Bases markup Insert	ISTS Bases markup Insert
ITS page 3.4.9-1	ITS page 3.4.9-1
ITS Bases pages B 3.4.9-1 through B 3.4.9-5	ITS Bases pages B 3.4.9-1 through B 3.4.9-4
SECTION 3.4.10	
DISCARD	**INSERT**
DOC pages 1 of 4 through 4 of 4	DOC pages 1 of 3 through 3 of 3
CTS markup pages 1 of 4, 2 of 4 and 4 of 4	CTS markup pages 1 of 4, 2 of 4 and 4 of 4
JFD pages 1 of 2 and 2 of 2	JFD page 1 of 1
ISTS markup page 3.4-21	ISTS markup page 3.4-21
ISTS Bases markup pages B 3.4.10-45, B 3.4.10-47 and B 3.4.10-48	ISTS Bases markup pages B 3.4.10-45, B 3.4.10-47 and B 3.4.10-48
NSHC pages 1 of 6 through 6 of 6	NSHC pages 1 of 5 through 5 of 5

ATTACHMENT 2
DISCARD AND INSERTION INSTRUCTIONS

SECTION 3.4.10 (continued)	
DISCARD	**INSERT**
ITS page 3.4.10-1	ITS page 3.4.10-1
ITS Bases pages B 3.4.10-1 through B 3.4.10-4	ITS Bases pages B 3.4.10-1 through B 3.4.10-4
SECTION 3.4.11	
DISCARD	**INSERT**
DOC pages 1 of 4 and 2 of 4	DOC pages 1 of 4 and 2 of 4
CTS markup pages 3 of 6 and 4 of 6	CTS markup pages 3 of 6 and 4 of 6
NSHC page 2 of 4	NSHC page 2 of 4
ITS page 3.4.11-3	ITS page 3.4.11-3
SECTION 3.4.12	
DISCARD	**INSERT**
DOC pages 1 of 9 through 9 of 9	DOC pages 1 of 7 through 7 of 7
CTS markup pages 1 of 10, 2 of 10, 9 of 10 and 10 of 10	CTS markup pages 1 of 10, 2 of 10, 9 of 10 and 10 of 10
JFD pages 1 of 8 through 8 of 8	JFD pages 1 of 7 through 7 of 7
ISTS markup page 3.4-27 through 3.4-32 and Insert	ISTS markup page 3.4-27 through 3.4-32 and Insert
ISTS Bases markup pages B 3.4.12-1, B 3.4.12-2, B 3.4.12-4 through B 3.4.12-14 and Insert pages 1 of 3 and 2 of 3	ISTS Bases markup pages B 3.4.12-1, B 3.4.12-2, B 3.4.12-4 through B 3.4.12-14 and Insert pages 1 of 2 and 2 of 2
ITS pages 3.4.12-1 through 3.4.12-5	ITS pages 3.4.12-1 through 3.4.12-5
ITS Bases pages B 3.4.12-1 through B 3.4.12-13	ITS Bases pages B 3.4.12-1 through B 3.4.12-11
SECTION 3.4.13	
DISCARD	**INSERT**
DOC pages 1 of 5 through 5 of 5	DOC pages 1 of 6 through 6 of 6
CTS markup page 1 of 9	CTS markup page 1 of 9
NSHC pages 1 of 6 through 6 of 6	NSHC pages 1 of 8 through 8 of 8

ATTACHMENT 2
DISCARD AND INSERTION INSTRUCTIONS

SECTION 3.4.14	
DISCARD	**INSERT**
ISTS Bases markup pages B 3.4.14-2 and B 3.4.14-7	ISTS Bases markup pages B 3.4.14-2 and B 3.4.14-7
ITS Bases pages B 3.4.14-1 through B 3.4.14-6	ITS Bases pages B 3.4.14-1 through B 3.4.14-5
SECTION 3.4.16	
DISCARD	**INSERT**
DOC page 3 of 6	DOC page 3 of 6
CTS markup pages 6 of 7 and 7 of 7	CTS markup pages 6 of 7 and 7 of 7
NSHC pages 1 of 7 through 7 of 7	NSHC pages 1 of 8 through 8 of 8
VOLUME 10	
SECTION 3.9.1	
DISCARD	**INSERT**
JFD page 1 of 2	JFD page 1 of 2
ISTS Bases markup pages B3.9-1 and B3.9-4	ISTS Bases markup pages B3.9-1 and B3.9-4
ITS Bases pages B3.9.1-1 through B3.9.1-4	ITS Bases pages B3.9.1-1 through B3.9.1-3
SECTION 3.9.4	
DISCARD	**INSERT**
DOC pages 1 of 4 through 4 of 4	DOC pages 1 of 4 through 4 of 4
CTS markup pages 1 of 4 and 3 of 4	CTS markup pages 1 of 4 and 3 of 4
JFD page 3 of 3	JFD page 3 of 3
ISTS Bases markup pages B3.9.4-4 and B3.9.4-6	ISTS Bases markup pages B3.9.4-4 and B3.9.4-6
NSHC pages 1 of 5 through 5 of 5	NSHC pages 1 of 6 through 6 of 6
ITS Bases pages B3.9.3-1 through B3.9.3-5	ITS Bases pages B3.9.3-1 through B3.9.3-4

ENCLOSURE

Description of Changes - NUREG-1431 Section 3.04.01

03-Aug-00

DOC Number	DOC Text	
A.01 Rev. A	In the conversion of Point Beach current Technical Specifications (CTS) to the proposed plant specific Improved Technical Specifications (ITS), certain wording preferences or conventions are adopted which do not result in technical changes (either actual or interpretational). Editorial changes, reformatting, and revised numbering are adopted to make the ITS consistent with the Standard Technical Specifications, Westinghouse Plants, NUREG-1431, Revision 1 (i.e., Improved Standard Technical Specifications (ISTS)).	
	CTS:	**ITS:**
	15.03.01.G	LCO 3.04.01
	15.03.01.G.01	LCO 3.04.01 B
	15.03.01.G.02	LCO 3.04.01 A
	15.03.01.G.03	LCO 3.04.01 C
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 PBNP ITS, consistent with the Standard Technical Specifications for Westinghouse Plants, NUREG-1431. The revised Bases are as shown in the PBNP ITS Bases.	
	CTS:	**ITS:**
	BASES	B 3.04.01
L.01 Rev. A	A Note has been added to indicate the limit on pressurizer pressure is not applicable during short term operational transients such as a THERMAL POWER ramp increase > 5% RTP per minute or a THERMAL POWER step increase > 10% RTP. This Note relaxes the requirements on pressurizer pressure and is therefore less restrictive. This change is acceptable since these conditions represent short term perturbations where actions to control pressure variations might be counterproductive. Also, since they represent transients initiated from power levels < 100% RTP, an increased DNBR margin exists to offset the temporary pressure variations.	
	CTS:	**ITS:**
	NEW	LCO 3.04.01 APPL NOTE A LCO 3.04.01 APPL NOTE B

Description of Changes - NUREG-1431 Section 3.04.01

03-Aug-00

DOC Number	DOC Text
L.02 Rev. B	CTS 15.3.1.G does not provide actions in the event the DNB parameters are not maintained within limits. Therefore, CTS 15.3.0.B requires action be initiated within 1 hour to place the plant in a condition whereby the specification does not apply. ITS LCO 3.4.1, Condition A, addresses a condition where one or more RCS DNB parameter(s) are not within limits. With any DNB parameter not within LCO limits, Required Action A.1 requires the restoration of the DNB parameter(s) within 2 hours. If Required Action A.1 is not met within the associated Completion Time, Required Action B.1 requires placing the plant in a MODE in which the LCO does not apply. To achieve this status, the plant must be brought to at least MODE 2 within 6 hours. In MODE 2, the reduced power condition eliminates the potential for violation of the accident analysis bounds. The proposed Completion Time of 6 hours is reasonable to reach the required plant conditions in an orderly manner. Adopting the allowance of 2 hours to restore the DNB parameters to within limits is a less restrictive change. This change is acceptable in order to provide sufficient time to adjust plant parameters, to determine the cause for the off normal condition, and to restore the readings to within limits, and does not result in a significant reduction in the margin of safety.

CTS:	ITS:
NEW	LCO 3.04.01 COND A
	LCO 3.04.01 COND A RA A.1
	LCO 3.04.01 COND B
	LCO 3.04.01 COND B RA B.1

DOC Number	DOC Text
LA.01 Rev. B	CTS 15.3.1.G.1 specifies a Tavg range based on fuel assemblies in the reactor cores during "rated power operation". As discussed in Description of Change M.1 of this section, CTS 15.3.1.G.1 applicability for the limitation on Tavg is revised from ". . . during Rated Power operation," to "MODE 1". CTS defines Rated Power as, "... steady state reactor core output of 1518.5 MWt." ITS defines MODE 1 as "power operation with keff greater than or equal to 0.99 and > 5% Rated Thermal Power, excluding decay heat." Therefore the Tavg limit in the proposed ITS (COLR - see below discussion) is revised to include all operation with Rated Thermal Power > 5% (ie. graphed Tavg on y-axis vs. RTP on x-axis).

This change will necessitate the addition of two figures to delineate the operating envelope of minimum and maximum Tavg over the operating range of 5% to 100% Rated Thermal Power (the two figures are dependent on the type of fuel assemblies in the reactor cores). As described below, these figures will be contained in the COLR. The values contained in these figures are consistent with the assumptions made in the safety analysis.

CTS 15.3.1.G.3 specifies different minimum RCS total flow rates, dependent on the type of fuel in the reactor cores. The proposed ITS will specify the larger of the two flow rates (182,400gpm), since this flow rate bounds the lower flow rate and the PBNP cores will eventually contain all 422V+ fuel assemblies.

The specific limits for RCS Tave, Pressurizer Pressure, and RCS total flow rate are relocated to the 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."

These limits can be relocated with no impact on safety. The limits alert the licensee of a potential violation of a DNB related parameter. Additional evaluation will be required to determine if an actual safety limit (DNBR and fuel centerline melt design basis limits), which are included in the proposed ITS, has been violated. Therefore, there is no reduction in a level of safety by relocating these values to the COLR as controls are still in place to define and ensure appropriate action is taken in the event of a violation of a safety limit.
The limit on RCS flow is retained with the cycle specific value located to the COLR. The DNBR limit is retained in ITS 2.1.1, allowing the relocation of the cycle specific limits to the COLR with no reduction in a margin of safety. This change is less restrictive, since the curves are being relocated out of the Technical Specifications and into the COLR, which is under licensee control.

CTS:	ITS:
15.03.01.G.01	COLR
15.03.01.G.02	COLR
15.03.01.G.03	COLR

DOC Number	DOC Text	
M.01 Rev. A	CTS 15.3.1.G is applicable during Rated Power operations. CTS defines Rated Power as, "... steady state reactor core output of 1518.5 MWt." ITS 3.4.1 is applicable in MODE 1. ITS defines MODE 1 as power operation with keff greater than or equal to 0.99 and > 5% Rated Thermal Power, excluding decay heat. This change results in increasing the plant operating conditions over which this specification is applicable and is therefore more restrictive. In MODE 1, the limits on pressurizer pressure, RCS coolant average temperature, and RCS flow rate must be maintained during steady state operation in order to ensure DNBR criteria will be met in the event of an unplanned loss of forced coolant flow or other DNB limited transient.	
	CTS:	**ITS:**
	15.03.01.G	LCO 3.04.01
M.02 Rev. B	Not Used	
	CTS:	**ITS:**
	N/A	N/A
M.03 Rev. B	Not Used	
	CTS:	**ITS:**
	N/A	N/A

DOC Number	DOC Text
M.04 Rev. A	CTS 15.3.1.G is revised to adopt ITS SR 3.4.1.1, SR 3.4.1.2 and SR 3.4.1.3. Proposed SR 3.4.1.1 and SR 3.4.1.2 require the verification that pressurizer pressure and RCS average temperature, respectively, are within limits every 12 hours. Since Required Action A.1 allows a Completion Time of 2 hours to restore parameters that are not within limits, the 12 hour Surveillance Frequency is sufficient to ensure the parameter(s) can be restored to a normal operation, steady state condition following load changes and other expected transient operations. The 12 hour interval has been shown by operating practice to be sufficient to regularly assess for potential degradation and to verify operation is within safety analysis assumptions. Proposed SR 3.4.1.3 requires the verification that measured RCS total flow rate is within limits every 18 months. Measurement of RCS total flow rate every 18 months allows the installed RCS flow instrumentation to be calibrated and verifies the actual RCS flow rate is greater than or equal to the minimum required RCS flow rate. This verification is performed via a precision calorimetric heat balance. The Frequency of 18 months reflects the importance of verifying flow after a refueling outage when the core has been altered, which may have caused an alteration of flow resistance. This SR is modified by a Note that allows entry into MODE 1, without having performed the SR, and placement of the unit in the best condition for performing the SR. The Note states that the SR is not required to be performed until 24 hours after greater than or equal to 90% RTP. This exception is appropriate since the heat balance requires the plant to be at a minimum of 90% RTP to obtain the stated RCS flow accuracies. The Surveillance shall be performed within 24 hours after reaching 90% RTP. This change imposes new requirements on plant operations and is more restrictive.

CTS:	ITS:
NEW	SR 3.04.01.01
	SR 3.04.01.02
	SR 3.04.01.03

A.1

Pressurizer Pressure is greater than or equal to the limits specified in the COLR.

LA.1

RCS average temperature is within the limits specified in the COLR.

G. OPERATIONAL LIMITATIONS

The following DNB related parameters shall be maintained within the limits shown ~~during Rated Power operation:~~ in MODE 1:

M.1

Add LCO 3.4.1 Applicability Note. See Insert 3.4.1-1.

L.1

1. ~~T_{avg} shall be maintained ≥558.1°F and ≤574.0°F for cores containing 422V+ fuel assemblies. T_{avg} shall be maintained ≥557°F and ≤573.9°F for cores not containing 422V+ fuel assemblies.~~

2. ~~Reactor Coolant System (RCS) pressurizer pressure shall be maintained: ≥2205 psig during operation at 2250 psia, or ≥1955 psig during operation at 2000 psia for cores not containing 422V+ fuel assemblies.~~

3. ~~Reactor Coolant System raw measured Total Flow Rate shall be maintained ≥182,400gpm for cores containing 422V+ fuel assemblies, or ≥181,800 gpm for cores not containing 422V+ fuel assemblies.~~

RCS total flow rate is ≥ 182,400 gpm and greater than or equal to the limit specified in the COLR.

LA.1

Add Action A and Action B. See Insert 3.4.1-2.

L.2

Add SR 3.4.1.1, SR 3.4.1.2 and SR 3.4.1.3. See Insert 3.4.1-3.

M.4

B
COLR & Amendment 193/198

Basis:

The reactor coolant system total flow rate of 182,400 gpm for cores containing 422V+ fuel assemblies is based on an assumed measurement uncertainty of 2.4 percent over thermal design flow (178,000 gpm). The reactor coolant system total flow rate of 181,800 gpm for cores not containing 422V+ fuel assemblies is based on an assumed measurement uncertainty of 2.1 percent over thermal design flow (178,000 gpm). The raw measured flow is based upon the use of normalized elbow tap differential pressure which is calibrated against a precision flow calorimetric at the beginning of each cycle.

A.2

B
Amendment 193/198

LCO 3.4.1 CTS Mark up Inserts

Insert 3.4.1-1:

---------------------------NOTE---------------------------
Pressurizer pressure limit does not apply during:

a. THERMAL POWER ramp > 5% RTP per minute; or

b. THERMAL POWER step > 10% RTP.

Insert 3.4.1-2:

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more RCS DNB parameters not within limits.	A.1 Restore RCS DNB parameter(s) to within limit.	2 hours
B. Required Action and associated Completion Time not met.	B.1 Be in MODE 2.	6 hours

Insert 3.4.1-3:

SURVEILLANCE		FREQUENCY
SR 3.4.1.1	Verify pressurizer pressure is greater than or equal to the limits specified in the COLR.	12 hours
SR 3.4.1.2	Verify RCS average temperature is within the limits specified in the COLR.	12 hours
SR 3.4.1.3	-------------------NOTE------------------- Not required to be performed until 24 hours after ≥ 90% RTP. -- Verify by precision heat balance that RCS total flow rate is ≥ 182,400 gpm and greater than or equal to the limit specified in the COLR.	18 months

B
COLR

Justification For Deviations - NUREG-1431 Section 3.04.01

01-Aug-00

JFD Number	JFD Text	
01 Rev. B	The brackets have been removed and the proper plant specific information has been provided. "Limit" was changed to "Limits" in proposed ITS LCO 3.4.1.a because different limits are allowed in the CTS (which will also be retained in the COLR) based on what pressure the units are operating at (i.e. 2250 psia or 2000 psia).	
	ITS:	**NUREG:**
	B 3.04.01	B 3.04.01
	COLR	N/A N/A
	LCO 3.04.01 A	LCO 3.04.01 A
	SR 3.04.01.01	SR 3.04.01.01
	SR 3.04.01.03	SR 3.04.01.04
02 Rev. B	NUREG-1431, LCO 3.4.1.b is modified by changing "less than or equal to" to "within" and changing "limit" to "limits". These changes were necessary because figures have been added to the COLR (RCS Average Temperature Limits) to facilitate the Point Beach minimum and maximum Tavg limits for Rated Power operations. Two figures are necessary based on what type of fuel assemblies are in the reactor cores. The Tavg limits are established for unit operation from 5% to 100% Rated Thermal Power (ITS Mode 1 operation). The maximum Tavg for operation at 100% Rated Thermal Power is used to establish the maximum Tavg for unit operation between 5% and 100% power. Utilizing a Minimum Temperature for Criticality at 5% Rated Thermal Power, a linear progression is established for minimum Tavg up to 100% Rated Thermal Power. These limits are consistent with the changes in Tavg as power increases. The Bases was changed to remove the word "full power", since the values are for operation from 5% to 100% RTP. Additional administrative changes to the Bases have also been made to reflect the above discussion.	
	ITS:	**NUREG:**
	B 3.04.01	B 3.04.01
	COLR	N/A
	SR 3.04.01.02	SR 3.04.01.02

Justification For Deviations - NUREG-1431 Section 3.04.01

01-Aug-00

JFD Number	JFD Text	
03 Rev. A	NUREG-1431 SR 3.4.1.3, 12 hour verification of RCS total flow rate, is not being retained. PBNP does not currently perform this verification and does not have adequate control board mounted instrumentation that can be utilized to perform this verification.	
	ITS:	**NUREG:**
	B 3.04.01	B 3.04.01
	N/A	SR 3.04.01.03
	SR 3.04.01.03	SR 3.04.01.04
04 Rev. A	With the incorporation of TSTF-9 (relocation of SDM to COLR), the differences between LCO 3.1.1 and LCO 3.1.2 are removed and LCO 3.1.2 is incorporated into LCO 3.1.1, therefore subsequent Section 3.1 LCOs have been renumbered. Accordingly, the reference to LCOs 3.1.7 within the Bases has been revised, to reflect this change. This change is consistent with TSTF 136, which has been approved for incorporation into revision two of NUREG 1431.	
	ITS:	**NUREG:**
	B 3.04.01	B 3.04.01
05 Rev. B	PBNP utilizes LEFMs in determining feedwater flow for the precision heat balance and calibration of RCS total flow rate indicators. Therefore, the discussion of a penalty associated with the potential fouling of venturis is not retained in ITS. Accordingly, the changes to the Bases on this section done under TSTF 339 have not been incorporated.	
	ITS:	**NUREG:**
	B 3.04.01	B 3.04.01
06 Rev. A	The sentence was added to the bases to clarify that the THERMAL POWER ramp and step increase continue to be in effect until steady state conditions are established.	
	ITS:	**NUREG:**
	B 3.04.01	B 3.04.01

Approved TSTF 339, rev. 1

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.1 RCS Pressure, Temperature, and Flow Departure from Nucleate Boiling (DNB) Limits

s 1

LCO 3.4.1 RCS DNB parameters for pressurizer pressure, RCS average temperature, and RCS total flow rate shall be within the limits specified below:

a. Pressurizer pressure ≥ [2200] psig

is greater than or equal to the limit specified in the COLR

b. RCS average temperature ≤ [581]°F and

is ~~less than or equal to~~ the limit specified in the COLR

within 2

c. RCS total flow rate ≥ [284,000] gpm.

182,400 1

and greater than or equal to the limit specified in the COLR

Approved TSTF 339, rev. 1

s 2

B COLR

APPLICABILITY: MODE 1.

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

Pressurizer pressure limit does not apply during:

a. THERMAL POWER ramp > 5% RTP per minute; or

b. THERMAL POWER step > 10% RTP.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more RCS DNB parameters not within limits.	A.1 Restore RCS DNB parameter(s) to within limit.	2 hours
B. Required Action and associated Completion Time not met.	B.1 Be in MODE 2.	6 hours

1 2

B COLR

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.4.1.1	Verify pressurizer pressure is ≥ [2200] psig.	12 hours
SR 3.4.1.2	Verify RCS average temperature is ≤ [581]°F.	12 hours
~~SR 3.4.1.3~~	~~Verify RCS total flow rate is ≥ [284,000] gpm.~~	~~12 hours~~
SR 3.4.1.4	-------------------NOTE------------------- Not required to be performed until 24 hours after ≥ [90]% RTP. -- Verify by precision heat balance that RCS total flow rate is ≥ [284,000] gpm.	[18] months

greater than or equal to the limit specified in the COLR

s

within

Approved TSTF 339

less than or equal to the limit specified in the COLR

s

3

3 3

90 1

and greater than or equal to the limit specified in the COLR

Approved TSTF 339

182,400 18

1

B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.1 RCS Pressure, Temperature, and Flow Departure from Nucleate Boiling (DNB) Limits

BASES

BACKGROUND

The RCS average temperature limits are established for unit operation from 5% to 100% RTP. The maximum RCS average temperature for operation at 100% RTP is used to establish the maximum RCS average temperature for unit operation between 5% and 100% RTP. Utilizing a Minimum Temperature for Criticality at 5% RTP, a linear progression is established for minimum RCS average temperature up to 100% RTP.

2

These Bases address requirements for maintaining RCS pressure, temperature, and flow rate within limits assumed in the safety analyses. The safety analyses (Ref. 1) of normal operating conditions and anticipated operational occurrences assume initial conditions within the normal steady state envelope. The limits placed on RCS pressure, temperature, and flow rate ensure that the minimum departure from nucleate boiling ratio (DNBR) will be met for each of the transients analyzed.

B
RAI 3.4.1-2

The RCS pressure limit is consistent with operation within the nominal operational envelope. Pressurizer pressure indications are averaged to come up with a value for comparison to the limit. A lower pressure will cause the reactor core to approach DNB limits.

limits are 2

B
COLR

The RCS coolant average temperature ~~limit is~~ consistent with ~~full power~~ operation within the nominal operational envelope. Indications of temperature are averaged to determine a value for comparison to the limit. A higher average temperature will cause the core to approach DNB limits.

The RCS flow rate normally remains constant during an operational fuel cycle with all pumps running. The minimum RCS flow limit corresponds to that assumed for DNB analyses. Flow rate indications are averaged to come up with a value for comparison to the limit. A lower RCS flow will cause the core to approach DNB limits.

Operation for significant periods of time outside these DNB limits increases the likelihood of a fuel cladding failure in a DNB limited event.

APPLICABLE SAFETY ANALYSES

The requirements of this LCO represent the initial conditions for DNB limited transients analyzed in the plant safety analyses (Ref. 1). The safety analyses have shown that transients initiated from the limits of this LCO will

BASES

1.3

1

APPLICABLE SAFETY ANALYSES (continued)

result in meeting the DNBR criterion of ≥ [1.3]. This is the acceptance limit for the RCS DNB parameters. Changes to the unit that could impact these parameters must be assessed for their impact on the DNBR criteria. The transients analyzed for include loss of coolant flow events and dropped or stuck rod events. A key assumption for the analysis of these events is that the core power distribution is within the limits of LCO 3.1.[7] "Control Bank Insertion Limits"; LCO 3.2.3, "AXIAL FLUX DIFFERENCE (AFD)"; and LCO 3.2.4, "QUADRANT POWER TILT RATIO (QPTR)."

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Approved TSTF 339

specified in the COLR

Approved TSTF 339

The pressurizer pressure limit ~~of [2200] psig~~ and the RCS average temperature limit ~~of [581]°F~~ correspond to the analytical limits ~~of [2205] psig and [595]°F~~ used in the safety analyses, with allowance for measurement uncertainty.

The RCS DNB parameters satisfy Criterion 2 of the NRC Policy Statement.

Approved TSTF 339

LCO

Approved TSTF 339

These variables are contained in the COLR to provide operating and analysis flexibility from cycle to cycle. However, the minimum RCS flow, usually based on [maximum analyzed steam generator tube plugging], is retained in the TS LCO.

maximum analyzed steam generator tube plugging

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This LCO specifies limits on the monitored process variables—pressurizer pressure, RCS average temperature, and RCS total flow rate—to ensure the core operates within the limits assumed in the safety analyses. Operating within these limits will result in meeting the DNBR criterion in the event of a DNB limited transient.

RCS total flow rate contains a measurement error ~~of [2.0]%~~ based on performing a precision heat balance and using the result to calibrate the RCS flow rate indicators. ~~Potential fouling of the feedwater venturi, which might not be detected, could bias the result from the precision heat balance in a nonconservative manner. Therefore, a penalty of [0.1]% for undetected fouling of the feedwater venturi raises the nominal flow measurement allowance to [2.1]% for no fouling.~~

~~Any fouling that might bias the flow rate measurement greater than [0.1]% can be detected by monitoring and trending various plant performance parameters. If detected, either the effect of the fouling shall be quantified and compensated for in the RCS flow rate measurement or the venturi shall be cleaned to eliminate the fouling.~~

B

COLR

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(continued)

BASES

LCO (continued)

~~The LCO numerical values for pressure, temperature, and flow rate are given for the measurement location but have not been adjusted for instrument error.~~

The numerical values for pressure, temperature, and flow rate specified in the COLR are given for the measurement location and have been adjusted for instrument error.

Approved TSTF 339

6

The THERMAL POWER ramp and step increase continue to be in effect until steady state conditions are reached.

The DNBR limit

The conditions which define the DNBR limit

APPLICABILITY

In MODE 1, the limits on pressurizer pressure, RCS coolant average temperature, and RCS flow rate must be maintained during steady state operation in order to ensure DNBR criteria will be met in the event of an unplanned loss of forced coolant flow or other DNB limited transient. In all other MODES, the power level is low enough that DNB is not a concern.

B
COLR

A Note has been added to indicate the limit on pressurizer pressure is not applicable during short term operational transients such as a THERMAL POWER ramp increase > 5% RTP per minute or a THERMAL POWER step increase > 10% RTP. These conditions represent short term perturbations where actions to control pressure variations might be counterproductive. Also, since they represent transients initiated from power levels < 100% RTP, an increased DNBR margin exists to offset the temporary pressure variations.

~~Another set of limits on DNB related parameters~~ is provided in SL 2.1.1, "Reactor Core SLs." ~~Those limits~~ are less restrictive than the limits of this LCO, but violation of a Safety Limit (SL) merits a stricter, more severe Required Action. Should a violation of this LCO occur, the operator must check whether or not an SL may have been exceeded.

B
COLR

ACTIONS

A.1

RCS pressure and RCS average temperature are controllable and measurable parameters. With one or both of these parameters not within LCO limits, action must be taken to restore the parameter(s).

RCS total flow rate is not a controllable parameter and is not expected to vary during steady state operation. If the indicated RCS total flow rate is below the LCO limit, power must be reduced, as required by Required Action B.1, to restore DNB margin and eliminate the potential for violation of the accident analysis bounds.

No Significant Hazards Considerations - NUREG-1431 Section 3.04.01

03-Aug-00

NSHC Number	NSHC Text
A Rev. A	In accordance with the criteria set forth in 10 CFR 50.92, PBNP has evaluated this proposed Technical Specifications change and determined it does not represent a significant hazards consideration. The following is provided in support of this conclusion. 1. Does the change involve a significant increase in the probability or consequences of an accident previously evaluated? The proposed change involves reformatting and rewording of the current Technical Specifications. The reformatting and rewording process involves no technical changes to existing requirements. As such, this change is administrative in nature and does not impact initiators of analyzed events or assumed mitigation of accident or transient events. Therefore, this change does not increase the probability or consequences of an accident previously evaluated. 2. Does the change create the possibility of a new or different kind of accident from any accident previously evaluated? The proposed change does not require a physical alteration of the plant (no new or different type of equipment will be installed) or changes in parameters governing normal plant operation. The proposed change will not impose any new or eliminate any old requirements. Thus, this change does not create the possibility of a new or different kind of accident from any accident previously evaluated. 3. Does this change involve a significant reduction in a margin of safety? The proposed change will not significantly reduce the margin of safety because it has no impact on any safety analysis assumptions. This change is administrative. As such, there is no technical change to the requirements and, therefore, there is no reduction in the margin of safety.

NSHC Number	NSHC Text
L.01 Rev. A	In accordance with the criteria set forth in 10 CFR 50.92, PBNP has evaluated this proposed Technical Specifications change and determined it does not represent a significant hazards consideration. The following is provided in support of this conclusion. 1. Does the change involve a significant increase in the probability or consequences of an accident previously evaluated? The proposed change 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 adds a Note to indicate the limit on pressurizer pressure is not applicable during short term operational transients such as a THERMAL POWER ramp increase > 5% RTP per minute or a THERMAL POWER step increase > 10% RTP. This change is acceptable since these conditions represent short term perturbations where actions to control pressure variations might be counterproductive. 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. Short term operational transients such as a THERMAL POWER ramp increase > 5% RTP per minute or a THERMAL POWER step increase > 10% RTP, represent transients initiated from power levels < 100% RTP, where an increased DNBR margin exists to offset the temporary pressure variations. Therefore, this change does not involve a significant reduction in a margin of safety.

NSHC Number	NSHC Text
L.02 Rev. B	In accordance with the criteria set forth in 10 CFR 50.92, PBNP has evaluated this proposed Technical Specifications change and determined it does not represent a significant hazards consideration. The following is provided in support of this conclusion. The CTS does not provide actions in the event the DNB parameters are not maintained within limits. Therefore CTS 15.3.0.B requires action be initiated within 1 hour to place the plant in a condition whereby the specification does not apply. The proposed ITS LCO will require restoration of the DNB parameter(s) within 2 hours, or be in MODE 2 in 8 hours, in the event the DNB parameters are not maintained within limits. 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 extends the time from 1 hour to 2 hours to restore DNB parameters to within limits. This change is acceptable in order to provide sufficient time to adjust plant parameters, to determine the cause for the off normal condition, and to restore the readings to within limits. 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 introduce a new mode of operation or alter the method of normal plant operation. Therefore, the possibility of a new or different kind of accident from any accident previously evaluated is not created. 3. Does this change involve a significant reduction in a margin of safety? The proposed changes do not alter any assumed conditions or limitation in any previously evaluated accidents. Therefore, this change does not involve a significant reduction in a margin of safety.

NSHC Number	NSHC Text
LA Rev. B	In accordance with the criteria set forth in 10 CFR 50.92, PBNP has evaluated this proposed Technical Specifications change and determined it does not represent a significant hazards consideration. The following is provided in support of this conclusion. 1. Does the change involve a significant increase in the probability or consequences of an accident previously evaluated? The proposed change 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. These processes will ensure that appropriate margins of safety are maintained or required approval of changes obtained. Therefore, these changes will not result in a significant reduction in a 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 are consistent with assumptions made in the safety analysis. Thus, this change does not create the possibility of a new or different kind of accident from any accident previously evaluated. 3. Does this change involve a significant reduction in a margin of safety? The imposition of more restrictive requirements either has no affect on or increases the margin of safety. Each change is providing additional restrictions to enhance plant safety. These changes are consistent with the safety analysis. Therefore, this change does not involve a reduction in a margin of safety.

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.1 RCS Pressure, Temperature, and Flow Departure from Nucleate Boiling (DNB) Limits

LCO 3.4.1 RCS DNB parameters for pressurizer pressure, RCS average temperature, and RCS total flow rate shall be within the limits specified below:

a. Pressurizer pressure is greater than or equal to the limits specified in the COLR;

b. RCS average temperature is within the limits specified in the COLR; and

c. RCS total flow rate ≥ 182,400 gpm and greater than or equal to the limit specified in the COLR.

B COLR

APPLICABILITY: MODE 1.

-----------------------------------NOTE-----------------------------------
Pressurizer pressure limit does not apply during:

a. THERMAL POWER ramp > 5% RTP per minute; or

b. THERMAL POWER step > 10% RTP.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more RCS DNB parameters not within limits.	A.1 Restore RCS DNB parameter(s) to within limit.	2 hours
B. Required Action and associated Completion Time not met.	B.1 Be in MODE 2.	6 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.4.1.1	Verify pressurizer pressure is greater than or equal to the limits specified in the COLR.	12 hours
SR 3.4.1.2	Verify RCS average temperature is within the limits specified in the COLR.	12 hours
SR 3.4.1.3	-------------------NOTE------------------- Not required to be performed until 24 hours after ≥ 90% RTP. -- Verify by precision heat balance that RCS total flow rate is ≥ 182,400 gpm and greater than or equal to the limit specified in the COLR.	18 months

B
COLR

B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.1 RCS Pressure, Temperature, and Flow Departure from Nucleate Boiling (DNB) Limits

BASES

BACKGROUND

These Bases address requirements for maintaining RCS pressure, temperature, and flow rate within limits assumed in the safety analyses. The safety analyses (Ref. 1) of normal operating conditions and anticipated operational occurrences assume initial conditions within the normal steady state envelope. The limits placed on RCS pressure, temperature, and flow rate ensure that the minimum departure from nucleate boiling ratio (DNBR) will be met for each of the transients analyzed.

The RCS pressure limit is consistent with operation within the nominal operational envelope. Pressurizer pressure indications are averaged to come up with a value for comparison to the limit. A lower pressure will cause the reactor core to approach DNB limits.

The RCS average temperature limits are established for unit operation from 5% to 100% RTP. The maximum RCS average temperature for operation at 100% RTP is used to establish the maximum RCS average temperature for unit operation between 5% and 100% RTP. Utilizing a Minimum Temperature for Criticality at 5% RTP, a linear progression is established for minimum RCS average temperature up to 100% RTP. B RAI 3.4.1-2

B COLR

The RCS coolant average temperature limits are consistent with operation within the nominal operational envelope. Indications of temperature are averaged to determine a value for comparison to the limit. A higher average temperature will cause the core to approach DNB limits.

The RCS flow rate normally remains constant during an operational fuel cycle with all pumps running. The minimum RCS flow limit corresponds to that assumed for DNB analyses. Flow rate indications are averaged to come up with a value for comparison to the limit. A lower RCS flow will cause the core to approach DNB limits.

Operation for significant periods of time outside these DNB limits increases the likelihood of a fuel cladding failure in a DNB limited event.

APPLICABLE SAFETY ANALYSES

The requirements of this LCO represent the initial conditions for DNB limited transients analyzed in the plant safety analyses (Ref. 1). The safety analyses have shown that transients initiated from the limits of this LCO will result in meeting the DNBR criterion of ≥ 1.3. This is the acceptance limit for the RCS DNB parameters. Changes to the unit

BASES

APPLICABLE SAFETY ANALYSES (continued)

that could impact these parameters must be assessed for their impact on the DNBR criteria. The transients analyzed for include loss of coolant flow events and dropped or stuck rod events. A key assumption for the analysis of these events is that the core power distribution is within the limits of LCO 3.1.6, "Control Bank Insertion Limits"; LCO 3.2.3, "AXIAL FLUX DIFFERENCE (AFD)"; and LCO 3.2.4, "QUADRANT POWER TILT RATIO (QPTR) ."

The pressurizer pressure limit and the RCS average temperature limit specified in the COLR correspond to the analytical limits used in the safety analyses, with allowance for measurement uncertainty.

The RCS DNB parameters satisfy Criterion 2 of the NRC Policy Statement.

LCO

This LCO specifies limits on the monitored process variables - pressurizer pressure, RCS average temperature, and RCS total flow rate - to ensure the core operates within the limits assumed in the safety analyses. These variables are contained in the COLR to provide operating and analysis flexibility from cycle to cycle. However, the minimum RCS flow, usually based on maximum analyzed steam generator tube plugging, is retained in the TS LCO. Operating within these limits will result in meeting the DNBR criterion in the event of a DNB limited transient.

RCS total flow rate contains a measurement error based on performing a precision heat balance and using the result to calibrate the RCS flow rate indicators.

The numerical values for pressure, temperature, and flow rate specified in the COLR are given for the measurement location and have not been adjusted for instrument error.

APPLICABILITY

In MODE 1, the limits on pressurizer pressure, RCS coolant average temperature, and RCS flow rate must be maintained during steady state operation in order to ensure DNBR criteria will be met in the event of an unplanned loss of forced coolant flow or other DNB limited transient. In all other MODES, the power level is low enough that DNB is not a concern.

A Note has been added to indicate the limit on pressurizer pressure is not applicable during short term operational transients such as a THERMAL POWER ramp increase > 5% RTP per minute or a THERMAL POWER step increase > 10% RTP. The THERMAL POWER ramp and step increase continue to be in effect until steady state conditions are reached. These conditions represent short term perturbations where actions to control pressure variations might be

BASES

APPLICABILITY (continued)

counterproductive. Also, since they represent transients initiated from power levels < 100% RTP, an increased DNBR margin exists to offset the temporary pressure variations.

The DNBR limit is provided in SL 2.1.1, "Reactor Core SLs." The conditions which define the DNBR limit are less restrictive than the limits of this LCO, but violation of a Safety Limit (SL) merits a stricter, more severe Required Action. Should a violation of this LCO occur, the operator must check whether or not an SL may have been exceeded. B COLR

ACTIONS

A.1

RCS pressure and RCS average temperature are controllable and measurable parameters. With one or both of these parameters not within LCO limits, action must be taken to restore the parameter(s).

RCS total flow rate is not a controllable parameter and is not expected to vary during steady state operation. If the indicated RCS total flow rate is below the LCO limit, power must be reduced, as required by Required Action B.1, to restore DNB margin and eliminate the potential for violation of the accident analysis bounds.

The 2 hour Completion Time for restoration of the parameters provides sufficient time to adjust plant parameters, to determine the cause for the off normal condition, and to restore the readings within limits, and is based on plant operating experience.

B.1

If Required Action A.1 is not met within the associated Completion Time, the plant must be brought to a MODE in which the LCO does not apply. To achieve this status, the plant must be brought to at least MODE 2 within 6 hours. In MODE 2, the reduced power condition eliminates the potential for violation of the accident analysis bounds. The Completion Time of 6 hours is reasonable to reach the required plant conditions in an orderly manner.

SURVEILLANCE REQUIREMENTS

SR 3.4.1.1

Since Required Action A.1 allows a Completion Time of 2 hours to restore parameters that are not within limits, the 12 hour Surveillance Frequency for pressurizer pressure is sufficient to ensure the pressure can be restored to a normal operation, steady state condition following load changes and other expected transient operations. The 12 hour

BASES

SURVEILLANCE REQUIREMENTS (continued)

interval has been shown by operating practice to be sufficient to regularly assess for potential degradation and to verify operation is within safety analysis assumptions.

SR 3.4.1.2

Since Required Action A.1 allows a Completion Time of 2 hours to restore parameters that are not within limits, the 12 hour Surveillance Frequency for RCS average temperature is sufficient to ensure the temperature can be restored to a normal operation, steady state condition following load changes and other expected transient operations. The 12 hour interval has been shown by operating practice to be sufficient to regularly assess for potential degradation and to verify operation is within safety analysis assumptions.

SR 3.4.1.3

Measurement of RCS total flow rate by performance of a precision calorimetric heat balance once every 18 months allows the installed RCS flow instrumentation to be calibrated and verifies the actual RCS flow rate is greater than or equal to the minimum required RCS flow rate.

The Frequency of 18 months reflects the importance of verifying flow after a refueling outage when the core has been altered, which may have caused an alteration of flow resistance.

This SR is modified by a Note that allows entry into MODE 1, without having performed the SR, and placement of the unit in the best condition for performing the SR. The Note states that the SR is not required to be performed until 24 hours after ≥ 90% RTP. This exception is appropriate since the heat balance requires the plant to be at a minimum of 90% RTP to obtain the stated RCS flow accuracies. The Surveillance shall be performed within 24 hours after reaching 90% RTP.

REFERENCES

1. FSAR. Section 14.

Justification For Deviations - NUREG-1431 Section 3.04.05

01-Aug-00

JFD Number	JFD Text	
03 Rev. A	The wording of the LCO 3.4.5 Note and Bases was changed from "...may be de-energized..." to "...may not be in operation...", per approved TSTF 153. However, "...may not be in operation..." could easily be interpreted to imply a condition that forbids RCP operation. To prevent this misunderstanding, the wording has been changed to, "...may be not in operation..."	
	ITS:	**NUREG:**
	B 3.04.05	B 3.04.05
	LCO 3.04.05 NOTE	LCO 3.04.05 NOTE
04 Rev. A	With the RTB's in the closed position and Rod Control System capable of rod withdrawal, accidental control rod withdrawal from subcritical is postulated and requires one RCS loop to be OPERABLE and in operation to ensure that the accident analysis limits are met. This analysis is, therefore, bounded by the decay heat removal redundancy requirements. Therefore, the requirement for the Rod Control System to be made incapable of rod withdrawal is necessary to prevent an inadvertent control rod withdrawal and the potential heat input to the reactor coolant with neither RCP in operation.	
	ITS:	**NUREG:**
	B 3.04.05	B 3.04.05
	LCO 3.04.05 NOTE	LCO 3.04.05 NOTE LCO 3.04.05 NOTE N/A
05 Rev. B	Information regarding the performance of rod drop tests under no flow conditions is being deleted from the LCO 3.4.5 Bases. Point Beach has no requirement to perform this test and, therefore, need not be discussed as a reason for allowing both RCP's to be de-energized for up to 1 hour in an 8 hour period in Mode 3.	
	ITS:	**NUREG:**
	B 3.04.05	B 3.04.05
06 Rev. A	LCO 3.9.2 "Unborated Water Source Isolation Valves" was not adopted based on the Point Beach design. Accordingly, the references to LCO 3.9.5 and 6 have been revised to reflect the renumbering that has occurred in ITS Section 3.9.	
	ITS:	**NUREG:**
	B 3.04.05	B 3.04.05

Justification For Deviations - NUREG-1431 Section 3.04.05

01-Aug-00

JFD Number	JFD Text	
07 Rev. A	A sentence has been added to the LCO 3.4.5 Bases to clarify that the OPERABLE RCP and SG must be in the same loop for the RCS loop to be considered OPERABLE. This sentence was added because the NUREG-1431 Bases did not specify this condition for an OPERABLE RCS loop, and this condition was considered to be a necessary attribute for Point Beach.	
	ITS:	**NUREG:**
	B 3.04.05	B 3.04.05
08 Rev. B	"Narrow range" was added to the description of the required secondary side water level of the steam generators. NUREG-1431 did not specify a level indication instrumentation reference for the steam generator water level. To avoid possible interpretation, "narrow range" was added to specify that the required steam generator water level percentage is indicated narrow range. 30% narrow range level indication is a much higher water level (i.e. more conservative) than 30% wide range indication and ensures that the steam generator tubes are covered.	
	ITS:	**NUREG:**
	SR 3.04.05.02	SR 3.04.05.02

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.4.5.2	Verify steam generator secondary side water levels are ≥ [17]% for required RCS loops. 1 30 narrow range 8	12 hours
SR 3.4.5.3	Verify correct breaker alignment and indicated power are available to the required pump that is not in operation.	7 days

B
RAI 3.4.7-3

BASES

APPLICABLE SAFETY ANALYSES (continued)

met. For those conditions when the Rod Control System is not capable of rod withdrawal, two RCS loops are required to be OPERABLE, but only one RCS loop is required to be in operation to be consistent with MODE 3 accident analyses.

Failure to provide decay heat removal may result in challenges to a fission product barrier. The RCS loops are part of the primary success path that functions or actuates to prevent or mitigate a Design Basis Accident or transient that either assumes the failure of, or presents a challenge to, the integrity of a fission product barrier.

RCS Loops – MODE 3 satisfy Criterion 3 of the NRC Policy Statement.

two 1

LCO

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The purpose of this LCO is to require that at least [two] RCS loops be OPERABLE. In MODE 3 with the ~~RTBs in the closed position and~~ Rod Control System capable of rod withdrawal, [two] RCS loop~~s~~ must be in operation. [Two] RCS loop~~s~~ are required to be in operation in MODE 3 with ~~RTBs closed and~~ Rod Control System capable of rod withdrawal due to the postulation of a power excursion because of an inadvertent control rod withdrawal. The required number of RCS loops in operation ensures that the Safety Limit criteria will be met for all of the postulated accidents.

Approved TSTF-87 R.2

When

~~With the RTBs in the open position, or the CRDMs de-energized,~~ the Rod Control System is not capable of rod withdrawal~~; therefore,~~ only one RCS loop in operation is necessary to ensure removal of decay heat from the core and homogenous boron concentration throughout the RCS. An additional RCS loop is required to be OPERABLE to ensure that safety analyses limits are met.

3 be not in operation

not be in operation Approved TSTF 153

The Note permits all RCPs to be de-energized for ≤ 1 hour per 8 hour period. The purpose of the Note is to perform tests that are designed to validate various accident analyses values.

5 An example of one

One of these tests is validation of the pump coastdown curve used as input to a number of accident analyses including a loss of flow accident. This test is generally performed in MODE 3 during the initial startup testing program, and as such should only be performed once. If, however, changes are made to the RCS that would cause a change to the flow characteristics of the RCS, the input

B
RAI 3.4.5-1

(continued)

BASES

LCO (continued)

values of the coastdown curve must be revalidated by conducting the test again. ~~Another test performed during the startup testing program is the validation of rod drop times during cold conditions, both with and without flow.~~

B
RAI 3.4.5-1

5

~~The no flow test may be performed in MODE 3, 4, or 5 and requires that the pumps be stopped for a short period of time. The Note permits the de-energizing of the pumps in order to perform this test and validate the assumed analysis values. As with the validation of the pump coastdown curve, this test should be performed only once unless the flow characteristics of the RCS are changed.~~ The 1 hour time period specified is adequate to perform the desired tests, and operating experience has shown that boron stratification is not a problem during this short period with no forced flow.

Utilization of the Note is permitted provided the following conditions are met, along with any other conditions imposed by initial startup test procedures:

a. No operations are permitted that would dilute the RCS boron concentration, thereby maintaining the margin to criticality. Boron reduction is prohibited because a uniform concentration distribution throughout the RCS cannot be ensured when in natural circulation; ~~and~~

b. Core outlet temperature is maintained at least 10 °F below saturation temperature, so that no vapor bubble may form and possibly cause a natural circulation flow obstruction~~.~~ ; and

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c. The Rod Control System is not capable of rod withdrawal, to preclude the possibility of an inadvertent control rod withdrawal and associated power excursion.

An OPERABLE RCS loop consists of one OPERABLE RCP and one OPERABLE SG in accordance with the Steam Generator Tube Surveillance Program, which has the minimum water level specified in SR 3.4.5.2. The OPERABLE RCP and SG must be in the same loop for the RCS loop to be considered OPERABLE. An RCP is OPERABLE if it is capable of being powered and is able to provide forced flow if required.

7

APPLICABILITY

In MODE 3, this LCO ensures forced circulation of the reactor coolant to remove decay heat from the core and to provide proper boron mixing. ~~The most stringent condition of the LCO, that is, two RCS loops OPERABLE and two RCS loops in operation, applies to MODE 3 with RTBs in the~~ One RCS loop provides sufficient circulation for these purposes. However, one additional RCS loop is required to be OPERABLE to ensure redundant capability for decay heat removal.

2

(continued)

CONDITION	REQUIRED ACTION		COMPLETION TIME
C. Two RCS loops inoperable. OR No RCS loop in operation.	C.1	Place the Rod Control System in a condition incapable of rod withdrawal.	Immediately
	AND		
	C.2	Suspend all operations involving a reduction of RCS boron concentration.	Immediately
	AND		
	C.3	Initiate action to restore one RCS loop to OPERABLE status and operation.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.4.5.1	Verify one RCS loop is in operation.	12 hours
SR 3.4.5.2	Verify steam generator secondary side water levels are $\geq$ 30% narrow range for required RCS loops.	12 hours
SR 3.4.5.3	Verify correct breaker alignment and indicated power are available to the required pump that is not in operation.	7 days

B
RAI 3.4.7-3

B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.5 RCS Loops - MODE 3

BASES

BACKGROUND

In MODE 3, the primary function of the reactor coolant is removal of decay heat and transfer of this heat, via the steam generator (SG), to the secondary plant fluid. The secondary function of the reactor coolant is to act as a carrier for soluble neutron poison, boric acid.

The reactor coolant is circulated through two RCS loops, connected in parallel to the reactor vessel, each containing an SG, a reactor coolant pump (RCP), and appropriate flow, pressure, level, and temperature instrumentation for control, protection, and indication. The reactor vessel contains the clad fuel. The SGs provide the heat sink. The RCPs circulate the water through the reactor vessel and SGs at a sufficient rate to ensure proper heat transfer and prevent fuel damage.

In MODE 3, RCPs are used to provide forced circulation for heat removal during heatup and cooldown. The MODE 3 decay heat removal requirements are low enough that a single RCS loop with one RCP running is sufficient to remove core decay heat. However, two RCS loops are required to be OPERABLE to ensure redundant capability for decay heat removal.

APPLICABLE SAFETY ANALYSES

Whenever the reactor trip breakers (RTBs) are in the closed position and the control rod drive mechanisms (CRDMs) are energized, an inadvertent rod withdrawal from subcritical, resulting in a power excursion, is possible. Such a transient could be caused by a malfunction of the rod control system. In addition, the possibility of a power excursion due to the ejection of an inserted control rod is possible with the breakers closed or open. Such a transient could be caused by the mechanical failure of a CRDM.

Therefore, in MODE 3 with the Rod Control System capable of rod withdrawal, accidental control rod withdrawal from subcritical is postulated and requires at least one RCS loop to be OPERABLE and in operation to ensure that the accident analyses limits are met. For those conditions when the Rod Control System is not capable of rod withdrawal, two RCS loops are required to be OPERABLE, but only one RCS loop is required to be in operation to be consistent with MODE 3 accident analyses.

BASES

APPLICABLE SAFETY ANALYSES (continued)	Failure to provide decay heat removal may result in challenges to a fission product barrier. The RCS loops are part of the primary success path that functions or actuates to prevent or mitigate a Design Basis Accident or transient that either assumes the failure of, or presents a challenge to, the integrity of a fission product barrier. RCS Loops - MODE 3 satisfy Criterion 3 of the NRC Policy Statement.

LCO

The purpose of this LCO is to require that at least two RCS loops be OPERABLE. In MODE 3 with the Rod Control System capable of rod withdrawal, one RCS loop must be in operation. One RCS loop is required to be in operation in MODE 3 with the Rod Control System capable of rod withdrawal due to the postulation of a power excursion because of an inadvertent control rod withdrawal. The required number of RCS loops in operation ensures that the Safety Limit criteria will be met for all of the postulated accidents.

When the Rod Control System is not capable of rod withdrawal only one RCS loop in operation is necessary to ensure removal of decay heat from the core and homogenous boron concentration throughout the RCS. An additional RCS loop is required to be OPERABLE to ensure that safety analyses limits are met.

The Note permits all RCPs to be not in operation for ≤ 1 hour per 8 hour period. The purpose of the Note is to perform tests that are designed to validate various accident analyses values. An example of one of these tests is validation of the pump coastdown curve used as input to a number of accident analyses including a loss of flow accident. This test is generally performed in MODE 3 during the initial startup testing program, and as such should only be performed once. If, however, changes are made to the RCS that would cause a change to the flow characteristics of the RCS, the input values of the coastdown curve must be revalidated by conducting the test again. B RAI 3.4.5-1

The 1 hour time period specified is adequate to perform the desired tests, and operating experience has shown that boron stratification is not a problem during this short period with no forced flow.

Utilization of the Note is permitted provided the following conditions are met, along with any other conditions imposed by initial startup test procedures:

a. No operations are permitted that would dilute the RCS boron concentration, thereby maintaining the margin to criticality. Boron reduction is prohibited because a uniform concentration distribution throughout the RCS cannot be ensured when in natural circulation;

BASES

LCO (continued)

b. Core outlet temperature is maintained at least 10°F below saturation temperature, so that no vapor bubble may form and possibly cause a natural circulation flow obstruction; and

c. The Rod Control System is not capable of rod withdrawal, to preclude the possibility of an inadvertent control rod withdrawal and associated power excursion.

An OPERABLE RCS loop consists of one OPERABLE RCP and one OPERABLE SG in accordance with the Steam Generator Tube Surveillance Program, which has the minimum water level specified in SR 3.4.5.2. The OPERABLE RCP and SG must be in the same loop for the RCS loop to be considered OPERABLE. An RCP is OPERABLE if it is capable of being powered and is able to provide forced flow if required.

APPLICABILITY

In MODE 3, this LCO ensures forced circulation of the reactor coolant to remove decay heat from the core and to provide proper boron mixing. One RCS loop provides sufficient circulation for these purposes. However, one additional RCS loop is required to be OPERABLE to ensure redundant capability for decay heat removal.

Operation in other MODES is covered by:

LCO 3.4.4, "RCS Loops - MODES 1 and 2";
LCO 3.4.6, "RCS Loops - MODE 4";
LCO 3.4.7, "RCS Loops - MODE 5, Loops Filled";
LCO 3.4.8, "RCS Loops - MODE 5, Loops Not Filled";
LCO 3.9.4, "Residual Heat Removal (RHR) and Coolant Circulation-High Water Level" (MODE 6); and
LCO 3.9.5, "Residual Heat Removal (RHR) and Coolant Circulation-Low Water Level" (MODE 6).

ACTIONS

A.1

If one required RCS loop is inoperable, redundancy for heat removal is lost. The Required Action is restoration of the required RCS loop to OPERABLE status within the Completion Time of 72 hours. This time allowance is a justified period to be without the redundant, nonoperating loop because a single loop in operation has a heat transfer capability greater than that needed to remove the decay heat produced in the reactor core and because of the low probability of a failure in the remaining loop occurring during this period.

BASES

ACTIONS (continued)

B.1

If restoration is not possible within 72 hours, the unit must be brought to MODE 4. In MODE 4, the unit may be placed on the Residual Heat Removal System. The additional Completion Time of 12 hours is compatible with required operations to achieve cooldown and depressurization from the existing plant conditions in an orderly manner and without challenging plant systems.

C.1, C.2, and C.3

If two RCS loops are inoperable or no RCS loop is in operation, except as during conditions permitted by the Note in the LCO section, place the Rod Control System in a condition incapable of rod motion (e.g., CRDMs must be de-energized by opening the RTBs or de-energizing the MG sets). All operations involving a reduction of RCS boron concentration must be suspended, and action to restore one of the RCS loops to OPERABLE status and operation must be initiated. Boron dilution requires forced circulation for proper mixing, and opening the RTBs or de-energizing the MG sets removes the possibility of an inadvertent rod withdrawal. The immediate Completion Time reflects the importance of maintaining operation for heat removal. The action to restore must be continued until one loop is restored to OPERABLE status and operation.

SURVEILLANCE REQUIREMENTS

SR 3.4.5.1

This SR requires verification every 12 hours that one RCS loop is in operation. Verification includes flow rate, temperature, and pump status monitoring, which help ensure that forced flow is providing heat removal. The Frequency of 12 hours is sufficient considering other indications and alarms available to the operator in the control room to monitor RCS loop performance.

SR 3.4.5.2

SR 3.4.5.2 requires verification of SG OPERABILITY.
SG OPERABILITY is verified by ensuring that the secondary side narrow range water level is ≥ 30% for required RCS loops. If the SG secondary side narrow range water level is < 30%, the tubes may become uncovered and the associated loop may not be capable of providing the heat sink for removal of the decay heat. The 12 hour Frequency is considered adequate in view of other indications available in the control room to alert the operator to a loss of SG level.

BASES

SURVEILLANCE REQUIREMENTS (continued)	SR 3.4.5.3 Verification that the required RCPs are OPERABLE ensures that safety analyses limits are met. The requirement also ensures that an additional RCP can be placed in operation, if needed, to maintain decay heat removal and reactor coolant circulation. Verification is performed by verifying proper breaker alignment and power availability to the required RCPs.
REFERENCES	None.

Justification For Deviations - NUREG-1431 Section 3.04.06

01-Aug-00

JFD Number	JFD Text	
01 Rev. A	The wording of the LCO 3.4.6, Note 1, and associated Bases was changed from "...may be de-energized..." to "...may not be in operation...", per approved TSTF 153. However, "...may not be in operation..." could easily be interpreted to imply a condition that forbids RCP operation. To prevent this misunderstanding, the wording has been changed to, "...may be not in operation..."	
	ITS:	**NUREG:**
	B 3.04.06	B 3.04.06
	LCO 3.04.06 NOTE 1	LCO 3.04.06 NOTE 1
02 Rev. A	The actual numerical values for LTOP enabling temperature are replaced with a reference to the temperature specified in the PTLR. The LTOP enabling temperature will then be calculated and controlled by the licensee in accordance with the topical reports identified in the PTLR.	
	ITS:	**NUREG:**
	B 3.04.06	B 3.04.06
	LCO 3.04.06 NOTE 2	LCO 3.04.06 NOTE 2
03 Rev. A	The brackets have been removed and the proper plant specific information has been provided.	
	ITS:	**NUREG:**
	B 3.04.06	B 3.04.06
	LCO 3.04.06 NOTE 2	LCO 3.04.06 NOTE 2
	SR 3.04.06.02	SR 3.04.06.02
04 Rev. B	NUREG-1431, LCO 3.4.6 Bases description of startup testing is revised to reflect the actual testing performed at PBNP. Per CTS 15.4.1, Table 15.4.1-2, Note (3), the rod drop test is only performed at rated reactor coolant flow. Therefore, this specific example for testing is replaced with one that may be used at Point Beach.	
	ITS:	**NUREG:**
	B 3.04.06	B 3.04.06
05 Rev. A	LCO 3.9.2 "Unborated Water Source Isolation Valves" was not adopted, based on the Point Beach design. Accordingly, the references to LCO 3.9.5 and 6 within the Bases for LCO 3.4.6 have been revised to reflect the renumbering that has occurred in Section 3.9 of the ITS.	
	ITS:	**NUREG:**
	B 3.04.06	B 3.04.06

Justification For Deviations - NUREG-1431 Section 3.04.06

01-Aug-00

JFD Number	JFD Text	
06 Rev. A	LCO 3.4.6 Bases, Action B.1, provides a temperature band of 200 to 300 F, for MODE 4. This band has been revised to 200 to 350 F, to more closely coincide with the Section 1.1 definition of MODE 4.	
	ITS:	**NUREG:**
	B 3.04.06	B 3.04.06
07 Rev. A	A sentence has been added to the LCO 3.4.6 Bases to clarify that the OPERABLE RCP and SG must be in the same loop for the RCS loop to be considered OPERABLE. This sentence was added because the NUREG-1431 Bases did not specify this condition for an OPERABLE RCS loop, and this condition was considered to be a necessary attribute for Point Beach.	
	ITS:	**NUREG:**
	B 3.04.06	B 3.04.06
08 Rev. B	"Narrow range" was added to the description of the required secondary side water level of the steam generators. NUREG-1431 did not specify a level indication instrumentation reference for the steam generator water level. To avoid possible interpretation, "narrow range" was added to specify that the required steam generator water level percentage is indicated narrow range. 30% narrow range level indication is a much higher water level (i.e. more conservative) than 30% wide range indication and ensures that the steam generator tubes are covered.	
	ITS:	**NUREG:**
	SR 3.04.06.02	SR 3.04.06.02
09 Rev. B	A sentence has been added to the LCO 3.4.6 Bases to clarify that SG secondary side water temperature can be closely approximated by using the SG metal temperature indicator. This method is necessary due to the Point Beach design which does not include instrumentation from which a direct indication of the SG secondary side water temperature can be obtained.	
	ITS:	**NUREG:**
	B 3.04.06	B 3.04.06

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
B. One required RHR loop inoperable. AND Two required RCS loops inoperable.	B.1 Be in MODE 5.	24 hours
C. Required RCS or RHR loops inoperable. OR No RCS or RHR loop in operation.	C.1 Suspend all operations involving a reduction of RCS boron concentration. AND C.2 Initiate action to restore one loop to OPERABLE status and operation.	Immediately Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.4.6.1	Verify one RHR or RCS loop is in operation.	12 hours
SR 3.4.6.2	Verify SG secondary side water levels are ≥ [17]% narrow range for required RCS loops. 30 3 8	12 hours

B
RAI 3.4.7-3

(continued)

Approved TSTF 153

BASES

LCO (continued)

1 be not in operation

loops and RHR loops. Any one loop in operation provides enough flow to remove the decay heat from the core with forced circulation. An additional loop is required to be OPERABLE to provide redundancy for heat removal.

not be in operation

4

An example of one of the tests is validation of the pump coastdown curve used as input to a number of accident analyses including a loss of flow accident. This test is generally performed during the initial startup testing program, and as such should only be performed once.

Note 1 permits all RCPs or RHR pumps to be de -energized for ≤ 1 hour per 8 hour period. The purpose of the Note is to permit tests that are designed to validate various accident analyses values. ~~One of the tests performed during the startup testing program is the validation of rod drop times during cold conditions, both with and without flow. The no flow test may be performed in MODE 3, 4, or 5 and requires that the pumps be stopped for a short period of time. The Note permits the de-energizing of the pumps in order to perform this test and validate the assumed analysis values.~~ If changes are made to the RCS that would cause a change to the flow characteristics of the RCS, the input values must be revalidated by conducting the test again. The 1 hour time period is adequate to perform the test, and operating experience has shown that boron stratification is not a problem during this short period with no forced flow.

B

RAI 3.4.6-2

Utilization of Note 1 is permitted provided the followin g conditions are met along with any other conditions imposed by initial startup test procedures:

a. No operations are permitted that would dilute the RCS boron concentration, therefore maintaining the margin to criticality. Boron reduction is prohibited because a uniform concentration distribution throughout the RCS cannot be ensured when in natural circulation; and

b. Core outlet temperature is maintained at least 10°F below saturation temperature, so that no vapor bubble may form and possibly cause a natural circulation flow obstruction.

3 50

Note 2 requires that the secondary side water temperature of each SG be ≤ [50] °F above each of the RCS cold leg temperatures before the start of an RCP with any RCS cold leg temperature ≤ [275°F]. This restrai nt is to prevent a low temperature overpressure event due to a thermal transient when an RCP is started.

the Low Temperature Overpressure Protection (LTOP) enabling temperature specified in the PTLR

2

An OPERABLE RCS loop comprises an OPERABLE RCP and an OPERABLE SG in accordance with the Steam Generator Tube

SG secondary side water temperature can be approximated by using the SG metal temperature indicator.

9

B

ACTIONS (continued)

CONDITION	REQUIRED ACTION		COMPLETION TIME
B. One required RHR loop inoperable. AND Two required RCS loops inoperable.	B.1	Be in MODE 5.	24 hours
C. Required RCS or RHR loops inoperable. OR No RCS or RHR loop in operation.	C.1	Suspend all operations involving a reduction of RCS boron concentration.	Immediately
	AND		
	C.2	Initiate action to restore one loop to OPERABLE status and operation.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.4.6.1	Verify one RHR or RCS loop is in operation.	12 hours
SR 3.4.6.2	Verify SG secondary side water levels are ≥ 30% narrow range for required RCS loops.	12 hours
SR 3.4.6.3	Verify correct breaker alignment and indicated power are available to the required pump that is not in operation.	7 days

B
RAI 3.4.7-3

B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.6 RCS Loops - MODE 4

BASES

BACKGROUND

In MODE 4, the primary function of the reactor coolant is the removal of decay heat and the transfer of this heat to either the steam generator (SG) secondary side coolant or the component cooling water via the residual heat removal (RHR) heat exchangers. The secondary function of the reactor coolant is to act as a carrier for soluble neutron poison, boric acid.

The reactor coolant is circulated through two RCS loops connected in parallel to the reactor vessel, each loop containing an SG, a reactor coolant pump (RCP), and appropriate flow, pressure, level, and temperature instrumentation for control, protection, and indication. The RCPs circulate the coolant through the reactor vessel and SGs at a sufficient rate to ensure proper heat transfer and to prevent boric acid stratification.

In MODE 4, either RCPs or RHR loops can be used to provide forced circulation. The intent of this LCO is to provide forced flow from at least one RCP or one RHR loop for decay heat removal and transport. The flow provided by one RCP loop or RHR loop is adequate for decay heat removal. The other intent of this LCO is to require that two paths be available to provide redundancy for decay heat removal.

APPLICABLE SAFETY ANALYSES

In MODE 4, RCS circulation is considered in the determination of the time available for mitigation of the accidental boron dilution event. The RCS and RHR loops provide this circulation.

RCS Loops — MODE 4 have been identified in the NRC Policy Statement as important contributors to risk reduction.

LCO

The purpose of this LCO is to require that at least two loops be OPERABLE in MODE 4 and that one of these loops be in operation. The LCO allows the two loops that are required to be OPERABLE to consist of any combination of RCS loops and RHR loops. Any one loop in operation provides enough flow to remove the decay heat from the core with forced circulation. An additional loop is required to be OPERABLE to provide redundancy for heat removal.

Note 1 permits all RCPs or RHR pumps to be not in operation for ≤ 1 hour per 8 hour period. The purpose of the Note is to permit tests that are designed to validate various accident analyses values. An | B
RAI 3.4.6-2

BASES

LCO (continued)

example of one of the tests is validation of the pump coastdown curve used as input to a number of accident analyses including a loss of flow accident. This test is generally performed during the initial startup testing program, and as such should only be performed once. If changes are made to the RCS that would cause a change to the flow characteristics of the RCS, the input values must be revalidated by conducting the test again. The 1 hour time period is adequate to perform the test, and operating experience has shown that boron stratification is not a problem during this short period with no forced flow. B RAI 3.4.6-2

Utilization of Note 1 is permitted provided the following conditions are met along with any other conditions imposed by initial startup test procedures:

a. No operations are permitted that would dilute the RCS boron concentration, therefore maintaining the margin to criticality. Boron reduction is prohibited because a uniform concentration distribution throughout the RCS cannot be ensured when in natural circulation; and

b. Core outlet temperature is maintained at least 10°F below saturation temperature, so that no vapor bubble may form and possibly cause a natural circulation flow obstruction.

Note 2 requires that the secondary side water temperature of each SG be ≤ 50°F above each of the RCS cold leg temperatures before the start of an RCP with any RCS cold leg temperature ≤ the Low Temperature Overpressure Protection (LTOP) enabling temperature specified in the PTLR. This restraint is to prevent a low temperature overpressure event due to a thermal transient when an RCP is started. SG secondary side water temperature can be approximated by using the SG metal temperature indicator. B

An OPERABLE RCS loop comprises an OPERABLE RCP and an OPERABLE SG in accordance with the Steam Generator Tube Surveillance Program, which has the minimum water level specified in SR 3.4.6.2. The OPERABLE RCP and SG must be in the same loop for the RCS loop to be considered OPERABLE.

Similarly for the RHR System, an OPERABLE RHR loop comprises an OPERABLE RHR pump capable of providing forced flow to an OPERABLE RHR heat exchanger. RCPs and RHR pumps are OPERABLE if they are capable of being powered and are able to provide forced flow if required.

BASES

APPLICABILITY

In MODE 4, this LCO ensures forced circulation of the reactor coolant to remove decay heat from the core and to provide proper boron mixing. One loop of either RCS or RHR provides sufficient circulation for these purposes. However, two loops consisting of any combination of RCS and RHR loops are required to be OPERABLE to meet single failure considerations.

Operation in other MODES is covered by:

LCO 3.4.4, "RCS Loops — MODES 1 and 2";
LCO 3.4.5, "RCS Loops — MODE 3";
LCO 3.4.7, "RCS Loops — MODE 5, Loops Filled";
LCO 3.4.8, "RCS Loops — MODE 5, Loops Not Filled";
LCO 3.9.4, "Residual Heat Removal (RHR) and Coolant Circulation — High Water Level" (MODE 6); and
LCO 3.9.5, "Residual Heat Removal (RHR) and Coolant Circulation — Low Water Level" (MODE 6).

ACTIONS

A.1

If one required RCS loop is inoperable and two RHR loops are inoperable, redundancy for heat removal is lost. Action must be initiated to restore a second RCS or RHR loop to OPERABLE status. The immediate Completion Time reflects the importance of maintaining the availability of two paths for heat removal.

B.1

If one required RHR loop is OPERABLE and in operation and there are no RCS loops OPERABLE, an inoperable RCS or RHR loop must be restored to OPERABLE status to provide a redundant means for decay heat removal.

If the parameters that are outside the limits cannot be restored, the unit must be brought to MODE 5 within 24 hours. Bringing the unit to MODE 5 is a conservative action with regard to decay heat removal. With only one RHR loop OPERABLE, redundancy for decay heat removal is lost and, in the event of a loss of the remaining RHR loop, it would be safer to initiate that loss from MODE 5 (≤ 200°F) rather than MODE 4 (200 to 350°F). The Completion Time of 24 hours is a reasonable time, based on operating experience, to reach MODE 5 from MODE 4 in an orderly manner and without challenging plant systems.

BASES

ACTIONS (continued)

C.1 and C.2

If no loop is OPERABLE or in operation, except during conditions permitted by Note 1 in the LCO section, all operations involving a reduction of RCS boron concentration must be suspended and action to restore one RCS or RHR loop to OPERABLE status and operation must be initiated. Boron dilution requires forced circulation for proper mixing, and the margin to criticality must not be reduced in this type of operation. The immediate Completion Times reflect the importance of maintaining operation for decay heat removal. The action to restore must be continued until one loop is restored to OPERABLE status and operation.

SURVEILLANCE REQUIREMENTS

SR 3.4.6.1

This SR requires verification every 12 hours that one RCS or RHR loop is in operation. Verification includes flow rate, temperature, or pump status monitoring, which help ensure that forced flow is providing heat removal. The Frequency of 12 hours is sufficient considering other indications and alarms available to the operator in the control room to monitor RCS and RHR loop performance.

SR 3.4.6.2

SR 3.4.6.2 requires verification of SG OPERABILITY.
SG OPERABILITY is verified by ensuring that the secondary side narrow range water level is ≥ 30%. If the SG secondary side narrow range water level is < 30%, the tubes may become uncovered and the associated loop may not be capable of providing the heat sink necessary for removal of decay heat. The 12 hour Frequency is considered adequate in view of other indications available in the control room to alert the operator to the loss of SG level.

SR 3.4.6.3

Verification that the required pump is OPERABLE ensures that an additional RCS or RHR pump can be placed in operation, if needed, to maintain decay heat removal and reactor coolant circulation. Verification is performed by verifying proper breaker alignment and power available to the required pump. The Frequency of 7 days is considered reasonable in view of other administrative controls available and has been shown to be acceptable by operating experience.

REFERENCES

None.

DOC Number	DOC Text	
M.01 Rev. A	CTS 15.3.1.A.3.b provides decay heat removal requirements for conditions where reactor coolant temperature is < 140 F. The requirements of this specification with the reactor vessel head less than fully tensioned, are addressed in ITS LCO 3.9.4 and 3.9.5. Proposed ITS LCO 3.4.7 and 3.4.8 address the decay heat removal requirements in MODE 5, with LCO 3.4.7 addressing the condition with the RCS loops filled and LCO 3.4.8 addressing the condition with RCS loops not filled. The ITS definition of MODE 5 includes the conditions whereby Tavg is less than or equal to 200 F. Therefore, the proposed revision changes the applicability of the RHR requirements from less than 140 F to less than or equal to 200 F. The 140 F limit is based on the CTS definition of refueling shutdown and is an artificial limit not related to any physical system limitation or condition. While the lower temperature provided some additional subcooling margin in the event of a temporary loss of shutdown cooling, the Technical Specifications ensure appropriate redundancy of shutdown cooling such that the potential for a loss of cooling is minimized. Raising the temperature limit to 200 F does not increase the probability of a loss of cooling. The RHR System is designed, operated and maintained to ensure operability under these temperature conditions.	
	CTS:	**ITS:**
	15.03.01.A.03.B	LCO 3.04.07
M.02 Rev. B	The CTS 15.3.1.A.3.b(4) allows one of the two RHR loops to be temporarily out of service to meet surveillance requirements. Proposed ITS LCO 3.4.7, Note 2, allows one required RHR pump to be inoperable for a period of up to 2 hours for surveillance testing, provided that the other RHR loop is operable and in operation. Changing "temporarily" out of service to inoperable for "up to 2 hours", places additional requirements on plant operation and is more restrictive. Two hours is a reasonable time to conduct surveillances including those required by ASME Section XI and the Technical Specifications, without unnecessarily challenging decay heat removal. Note 2 also ensures that a residual heat removal loop is in operation as required by the existing Specifications and ITS LCO 3.4.7.	
	CTS:	**ITS:**
	15.03.01.A.03.B.04	LCO 3.04.07 NOTE 2
M.03 Rev. A	CTS 15.3.15.B.2 prohibits starting a RCP with RCS temperature < 355 F, unless compliance with one of the conditions provided in CTS 15.3.15.B.2.a or 15.3.15.B.2.b is met. One of the conditions provided in CTS 15.3.15.B.2.a is a pressure absorbing volume in the pressurizer. In order to retain this allowable condition in ITS 3.4.7, a quantifiable pressurizer water level would need to be specified, to ensure adequate volume exists in the pressurizer to accommodate the swell resulting from the RCP start, to prevent a low temperature overpressure event that could place the plant in an unanalyzed condition. No such value could be found in the Point Beach CLB; therefore, this condition is not being retained in ITS 3.4.7, resulting in more restrictive requirements for plant operation.	
	CTS:	**ITS:**
	15.03.15.B.02.A	N/A

DOC Number	DOC Text
M.04 Rev. A	CTS 15.3.15.B.2 prohibits starting a RCP with RCS temperature < 355 F, unless compliance with one of the conditions provided in CTS 15.3.15.B.2.a or 15.3.15.B.2.b is met. One of the conditions provided in CTS 15.3.15.B.2.a is a pressure absorbing volume in the steam generator tubes. This condition is not being retained in ITS LCO 3.4.7, because no method exists to verify the volume in the steam generator tubes that would be required to accommodate the swell resulting from a RCP start. Therefore, prevention of a low temperature overpressure event cannot be ensured, and the plant may be placed in an unanalyzed condition as a result of the RCP start.

CTS:	ITS:
15.03.15.B.02.A	N/A

DOC Number	DOC Text
M.05 Rev. A	CTS 15.3.1.A.3.b is revised to adopt the actions associated with ITS LCO 3.4.7 Condition A. If one RHR train is inoperable and the required SG has secondary side water level < 30% narrow range, redundancy for heat removal is lost. Action must be initiated immediately to restore a second RHR train to OPERABLE status or to restore the required SG secondary side water level. Either Required Action A.1 or Required Action A.2 will restore redundant heat removal paths. The immediate Completion Time reflects the importance of maintaining the availability of two paths for heat removal. This change imposes additional requirements on plant operation and, therefore, is more restrictive.

CTS:	ITS:
NEW	LCO 3.04.07 COND A
	LCO 3.04.07 COND A RA A.1
	LCO 3.04.07 COND A RA A.2

Section 3.4.7 CTS Markup Inserts

Insert 3.4.7-1:

LCO 3.4.7 One residual heat removal (RHR) loop shall be OPERABLE and in operation, and either:

A.4

a. One additional RHR loop shall be OPERABLE; or

L.1

b. The secondary side water level of at least one steam generator (SG) shall be ≥ 30% narrow range.

NOTES

1. The RHR pump of the loop in operation may be not in operation for ≤ 1 hour per 8 hour period provided:

L.2

a. No operations are permitted that would cause reduction of the RCS boron concentration; and

b. Core outlet temperature is maintained at least 10 °F below saturation temperature.

M.2

2. One required RHR loop may be inoperable for ≤ 2 hours for surveillance testing provided that the other RHR loop is OPERABLE and in operation.

B
RAI 3.4.7-1

3. No reactor coolant pump shall be started with one or more RCS cold leg temperatures ≤ Low Temperature Overpressure Protection (LTOP) enabling temperature specified in the PTLR unless the secondary side water temperature of each SG is ≤ 50°F above each of the RCS cold leg temperatures.

LA.1

L.3

4. All RHR loops may be remo ved from operation during planned heatup to MODE 4 or during the performance of SR 3.4.14.1 when at least one RCS loop is in operation.

APPLICABILITY: MODE 5 with RCS loops filled.

M.1

Justification For Deviations - NUREG-1431 Section 3.04.07

10-Aug-00

JFD Number	JFD Text	
01 Rev. A	The brackets have been removed and the proper plant specific information has been provided. In some instances, even though the information was designated as plant specific information in the LCO (bracketed), the corresponding Bases information was not bracketed. These cases are self evident, corresponding to the bracketed information in the LCO, and have had the appropriate site specific information provided.	
	ITS:	**NUREG:**
	B 3.04.07	B 3.04.07
	LCO 3.04.07 B	LCO 3.04.07 B
	LCO 3.04.07 COND A	LCO 3.04.07 COND A
	LCO 3.04.07 COND A RA A.2	LCO 3.04.07 COND A RA A.2
	LCO 3.04.07 NOTE 3	LCO 3.04.07 NOTE 3
	SR 3.04.07.02	SR 3.04.07.02
02 Rev. A	The actual numerical values for LTOP enabling temperature are replaced with a reference to the temperature specified in the PTLR. The LTOP enabling temperature will then be calculated and controlled by the licensee in accordance with the topical reports identified in the PTLR.	
	ITS:	**NUREG:**
	B 3.04.07	B 3.04.07
	LCO 3.04.07 NOTE 3	LCO 3.04.07 NOTE 3
03 Rev. B	LCO 3.4.7 Bases description of no flow testing is revised to reflect testing which may be performed at Point Beach. Although Point Beach does not currently require validation of rod drop times under no flow conditions, this is an example of testing which may be performed and would require stopping of all RHR pumps.	
	ITS:	**NUREG:**
	B 3.04.07	B 3.04.07
04 Rev. A	LCO 3.9.2 "Unborated Water Source Isolation Valves" was not adopted, based on the Point Beach design. Accordingly, the references to LCO 3.9.5 and 6 within the Bases for LCO 3.4.7 have been revised to reflect the renumbering that has occurred in Section 3.9 of the ITS.	
	ITS:	**NUREG:**
	B 3.04.07	B 3.04.07

Justification For Deviations - NUREG-1431 Section 3.04.07

01-Aug-00

JFD Number	JFD Text	
05 Rev. B	Not Used.	
	ITS:	**NUREG:**
	B 3.04.07	B 3.04.07
	N/A	N/A
06 Rev. A	The wording of the LCO 3.4.7 Note and Bases was changed from "...may be de-energized..." to "...may not be in operation...", per approved TSTF 153. However, "…may not be in operation…" could easily be interpreted to imply a condition that forbids RCP operation. To prevent this misunderstanding, the wording has been changed to, "…may be not in operation…"	
	ITS:	**NUREG:**
	B 3.04.07	B 3.04.07
	LCO 3.04.07 NOTE 1	LCO 3.04.07 NOTE 1
07 Rev. A	"Narrow range" was added to the description of the required secondary side water level of the steam generators. NUREG-1431 did not specify a level indication instrumentation reference for the steam generator water level. To avoid possible interpretation, "narrow range" was added to specify that the required steam generator water level percentage is indicated narrow range. 30% narrow range level indication is a much higher water level (i.e. more conservative) than 30% wide range indication and ensures that the steam generator tubes are covered.	
	ITS:	**NUREG:**
	B 3.04.07	B 3.04.07
	LCO 3.04.07 B	LCO 3.04.07 B
	SR 3.04.07.02	SR 3.04.07.02
08 Rev. A	An allowance is being added to LCO 3.4.7 NOTE 4 and the applicable Bases to allow both RHR loops to be removed from operation when at least one RCS loop is in operation to allow for the performance of SR 3.4.14.1, RCS PIV leakage testing. The CTS allows reactor coolant loops for decay heat removal when the RCS temperature is > 140 °F and < 350 °F in accordance with CTS 15.3.1.A.3.a(1). This allowance is necessary based on the design of the Point Beach RHR System configuration, which requires the system to be removed from service to perform the required PIV leakage testing.	
	ITS:	**NUREG:**
	B 3.04.07	B 3.04.07
	LCO 3.04.07 NOTE 4	LCO 3.04.07 NOTE 4

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.7 RCS Loops – MODE 5, Loops Filled

LCO 3.4.7 One residual heat removal (RHR) loop shall be OPERABLE and in operation, and either:

7

a. One additional RHR loop shall be OPERABLE; or

narrow range

b. The secondary side water level of at least [two] steam generator~~s~~ (SG~~s~~) shall be ≥ [17]%.

1 30 1 one

Approved TSTF-153

------------------------------NOTES------------------------------

not be in operation

1. The RHR pump of the loop in operation may ~~be~~ Approved TSTF-153 de-energized for ≤ 1 hour per 8 hour period provided:

be not in operation

6

a. No operations are permitted that would cause reduction of the RCS boron concentration; and

b. Core outlet temperature is maintained at least 10 °F below saturation temperature.

B
RAI 3.4.7-1

2

2. One required RHR loop may be inoperable for up to 2 hours for surveillance testing provided that the other RHR loop is OPERABLE and in operation.

Low Temperature Overpressure Protection (LTOP) enabling temperature specified in the PTLR.

3. No reactor coolant pump shall be started with one or more RCS cold leg temperatures ≤ [275]°F unless the secondary side water temperature of each SG is ≤ [50]°F above each of the RCS cold leg temperatures.

50 1

4. All RHR loops may be removed from operation during planned heatup to MODE 4 when at least one RCS loop is in operation.

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8 or during the performance of SR 3.4.14.1

APPLICABILITY: MODE 5 with RCS loops filled.

APPLICABLE SAFETY ANALYSES

In MODE 5, RCS circulation is considered in the determination of the time available for mitigation of the accidental boron dilution event. The RHR loops provide this circulation.

RCS Loops – MODE 5 (Loops Filled) have been identified in the NRC Policy Statement as important contributors to risk reduction.

LCO

The purpose of this LCO is to require that at least one of the RHR loops be OPERABLE and in operation with an ~~additional RHR loop~~ OPERABLE or ~~two~~ one SG ~~s~~ with secondary side narrow range water level ≥ [17] 30 %. One RHR loop provides sufficient forced circulation to perform the safety functions of the reactor coolant under these conditions. An additional RHR loop is required to be OPERABLE to meet single failure considerations. However, if the standby RHR loop is not OPERABLE, an acceptable alternate method is ~~two~~ one SG ~~s~~ with ~~their~~ its secondary side water level ~~s~~ ≥ [17] 30 %. Should the operating RHR loop fail, the SG ~~s~~ could be used to remove the decay heat via natural circulation (Ref. 1).

7 1 1 1 1 1

Approved TSTF-114

~~Note 1 permits all RHR pumps to be de-energized ≤ 1 hour per 8 hour period. The purpose of the Note is to permit tests designed to validate various accident analyses values. One of the tests performed during the startup testing program is the validation of rod drop times during cold conditions, both with and without flow. The no flow test may be performed in MODE 3, 4, or 5 and requires that the pumps be stopped for a short period of time. The Note permits de-energizing of the pumps in order to perform this test and validate the assumed analysis values. If changes are made to the RCS that would cause a change to the flow characteristics of the RCS, the input values must be revalidated by conducting the test again.~~

Note 1 permits all RHR pumps to be not in operation ≤ 1 hour per 8 hour period. The Note permits the performance of tests which require that the pumps be stopped for a short period of time. One example of a test which may be performed is the validation of rod drop times during cold conditions without flow. This no flow test may be performed in MODE 3, 4, or 5 and requires that the pumps be stopped for a short period of time. The Note permits stopping of the pumps in order to perform the test.

3

The 1 hour time period is adequate to perform the test, and operating experience has shown that boron stratification is not likely during this short period with no forced flow.

B

RAI 3.4.7-2

Utilization of Note 1 is permitted provided the following conditions are met, along with any other conditions imposed by initial startup test procedures:

LCO (continued)

a. No operations are permitted that would dilute the RCS boron concentration, therefore maintaining the margin to criticality. Boron reduction is prohibited because a uniform concentration distribution throughout the RCS cannot be ensured when in natural circulation; and

b. Core outlet temperature is maintained at least 10 °F below saturation temperature, so that no vapor bubble may form and possibly cause a natural circulation flow obstruction.

Note 2 allows one RHR loop to be inoperable for a period of up to 2 hours, provided that the other RHR loop is OPERABLE and in operation. This permits periodic surveillance tests to be performed on the inoperable loop during the only time when such testing is safe and possible.

B
RAI 3.4.7-1

2

Low Temperature Overpressure Protection (LTOP) arming temperature specified in the PTLR.

1 50

Note 3 requires that the secondary side water temperature of each SG be ≤ [50]°F above each of the RCS cold leg temperatures before the start of a reactor coolant pump (RCP) with an RCS cold leg temperature ≤ [275]°F. This restriction is to prevent a low temperature overpressure event due to a thermal transient when an RCP is started.

Note 4 also allows both RHR loops to be removed from operation when at least one RCS loop is in operation to allow for the performance of SR 3.4.14.1, RCS PIV leakage testing. This allowance is necessary based on the design of the Point Beach RHR System configuration, which requires the system to be removed from service to perform the required PIV leakage testing.

8

Note 4 provides for an orderly transition from MODE 5 to MODE 4 during a planned heatup by permitting removal of RHR loops from operation when at least one RCS loop is in operation. This Note provides for the transition to MODE 4 where an RCS loop is permitted to be in operation and replaces the RCS circulation function provided by the RHR loops.

via natural circulation (Ref. 1)

RHR pumps are OPERABLE if they are capable of being powered and are able to provide flow if required. An OPERABLE SG can perform as a heat sink ~~when it has an adequate water level and is OPERABLE in accordance with the Steam Generator Tube Surveillance Program.~~

Approved TSTF-114

APPLICABILITY

In MODE 5 with RCS loops filled, this LCO requires forced circulation of the reactor coolant to remove decay heat from the core and to provide proper boron mixing. One loop of RHR provides sufficient circulation for these purposes. However, one additional RHR loop is required to be OPERABLE,

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.7 RCS Loops—MODE 5, Loops Filled

LCO 3.4.7 One residual heat removal (RHR) loop shall be OPERABLE and in operation, and either:

a. One additional RHR loop shall be OPERABLE; or

b. The secondary side water level of at least one steam generator (SG) shall be ≥ 30% narrow range.

-----------------------------------NOTES-----------------------------------

1. The RHR pump of the loop in operation may be not in operation for ≤ 1 hour per 8 hour period provided:

   a. No operations are permitted that would cause reduction of the RCS boron concentration; and

   b. Core outlet temperature is maintained at least 10°F below saturation temperature.

2. One required RHR loop may be inoperable for up to 2 hours for surveillance testing provided that the other RHR loop is OPERABLE and in operation. B RAI 3.4.7-1

3. No reactor coolant pump shall be started with one or more RCS cold leg temperatures ≤ Low Temperature Overpressure Protection (LTOP) arming temperature specified in the PTLR unless the secondary side water temperature of each SG is ≤ 50°F above each of the RCS cold leg temperatures.

4. All RHR loops may be removed from operation during planned heatup to MODE 4 or during the performance of SR 3.4.14.1 when at least one RCS loop is in operation.

APPLICABILITY: MODE 5 with RCS loops filled.

B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.7 RCS Loops - MODE 5, Loops Filled

BASES

BACKGROUND

In MODE 5 with the RCS loops filled, the primary function of the reactor coolant is the removal of decay heat and transfer this heat either to the steam generator (SG) secondary side coolant via natural circulation (Ref. 1) or the component cooling water via the residual heat removal (RHR) heat exchangers. While the principal means for decay heat removal is via the RHR System, the SGs via natural circulation (Ref. 1) are specified as a backup means for redundancy. Even though the SGs cannot produce steam in this MODE, they are capable of being a heat sink due to their large contained volume of secondary water. As long as the SG secondary side water is at a lower temperature than the reactor coolant, heat transfer will occur. The rate of heat transfer is directly proportional to the temperature difference. The secondary function of the reactor coolant is to act as a carrier for soluble neutron poison, boric acid.

In MODE 5 with RCS loops filled, the reactor coolant is circulated by means of two RHR loops connected to the RCS, each loop containing an RHR heat exchanger, an RHR pump, and appropriate flow and temperature instrumentation for control, protection, and indication. One RHR pump circulates the water through the RCS at a sufficient rate to prevent boric acid stratification.

The number of loops in operation can vary to suit the operational needs. The intent of this LCO is to provide forced flow from at least one RHR loop for decay heat removal and transport. The flow provided by one RHR loop is adequate for decay heat removal. The other intent of this LCO is to require that a second path be available to provide redundancy for heat removal.

The LCO provides for redundant paths of decay heat removal capability. The first path can be an RHR loop that must be OPERABLE and in operation. The second path can be another OPERABLE RHR loop or maintaining one SG with secondary side water levels above 30% narrow range to provide an alternate method for decay heat removal.

APPLICABLE SAFETY ANALYSES

In MODE 5, RCS circulation is considered in the determination of the time available for mitigation of the accidental boron dilution event. The RHR loops provide this circulation.

RCS Loops - MODE 5 (Loops Filled) have been identified in the NRC Policy Statement as important contributors to risk reduction.

BASES

LCO

The purpose of this LCO is to require that at least one of the RHR loops be OPERABLE and in operation with an additional RHR loop OPERABLE or one SG with secondary side water level $\geq$ 30% narrow range. One RHR loop provides sufficient forced circulation to perform the safety functions of the reactor coolant under these conditions. An additional RHR loop is required to be OPERABLE to meet single failure considerations. However, if the standby RHR loop is not OPERABLE, an acceptable alternate method is one SG with its secondary side water level $\geq$ 30% narrow range. Should the operating RHR loop fail, the SG could be used to remove the decay heat via natural circulation (Ref. 1).

Note 1 permits all RHR pumps to be not in operation $\leq$ 1 hour per 8 hour period. The Note permits the performance of tests which require that the pumps be stopped for a short period of time. One example of a test which may be performed is the validation of rod drop times during cold conditions without flow. This no flow test may be performed in MODE 3, 4, or 5 and requires that the pumps be stopped for a short period of time. The Note permits stopping of the pumps in order to perform the test. The 1 hour time period is adequate to perform the test, and operating experience has shown that boron stratification is not likely during this short period with no forced flow. B RAI 3.4.7-2

Utilization of Note 1 is permitted provided the following conditions are met, along with any other conditions imposed by initial startup test procedures:

a. No operations are permitted that would dilute the RCS boron concentration, therefore maintaining the margin to criticality. Boron reduction is prohibited because a uniform concentration distribution throughout the RCS cannot be ensured when in natural circulation; and

b. Core outlet temperature is maintained at least 10°F below saturation temperature, so that no vapor bubble may form and possibly cause a natural circulation flow obstruction.

Note 2 allows one RHR loop to be inoperable for a period of up to 2 hours, provided that the other RHR loop is OPERABLE and in operation. This permits periodic surveillance tests to be performed on the inoperable loop during the only time when such testing is safe and possible. B RAI 3.4.7-1

Note 3 requires that the secondary side water temperature of each SG be $\leq$ 50°F above each of the RCS cold leg temperatures before the start of a reactor coolant pump (RCP) with an RCS cold leg

BASES

LCO (continued)

temperature ≤ Low Temperature Overpressure Protection (LTOP) arming temperature specified in the PTLR. This restriction is to prevent a low temperature overpressure event due to a thermal transient when an RCP is started.

Note 4 provides for an orderly transition from MODE 5 to MODE 4 during a planned heatup by permitting removal of RHR loops from operation when at least one RCS loop is in operation. This Note provides for the transition to MODE 4 where an RCS loop is permitted to be in operation and replaces the RCS circulation function provided by the RHR loops. Note 4 also allows both RHR loops to be removed from operation when at least one RCS loop is in operation to allow for the performance of SR 3.4.14.1, RCS PIV leakage testing. This allowance is necessary based on the design of the Point Beach RHR System configuration, which requires the system to be removed from service to perform the required PIV testing.

RHR pumps are OPERABLE if they are capable of being powered and are able to provide flow if required. An OPERABLE SG can perform as a heat sink via natural circulation (Ref. 1) when it has an adequate water level and is OPERABLE in accordance with the Steam Generator Tube Surveillance Program.

APPLICABILITY

In MODE 5 with RCS loops filled, this LCO requires forced circulation of the reactor coolant to remove decay heat from the core and to provide proper boron mixing. One loop of RHR provides sufficient circulation for these purposes.

However, one additional RHR loop is required to be OPERABLE, or the secondary side water level of at least one SGs is required to be ≥ 30% narrow range.

Operation in other MODES is covered by:

LCO 3.4.4, "RCS Loops - MODES 1 and 2";
LCO 3.4.5, "RCS Loops - MODE 3";
LCO 3.4.6, "RCS Loops - MODE 4";
LCO 3.4.8, "RCS Loops - MODE 5, Loops Not Filled";
LCO 3.9.4, "Residual Heat Removal (RHR) and Coolant Circulation - High Water Level" (MODE 6); and
LCO 3.9.5, "Residual Heat Removal (RHR) and Coolant Circulation - Low Water Level" (MODE 6).

BASES

ACTIONS

A.1 and A.2

If one RHR loop is inoperable and the required SG has secondary side water level < 30% narrow range, redundancy for heat removal is lost. Action must be initiated immediately to restore a second RHR loop to OPERABLE status or to restore the required SG secondary side water level. Either Required Action A.1 or Required Action A.2 will restore redundant heat removal paths. The immediate Completion Time reflects the importance of maintaining the availability of two paths for heat removal.

B.1 and B.2

If no RHR loop is in operation, except during conditions permitted by Note 1, or if no loop is OPERABLE, all operations involving a reduction of RCS boron concentration must be suspended and action to restore one RHR loop to OPERABLE status and operation must be initiated. To prevent boron dilution, forced circulation is required to provide proper mixing and preserve the margin to criticality in this type of operation. The immediate Completion Times reflect the importance of maintaining operation for heat removal.

SURVEILLANCE REQUIREMENTS

SR 3.4.7.1

This SR requires verification every 12 hours that the required loop is in operation. Verification includes flow rate, temperature, or pump status monitoring, which help ensure that forced flow is providing heat removal. The Frequency of 12 hours is sufficient considering other indications and alarms available to the operator in the control room to monitor RHR loop performance.

SR 3.4.7.2

Verifying that at least one SG is OPERABLE by ensuring its secondary side narrow range water level is ≥ 30% narrow range ensures an alternate decay heat removal method via natural circulation (Ref. 1) in the event that the second RHR loop is not OPERABLE. If both RHR loops are OPERABLE, this Surveillance is not needed. The 12 hour Frequency is considered adequate in view of other indications available in the control room to alert the operator to the loss of SG level.

BASES

SURVEILLANCE REQUIREMENTS (continued)	SR 3.4.7.3 Verification that a second RHR pump is OPERABLE ensures that an additional pump can be placed in operation, if needed, to maintain decay heat removal and reactor coolant circulation. Verification is performed by verifying proper breaker alignment and power available to the RHR pump. If secondary side water level is $\geq$ 30% narrow range in at least two SGs, this Surveillance is not needed. The Frequency of 7 days is considered reasonable in view of other administrative controls available and has been shown to be acceptable by operating experience.
REFERENCES	1. NRC Information Notice 95-35, "Degraded Ability of Steam Generators to Remove Decay Heat by Natural Circulation."

Description of Changes - NUREG-1431 Section 3.04.09

03-Aug-00

DOC Number	DOC Text	
A.01 Rev. A	In the conversion of Point Beach current Technical Specifications (CTS) to the proposed plant specific Improved Technical Specifications (ITS), certain wording preferences or conventions are adopted which do not result in technical changes (either actual or interpretational). Editorial changes, reformatting, and revised numbering are adopted to make the ITS consistent with the Standard Technical Specifications, Westinghouse Plants, NUREG-1431, Revision 1 (i.e., Improved Standard Technical Specifications (ISTS)).	
	CTS:	**ITS:**
	15.03.01.A.06	LCO 3.04.09
A.02 Rev. A	CTS 15.3.1.A.6 is revised to adopt proposed ITS 3.4.9, Actions B and C. The CTS does not provide explicit actions for non-compliance with the LCO. As a result, CTS 15.3.0.B applies, which requires the plant be placed in Hot Shutdown in 7 hours and Cold Shutdown in 37 hours. Per CTS 15.3.0.C, once the plant exits the applicability of TS 15.3.1.A.6 (critical operation), the required actions do not need to be completed. Therefore, the plant is required to be in Hot Shutdown within 7 hours. Proposed ITS 3.4.9, Action B, requires the restoration of the required pressurizer heaters to an operable status in 1 hour. If the pressurizer heaters cannot be restored to an operable status in one hour, Condition C requires the plant to be in MODE 3 in 6 hours and MODE 4 in 12 hours. This takes the unit out of the applicable MODES and restores the unit to operation within the bounds of the safety analyses. Although these required actions appear more restrictive, they are the same as the CTS 15.3.0 required actions. Requiring the operability of the pressurizer in MODE 3 is a new requirement to Point Beach's technical specifications and is discussed in LCO 3.4.9 DOC M.2. Therefore, this change is administrative.	
	CTS:	**ITS:**
	NEW	LCO 3.04.09 COND B
		LCO 3.04.09 COND B RA B.1
		LCO 3.04.09 COND C
		LCO 3.04.09 COND C RA C.1
		LCO 3.04.09 COND C RA C.2
A.03 Rev. A	CTS 15.4.1, Table 15.4.1-2, Item 30, requires a quarterly verification that 100 KW of pressurizer heaters are available. ITS SR 3.4.9.2 requires verification every 92 days that the capacity of the required pressurizer heaters is greater than or equal to 100 KW. Both surveillance requirements accomplish the same objective at virtually the same frequency. Therefore, this change is administrative.	
	CTS:	**ITS:**
	15.04.01 T 15.04.01-02 30	SR 3.04.09.02

Description of Changes - NUREG-1431 Section 3.04.09

03-Aug-00

DOC Number	DOC Text
A.04 Rev. A	The Bases of the current Technical Specifications for this section have been completely replaced by revised Bases that reflect the format and applicable content of PBNP ITS Chapter 3.4, consistent with the Standard Technical Specifications for Westinghouse Plants, NUREG-1431. The revised Bases are as shown in the PBNP ITS Bases. **CTS:** BASES **ITS:** B 3.04.09
A.05 Rev. B	If the pressurizer water level requirements of CTS 15.3.1.A.6 are not met, no required actions are specified. Therefore CTS 15.3.0.B is required to be entered, requiring action to be initiated within 1 hour to place the unit in a condition where the LCO does not apply. Proposed ITS 3.4.9, Required Action A.1, will require the restoration of the pressurizer water level within 1 hour, when pressurizer water level is not within the MODE 1 limit. Due to the availability of pressurizer water level indications in the control room, and alarms in the control room when pressurizer water level is above the programmed band, it is unlikely that exceeding MODE 1 pressurizer water level limit would result in an immediate threat of taking the pressurizer water solid. Therefore, allowing 1 hour to restore the pressurizer water level to within the initial condition assumptions of the loss of normal feedwater accident analyses is reasonable based on the probability of this accident occurring during this period of time, and is consistent with the actions required by CTS 15.3.0.B. **CTS:** NEW **ITS:** LCO 3.04.09 COND A LCO 3.04.09 COND A RA A.1
L.01 Rev. A	CTS 15.3.1.A.6 requires the pressurizer to be operable with a water level of greater than 10%. Specifying a minimum pressurizer water level is not being retained in ITS. Minimum water level is not required to preserve accident analysis assumptions. Proposed ITS 3.4.9 requires the pressurizer to be operable. The surveillance requirements associated with LCO 3.4.9 define the operability requirements of the pressurizer. More specifically, SR 3.4.9.1 requires verifying the pressurizer water level is less than or equal to 50.8% in MODE 1 and less than or equal to 95% in MODES 2 and 3. SR 3.4.9.2 requires verifying the capacity of the pressurizer heaters is greater than or equal to 100 KW. In order for the capacity of the required pressurizer heaters to be greater than or equal to 100 KW, the pressurizer water level must be above the pressurizer heater cutout setpoint. Therefore, pressurizer heater operability is dependent on adequate pressurizer water level. The actions of LCO 3.4.9, Condition B, would be required if the heaters become uncovered. Although this change is less restrictive, it is acceptable. The proposed actions of ITS LCO 3.4.9 are the same as the CTS 15.3.0.B actions, when the pressurizer heaters become uncovered. **CTS:** 15.03.01.A.06 **ITS:** LCO 3.04.09

Description of Changes - NUREG-1431 Section 3.04.09

03-Aug-00

DOC Number	DOC Text
M.01 Rev. A	CTS 15.3.1.A.6 requires the pressurizer to be operable with at least 100 KW of pressurizer heaters available. Additionally, at least one bank of pressurizer heaters is required to be supplied by an emergency bus power supply. Proposed ITS 3.4.9 requires the pressurizer heaters to be operable with a capacity of greater than or equal to 100 KW. As stated in the Bases of proposed ITS 3.4.9, the required heaters are those that are powered from a safeguards bus. CTS 15.3.1.A.6 does not place a requirement on the capacity of the pressurizer heaters powered from an emergency bus; therefore, the capacity of the pressurizer heaters supplied from the emergency bus could be less than 100 KW. Requiring the pressurizer heaters to be capable of being powered from an emergency power supply ensures the availability of the heaters to maintain reactor coolant system pressure. The capability to maintain and control system pressure is important for maintaining subcooled conditions in the RCS and ensuring the capability to remove decay core heat by either forced or natural circulation of reactor coolant. Since this change imposes additional requirements on plant operation, it is more restrictive.

CTS:	ITS:
15.03.01.A.06	LCO 3.04.09

DOC Number	DOC Text
M.02 Rev. B	CTS 15.3.1.A.6 requires the pressurizer water level be maintained less than 95% during steady-state power operation. CTS 15.3.1.F.5 requires the reactor be maintained subcritical by at least 1% dk/k until normal water level is established in the pressurizer. Proposed ITS 3.4.9 requires the pressurizer be operable in MODES 1, 2 and 3. LCO 3.4.9 and the associated surveillance requirements of LCO 3.4.9 define the operability requirements of the pressurizer. SR 3.4.9.1 requires a verification that the pressurizer water level is less than or equal to 50.8% in MODE 1 and less than or equal to 95% in MODES 2 and 3. The more restrictive requirement in MODE 1 is necessary to be consistent with the initial condition assumptions used in the accident analysis for a loss of normal feedwater. The results of the accident analysis show that there is a high probability that the pressurizer would become water solid, in the event that the accident assumed an initial pressurizer water level of 95%. The addition of MODE 3 to the applicability is made to prevent solid water RCS operation during heatup and cooldown, to avoid rapid pressure rises caused by normal operational perturbation, such as RCP startup.

CTS:	ITS:
15.03.01.A.06	LCO 3.04.09
15.03.01.F.05	LCO 3.04.09

DOC Number	DOC Text
M.03 Rev. B	Not used.

CTS:	ITS:
N/A	N/A
	N/A

Description of Changes - NUREG-1431 Section 3.04.09

03-Aug-00

DOC Number	DOC Text
M.04 Rev. A	CTS 15.3.1.A.6 is revised to adopt SR 3.4.9.1. SR 3.4.9.1 requires that during steady state operation, pressurizer level is maintained below the nominal upper limit to provide a minimum space for a steam bubble. The Surveillance is performed by observing the indicated level. The Frequency of 12 hours has been shown by operating practice to be sufficient to regularly assess level for any deviation and verify that operation is within safety analyses assumptions. Alarms are also available for early detection of abnormal level indications. Since this change imposes new requirements on plant operation, it is more restrictive.

CTS:	ITS:
NEW	SR 3.04.09.01

Section 3.4.9 CTS Markup Inserts

Insert 3.4.9-1:

CONDITION	REQUIRED ACTION	COMPLETION TIME
A.5 A. Pressurizer water level not within limit in MODE 1.	A.1 Restore presurizer water level to within limit.	1 hour B RAI 3.4.9-2
B. Required pressurizer heaters inoperable.	B.1 Restore required pressurizer heaters to OPERABLE status.	1 hour
C. Required Action and associated Completion Time not met. OR Pressurizer water level not within limit in MODES 2 and 3.	C.1 Be in MODE 3. AND C.2 Be in MODE 4.	6 hours A.2 12 hours

Insert 3.4.9-2:

SURVEILLANCE	FREQUENCY
SR 3.4.9.1 Verify pressurizer water level is ≤ 50.8% in MODE 1 OR ≤ 95% in MODES 2 and 3.	12 hours M.4

Justification For Deviations - NUREG-1431 Section 3.04.09

01-Aug-00

JFD Number	JFD Text
01 Rev. B	The parameters associated with the operability of the pressurizer have been modified to reflect the Point Beach design. The water level requirements ensure a steam bubble is maintained in the pressurizer, consistent with the accident analysis assumptions for a loss of feedwater transient in MODE 1. A loss of feedwater is not a concern and is less severe in MODES 2 and 3 allowing for a higher level. Heater requirements are based on the CTS which are based on heater capacity, not on groups of heaters.

ITS:	NUREG:
LCO 3.04.09	LCO 3.04.09
N/A	LCO 3.04.09 A LCO 3.04.09 B

JFD Number	JFD Text
02 Rev. A	LCO 3.4.9 Conditions A and C are revised to accommodate the changes made to the pressurizer water level operability requirements. The pressurizer water level requirement in MODE 1 is revised to be consistent with the initial condition assumptions used in the accident analysis for a loss of normal feedwater. The MODE 1 pressurizer water level requirement is based on the nominal pressurizer water level consistent with steady-state operation (45.8%) plus a 5% allowance for steady-state fluctuations and instrumentation error. Due to the availability of indicators in the control room and alarms when pressurizer water level is above the programmed band, it is unlikely that exceeding the pressurizer water level requirement would result in an immediate threat of taking the pressurizer solid. Therefore, a period of time is allowed to restore the pressurizer water level to within limit. If the pressurizer water level cannot be restored within this time frame, then Condition C requires placing the plant in a condition where the LCO no longer applies. This is accomplished by requiring the plant to be in MODE 3 in 6 hours and MODE 4 in 12 hours. The actions required when the pressurizer water level requirements of MODE 2 and MODE 3 are not met, are revised to no longer require opening the reactor trip breakers in MODE 3. Exceeding the pressurizer water level requirement in MODE 2 or 3 would not result in an ATWS condition and, therefore, does not require this accident mitigating action. Requiring the plant to be in MODE 3 in 6 hours and in MODE 4 in 12 hours restores the plant to operation within the bounds of the safety analyses by taking the unit out of the applicable MODES in an orderly manner without challenging plant systems. Based on the above, the changes included in TSTF-87, Rev. 2 and TSTF-162, Rev. 0 for LCO 3.4.9 and associated Bases were not adopted.

ITS:	NUREG:
B 3.04.09	B 3.04.09
LCO 3.04.09 COND A	LCO 3.04.09 COND A
LCO 3.04.09 COND A RA A.1	LCO 3.04.09 COND A RA A.1
LCO 3.04.09 COND C	LCO 3.04.09 COND C
N/A	LCO 3.04.09 COND A RA A.2

Justification For Deviations - NUREG-1431 Section 3.04.09

01-Aug-00

JFD Number	JFD Text	
03 Rev. A	LCO 3.4.9 Condition B and SR 3.4.9.2 are modified to reflect the Point Beach licensing basis, which only requires a minimum capacity of pressurizer heaters, and no minimum number of groups. Required Action B.1 Completion Time is changed to 1 hour to reflect the importance of restoring the required pressurizer heaters to an operable status. Without redundant sources of pressurizer heaters available, the hot, high pressure condition cannot be maintained indefinitely and still provide the required subcooling margin in the primary system. Inability to control the system pressure and maintain subcooling under conditions of natural circulation flow in the primary system could lead to a loss of single phase natural circulation and decreased capability to remove core decay heat.	
	ITS:	**NUREG:**
	B 3.04.09	B 3.04.09
	LCO 3.04.09 COND B	LCO 3.04.09 COND B
	LCO 3.04.09 COND B RA B.1	LCO 3.04.09 COND B RA B.1
	SR 3.04.09.02	SR 3.04.09.02
04 Rev. A	SR 3.4.9.1 is modified to require that a pressurizer water level of less than or equal to 50.8% be verified every 12 hours in MODE 1. This level requirement is consistent with initial condition assumptions used in the accident analysis for the loss of normal feedwater as described in FSAR Section 14. The results of the accident analysis show that there is a high probability that the pressurizer would become water solid in the event that the accident assumed an initial pressurizer water level of 92%, as included in the ISTS. The requirement is also modified to require that a pressurizer water level of less than or equal to 95% be verified every 12 hours in MODE 2 or MODE 3. A higher water level is necessary in the pressurizer during cooldown to maintain pressurizer cooldown limits.	
	ITS:	**NUREG:**
	B 3.04.09	B 3.04.09
	SR 3.04.09.01	SR 3.04.09.01
05 Rev. A	The brackets have been removed and the proper plant specific information has been provided. In some instances, even though the information was designated as being site specific information in the LCO (bracketed), the corresponding Bases information was not bracketed. These cases are self evident, corresponding to the bracketed information in the LCO, and have had the appropriate site specific information provided.	
	ITS:	**NUREG:**
	B 3.04.09	B 3.04.09
	N/A	SR 3.04.09.03
	SR 3.04.09.02	SR 3.04.09.02

Justification For Deviations - NUREG-1431 Section 3.04.09

01-Aug-00

JFD Number	JFD Text
06 Rev. A	A sentence was added to the LCO and surveillance requirements discussion of the LCO 3.4.9 Bases to state that the required pressurizer heaters are heaters that are powered from a safeguards bus. This sentence was added to identify an attribute for the required pressure heaters at Point Beach because the NUREG-1431 Bases did not specify any criteria. The Point Beach design contains 5 banks of pressurizer heaters (banks A, B, C , D and E). Bank E is considered the control bank and the other banks are considered backup banks. Bank C, D, and E are powered from safeguards buses. Therefore, specifying this attribute in the Bases is appropriate to avoid any confusion with respect to identifying the required pressurizer heaters. In addition, "design rating" was changed to "have a combined capacity of >= 100kW" in the SR 3.4.9.2 Bases discussion. The important parameter to verify via this SR is to ensure that the combined capacity of the heaters is >= 100 kW (the design bases of the system), not to ensure that they can meet their respective design ratings. Therefore, the SR 3.4.9.2 Bases was changed accordingly.

ITS:	NUREG:
B 3.04.09	B 3.04.09

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.9 Pressurizer

1 B RAI 3.4.9-1

LCO 3.4.9 The pressurizer shall be OPERABLE with: Replace with Insert 3.4.9-1.

~~a. Pressurizer water level ≤ [92]%; and~~

~~b. Two groups of pressurizer heaters OPERABL E with the capacity of each group ≥ [125] kW [and capable of being powered from an emergency power supply].~~

APPLICABILITY: MODES 1, 2, and 3.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Pressurizer water level not within limit in MODE 1 (2)	A.1 Be in MODE 3 with reactor trip breakers open. Restore presurizer water level to within limit. (2) ~~AND~~ ~~A.2 Be in MODE 4.~~	6 hours 1 hour (2) B RAI 3.4.9-2 ~~12 hours~~
B. ~~One~~ required ~~group of~~ pressurizer heaters inoperable. (3)	B.1 Restore required ~~group of~~ pressurizer heaters to OPERABLE status. (3)	72 hours 1 hour
C. Required Action and associated Completion Time ~~of Condition B~~ not met. (2) OR Pressurizer water level not within limit in MODES 2 and 3.	C.1 Be in MODE 3. AND C.2 Be in MODE 4.	6 hours 12 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.4.9.1	Verify pressurizer water level is ≤ [92]% 50.8% in MODE 1 OR ≤ 95% in MODES 2 and 3.	12 hours
SR 3.4.9.2	Verify capacity of ~~each~~ required ~~group of~~ pressurizer heaters is ≥ [125] 100 kW.	92 days
~~SR 3.4.9.3~~	~~Verify required pressurizer heaters are capable of being powered from an emergency power supply.~~	~~[18] months~~

4

3

5

5

5 92 days [18] months

Approved TSTF-93 R.3

Insert 3.4.9-1

a.	Pressurizer water level ≤ 50.8% in MODE 1 or ≤ 95% in MODES 2 and 3; and
b.	At least 100 kW of pressurizer heaters capable of being powered from an emergency power supply are OPERABLE.

B

RAI 3.4.9-1

Insert B3.4.9-1

The LCO requirement for the pressurizer to be OPERABLE with a water level of ≤ 50.8% in MODE 1, and ≤ 95% in MODE 2 and MODE 3, ensures that a steam bubble exists. The pressurizer water level of ≤ 50.8% in MODE 1 is consistent with the assumptions used in the accident analyses. The water level of ≤ 95% in MODE 2 and MODE 3 is adequate protection for the pressurizer when a loss of normal feedwater is not a concern. A higher water level is necessary in the pressurizer during cooldown to maintain pressurizer cooldown limits. Limiting the LCO maximum operating water level preserves the steam space for pressure control. The LCO has been established to ensure the capability to establish and maintain pressure control for steady state operation and to minimize the consequences of potential overpressure transients. Requiring the presence of a steam bubble is also consistent with analytical assumptions.

The LCO requires a capacity of ≥ 100 kW of OPERABLE pressurizer heaters. The required pressurizer heaters are heaters that are powered from a safeguards bus. The minimum heater capacity required is sufficient to maintain the RCS near normal operating pressure when accounting for heat losses through the pressurizer insulation. By maintaining the pressure near the operating conditions, a wide margin to subcooling can be obtained in the loops. The amount needed to maintain pressure is dependent on the heat losses.

Insert B3.4.9-2

To achieve this status, the pressurizer water level must be restored to within limit within 1 hour. The Completion Time is reasonable based on the availability of indicators in the control room and alarms when pressurizer water level is above the programmed band. It is unlikely that exceeding the pressurizer water level requirement would result in an immediate threat of taking the pressurizer solid. Therefore, 1 hour is allowed to restore the pressurizer water level to within limit.

B
RAI 3.4.9-2

Insert B3.4.9-3

Without redundant sources of pressurizer heaters available, the hot, high pressure condition cannot be maintained indefinitely and still provide the required subcooling margin in the primary system. Inability to control the system pressure and maintain subcooling under conditions of natural circulation flow in the primary system could lead to a loss of single phase natural circulation and decreased capability to remove core decay heat.

Insert B3.4.9-4

If the pressurizer cannot be restored to OPERABLE status within the associated Completion Time of Required Action A.1 or B.1, or the pressurizer water level is not within the limit of MODE 2 and MODE 3, the plant must be brought to a MODE in which the LCO does not apply.

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.9 Pressurizer

LCO 3.4.9 The pressurizer shall be OPERABLE with:

a. Pressurizer water level ≤ 50.8% in MODE 1 or ≤ 95% in MODES 2 and 3; and

b. At least 100 kW of pressurizer heaters capable of being powered from an emergency power supply are OPERABLE.

B RAI 3.4.9-1

APPLICABILITY: MODES 1, 2, and 3.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Pressurizer water level not within limit in MODE 1.	A.1 Restore pressurizer water level to within limit.	1 hour
B. Required pressurizer heaters inoperable.	B.1 Restore required pressurizer heaters to OPERABLE status.	1 hour
C. Required Action and associated Completion Time not met. OR Pressurizer water level not within limit in MODES 2 and 3.	C.1 Be in MODE 3. AND C.2 Be in MODE 4.	6 hours 12 hours

B RAI 3.4.9-2

B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.9 Pressurizer

BASES

BACKGROUND

The pressurizer provides a point in the RCS where liquid and vapor are maintained in equilibrium under saturated conditions for pressure control purposes to prevent bulk boiling in the remainder of the RCS. Key functions include maintaining required primary system pressure during steady state operation, and limiting the pressure changes caused by reactor coolant thermal expansion and contraction during normal load transients.

The pressure control components addressed by this LCO include the pressurizer water level, the required heaters, and their controls. Pressurizer safety valves and pressurizer power operated relief valves are addressed by LCO 3.4.10, "Pressurizer Safety Valves," and LCO 3.4.11, "Pressurizer Power Operated Relief Valves (PORVs)," respectively.

The intent of the LCO is to ensure that a steam bubble exists in the pressurizer prior to power operation to minimize the consequences of potential overpressure transients. The presence of a steam bubble is consistent with analytical assumptions. Relatively small amounts of noncondensible gases can inhibit the condensation heat transfer between the pressurizer spray and the steam, and diminish the spray effectiveness for pressure control.

Electrical immersion heaters, located in the lower section of the pressurizer vessel, keep the water in the pressurizer at saturation temperature and maintain a constant operating pressure. A minimum required available capacity of pressurizer heaters ensures that the RCS pressure can be maintained. The capability to maintain and control system pressure is important for maintaining subcooled conditions in the RCS and ensuring the capability to remove core decay heat by either forced or natural circulation of reactor coolant. Unless adequate heater capacity is available, the hot, high pressure condition cannot be maintained indefinitely and still provide the required subcooling margin in the primary system. Inability to control the system pressure and maintain subcooling under conditions of natural circulation flow in the primary system could lead to a loss of single phase natural circulation and decreased capability to remove core decay heat.

BASES

APPLICABLE SAFETY ANALYSES

In MODES 1, 2, and 3, the LCO requirement for a steam bubble is reflected implicitly in the accident analyses. Safety analyses performed for lower MODES are not limiting. All analyses performed from a critical reactor condition assume the existence of a steam bubble and saturated conditions in the pressurizer. In making this assumption, the analyses neglect the small fraction of noncondensible gases normally present.

Safety analyses presented in the FSAR (Ref. 1) do not take credit for pressurizer heater operation; however, an implicit initial condition assumption of the safety analyses is that the RCS is operating at normal pressure.

The maximum pressurizer water level limit satisfies Criterion 2 of the NRC Policy Statement. Although the heaters are not specifically used in accident analysis, the need to maintain subcooling in the long term during loss of offsite power, as indicated in NUREG-0737 (Ref. 2), is the reason for providing an LCO.

LCO

The LCO requirement for the pressurizer to be OPERABLE with a water level of ≤ 50.8% in MODE 1, and ≤ 95% in MODE 2 and MODE 3, ensures that a steam bubble exists. The pressurizer water level of ≤ 50.8% in MODE 1 is consistent with the assumptions used in the accident analyses. The water level of ≤ 95% in MODE 2 and MODE 3 is adequate protection for the pressurizer when a loss of normal feedwater is not a concern. A higher water level is necessary in the pressurizer during cooldown to maintain pressurizer cooldown limits. Limiting the LCO maximum operating water level preserves the steam space for pressure control. The LCO has been established to ensure the capability to establish and maintain pressure control for steady state operation and to minimize the consequences of potential overpressure transients. Requiring the presence of a steam bubble is also consistent with analytical assumptions.

The LCO requires a capacity of ≥ 100 kW of OPERABLE pressurizer heaters. The required pressurizer heaters are heaters that are powered from a safeguards bus. The minimum heater capacity required is sufficient to maintain the RCS near normal operating pressure when accounting for heat losses through the pressurizer insulation. By maintaining the pressure near the operating conditions, a wide margin to subcooling can be obtained in the loops. The amount needed to maintain pressure is dependent on the heat losses.

BASES

APPLICABILITY

The need for pressure control is most pertinent when core heat can cause the greatest effect on RCS temperature, resulting in the greatest effect on pressurizer level and RCS pressure control. Thus, applicability has been designated for MODES 1 and 2. The applicability is also provided for MODE 3. The purpose is to prevent solid water RCS operation during heatup and cooldown to avoid rapid pressure rises caused by normal operational perturbation, such as reactor coolant pump startup.

In MODES 1, 2, and 3, there is need to maintain the availability of pressurizer heaters. In the event of a loss of offsite power, the initial conditions of these MODES give the greatest demand for maintaining the RCS in a hot pressurized condition with loop subcooling for an extended period. For MODE 4, 5, or 6, it is not necessary to control pressure (by heaters) to ensure loop subcooling for heat transfer when the Residual Heat Removal (RHR) System is in service and, therefore, the LCO is not applicable.

ACTIONS

A.1 and A.2

Pressurizer water level control malfunctions or other plant evolutions may result in a pressurizer water level above the nominal upper limit, even with the plant at steady state conditions.

If the pressurizer water level is not within the limit in MODE 1, action must be taken to restore the plant to operation within the bounds of the safety analyses. To achieve this status, the pressurizer water level must be restored to within limit within 1 hour. The Completion Time is reasonable based on the availability of indicators in the control room and alarms when pressurizer water level is above the programmed band. It is therefore unlikely that exceeding the pressurizer water level requirement would result in an immediate threat of taking the pressurizer solid. Therefore, 1 hour are allowed to restore the pressurizer water level to within limit. B RAI 3.4.9-2

B.1

If the required pressurizer heaters are inoperable, restoration is required within 1 hour. Without redundant sources of pressurizer heaters available, the hot, high pressure condition cannot be maintained indefinitely and still provide the required subcooling margin in the primary system. Inability to control the system pressure and maintain subcooling under conditions of natural circulation flow in the primary system could lead to a loss of single phase natural circulation and decreased capability to remove core decay heat.
in an orderly manner and without challenging plant systems.

BASES

ACTIONS (continued)

C.1 and C.2

If the pressurizer cannot be restored to OPERABLE status within the associated Completion Time of Required Action A.1 or B.1, or the pressurizer water level is not within the limit of MODE 2 and MODE 3, the plant must be brought to a MODE in which the LCO does not apply. To achieve this status, the plant must be brought to MODE 3 within 6 hours and to MODE 4 within 12 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.4.9.1

This SR requires that during steady state operation, pressurizer level is maintained below the nominal upper limit to provide a minimum space for a steam bubble. The Surveillance is performed by observing the indicated level. The Frequency of 12 hours corresponds to verifying the parameter each shift. The 12 hour interval has been shown by operating practice to be sufficient to regularly assess level for any deviation and verify that operation is within safety analyses assumptions. Alarms are also available for early detection of abnormal level indications.

SR 3.4.9.2

The required pressurizer heaters are heaters that are powered from a safeguards bus. The SR is satisfied when the power supplies are demonstrated to be capable of producing the minimum power and the associated pressurizer heaters are verified to have a combined capacity of ≥ 100kW. This may be done by testing the power supply output and by performing an electrical check on heater element continuity and resistance. The Frequency of 92 days is considered adequate to detect heater degradation and has been shown by operating experience to be acceptable.

REFERENCES

1. FSAR, Section 14.

2. NUREG-0737, November 1980.

Description of Changes - NUREG-1431 Section 3.04.10

01-Aug-00

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

CTS:	ITS:
15.03.01.A.04.B	LCO 3.04.10
15.04.01 T 15.04.01-02 11	SR 3.04.10.01

DOC Number	DOC Text
L.01 Rev. B	CTS 15.3.1.A.4.b requires both pressurizer safety valves to be operable whenever the reactor is critical, but does not provide any actions if this LCO is not satisfied. Therefore, in accordance with CTS 15.3.0.b, the plant is placed in a non-applicable mode in 7 hours. Proposed ITS 3.4.10, Condition A, is entered whenever a pressurizer safety valve is inoperable. Condition A Actions require the restoration of the valve to an operable status within 15 minutes. The Completion Time of 15 minutes reflects the importance of maintaining the RCS overpressure protection system. An inoperable pressurizer safety valve coincident with an RCS overpressure event could challenge the integrity of the pressure boundary. In the event the pressurizer safety valve cannot be restored within 15 minutes, or both pressurizer safety valves are inoperable, Condition B is entered. Condition B Actions require the plant to be placed in MODE 3 in 6 hours and MODE 4 with any RCS cold leg temperature < the LTOP enabling temperature specified in the PTLR in 12 hours. These actions result in placing the plant in a non-applicable mode in 12 hours. The Completion Time of 12 hours is based on operating experience to reach the required plant condition from a full power condition in an orderly manner and without challenging plant systems. Extending the time allowed to place the plant in a non-applicable mode from 7 hours to 12 hours is less restrictive. This is acceptable, based on the broader LCO Applicability adopted as part of ITS 3.4.10, and the increased time required to place the plant in a non-applicable mode from full power conditions.

CTS:	ITS:
NEW	LCO 3.04.10 COND A
	LCO 3.04.10 COND A RA A.1
	LCO 3.04.10 COND B
	LCO 3.04.10 COND B RA B.1
	LCO 3.04.10 COND B RA B.2

Description of Changes - NUREG-1431 Section 3.04.10

01-Aug-00

DOC Number	DOC Text
L.02 Rev. B	CTS Specification 15.3.1.A.4.b, which requires that both pressurizer safety valves be operable when the reactor is critical, is revised to add ITS LCO 3.4.10 NOTE, which allows the safety valve lift settings to be outside the LCO limits for the purpose of setting the safety valves under ambient (hot) conditions. Because this note allows the pressurizer safety valves to be potentially inoperable in MODE 3 and MODE 4 until the safety valves can be tested and set, this change is less restrictive. This change is acceptable because the limitations included in the note (i.e., a maximum of 36 hours allowed following entry into MODE 3) assure that reactor decay heat is significantly reduced below the assumptions in the applicable safety analyses for LCO 3.4.10. This permits testing and examination of the safety valves at high pressure and temperature near their normal operating range, but only after the valves have had a preliminary cold setting. The cold setting gives assurance that the valves are OPERABLE near their design condition. Only one valve at a time will be removed from service for testing. The 36 hour exception is reasonable based on 18 hour outage time for each of the valves. The 18 hour period is derived from operating experience that hot testing can be performed in this time frame. **CTS:** NEW — **ITS:** LCO 3.04.10 NOTE
L.03 Rev. B	As described in TSCR 219, Adoption of PTLR and revised P-T and LTOP Limits, Attachment 1, Description of proposed Change #4. **CTS:** 15.03.01.A.04.A — **ITS:** N/A
LA.01 Rev. A	CTS 15.4.1, Table 15.4.1-2, item 11, requires that pressurizer safety valve setpoints be checked at a frequency of "every five years." The frequency is modified by Note (11), which specifies "An approximately equal number of valves shall be tested each refueling outage such that all valves will be tested within a five year period. If any valve fails its tests, an additional number of valves equal to the number originally tested shall be tested. If any of the additional tested valves fail, all remaining valves shall be tested." These details have been moved from the Technical Specification to licensee control as these details are not necessary to describe the actual regulatory requirement. Therefore, proposed ITS SR 3.4.10.1 requires verifying "each pressurizer safety valve is OPERABLE in accordance with the Inservice Testing Program", at a frequency of "In accordance with the Inservice Testing Program." The testing details located in CTS 15.4.1, Table 15.4.1-2, item 11, are not required to be in the ITS to provide adequate protection of public health and safety, as the regulatory requirement (IST Program) is being maintained in the Technical Specifications. Changes to plant procedures and other plant controlled documents are subject to controls imposed by plant administrative procedures, which endorse applicable regulations and standards. Inservice Testing of pressurizer safety valves will continue to be performed in accordance with the IST Program. **CTS:** 15.04.01 T 15.04.01-02 11 — **ITS:** SR 3.04.10.01 **CTS:** 15.04.01 T 15.04.01-02 11 (11) — **ITS:** N/A

Description of Changes - NUREG-1431 Section 3.04.10

01-Aug-00

DOC Number	DOC Text	
M.01 Rev. A	CTS 15.3.1.A.4.b requires both pressurizer safety valves to be operable. Proposed ITS 3.4.10 requires two pressurizer safety valves to be operable with lift settings greater than or equal to 2410 psig and less than or equal to 2560 psig. The pressurizer safety valve settings are not stated in the CTS, but are maintained in the ITS Program. Stating the safety valve settings in ITS LCO 3.4.10 is more restrictive and is consistent with NUREG 1431.	
	CTS:	**ITS:**
	15.03.01.A.04.B	LCO 3.04.10
M.02 Rev. B	As described in TSCR 219, Adoption of PTLR and revised P-T and LTOP Limits, Attachment 1, Description of proposed Change #4.	
	CTS:	**ITS:**
	15.03.01.A.04.B	LCO 3.04.10
R.01 Rev. B	Not Used.	
	CTS:	**ITS:**
	N/A	N/A

(c) Residual Heat Removal Loop (A)*

(d) Residual Heat Removal Loop (B)*

(2) If the conditions of specification (1) above cannot be met, corrective action to return a second decay heat removal method to operable status as soon as possible shall be initiated immediately.

(3) If no decay heat removal method is in operation, except as permitted by (4) below, all operations causing an increase in the reactor decay heat load or a reduction in reactor coolant system boron concentration shall be suspended. Corrective actions to return a decay heat removal method to operation shall be initiated immediately.

(4) At least one of the above decay heat removal methods shall be in operation.

(a) All reactor coolant pumps and residual heat removal pumps may be deenergized for up to 1 hour in any 8 hour period provided:

< See LCO 3.4.6 >

(1) No operations are permitted that would cause dilution of reactor coolant system boron concentration, and

(2) Core outlet temperature is maintained at least 10°F below saturation temperature.

b. Reactor Coolant Temperature Less Than 140°F

< See LCOs 3.4.7, 3.4.8, 3.9.5 & 3.9.6 >

(1) Both residual heat removal loops shall be operable except as permitted in items (3) or (4) below.

(2) If no residual heat removal loop is in operation, all operations causing an increase in the reactor decay heat load or a reduction in reactor coolant system boron concentration shall be suspended. Corrective actions to return a decay heat removal method to operation shall be initiated immediately.

(3) One residual heat removal loop may be out of service when the reactor vessel head is removed and the refueling cavity flooded.

(4) One of the two residual heat removal loops may be temporarily out of service to meet surveillance requirements.

4. Pressurizer Safety Valves

L.3

~~a. At least one pressurizer safety valve shall be operable whenever the reactor head is on the vessel.~~

Replace with LCO 3.4.10. See Insert 3.4.10-1.

M.1

b. Both pressurizer safety valves shall be operable whenever the reactor ~~is critical~~

L.2

coolant temperature is greater than or equal to the LTOP enable temperature in the PTLR.

M.2

L.1

Add Actions A & B. See Insert 3.4.10-2.

B

PTLR

*Mechanical design provisions of the residual heat removal system afford the necessary flexibility to allow an operable residual heat removal loop to consist of the RHR pump from one loop coupled with the RHR heat exchanger from the other loop. Electrical design provisions of the residual heat removal system afford the necessary flexibility to allow the normal or emergency power source to be inoperable or tied together when the reactor coolant temperature is less than 200°F.

< See LCO 3.4.6 >

A.1

TABLE 15.4.1-2 (Continued)

< See LCOs 3.7.15 and 3.7.16 >

		Test	Frequency
7.	Spent Fuel Pit	a) Boron Concentration b) Water Level Verification	Monthly Weekly
8.	Secondary Coolant	Gross Beta-gamma Activity or gamma isotopic analysis Iodine concentration	Weekly[6] Weekly when gross Beta-gamma activity equals or exceeds 1.0 μCi/g [6]
9.	Control Rods	a) Rod drop times of all full length rods [3] b) Rodworth measurement	Each refueling or after maintenance that could affect proper functioning [4] Following each refueling shutdown prior to commencing power operation
10.	Control Rod	Partial movement of all rods	Every 2 weeks [18]
11.	Pressurizer Safety Valves	Set point	Every five years [11]
12.	Main Steam Safety Valves	Set Point	Every five years [11]
13.	Containment Isolation Trip	Functioning	Each refueling shutdown
14.	Refueling System Interlocks	Functioning	Each refueling shutdown
15.	Service Water System	Functioning	Each refueling shutdown
16.	Primary System Leakage	Evaluate	Monthly [6]
17.	Diesel Fuel Supply	Fuel inventory	Daily
18.	Deleted		
19.	Deleted		
20.	Boric Acid System	Storage Tank and piping temperatures ≥ temperature required by Table 15.3.2-1	Daily[19]

< See LCO 3.7.18 >

< See LCO 3.1.5 >

Replace with SR 3.4.10.1. See Insert 3.4.10-3. LA.1

B
PTLR

< See LCO 3.7.1 >

< See LCO 3.6.3 and 3.7.2 >

< See LCO 3.9.1 >

< See LCO 3.7.8 >

< See LCO 3.4.13 >

< See LCO 3.8.3 >

< See LCO 3.5.2 >

Section 3.4.10 CTS Markup Inserts

M.1

Insert 3.4.10-1:

LCO 3.4.10 — Two pressurizer safety valves shall be OPERABLE with lift settings ≥ 2410 psig and ≤ 2560 psig.

APPLICABILITY: MODES 1, 2, and 3,
MODE 4 with all RCS cold leg temperatures > the LTOP enabling temperature specified in the PTLR

B
PTLR

L.2

-------------------------NOTE-------------------------
The lift settings are not required to be within the LCO limits during MODES 3 and 4 for the purpose of setting the pressurizer safety valves under ambient (hot) conditions. This exception is allowed for 36 hours following entry into MODE 3 provided a preliminary cold setting was made prior to heatup.

Insert 3.4.10-2:

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One pressurizer safety valve inoperable.	A.1 Restore valve to OPERABLE status.	15 minutes
B. Required Action and associated Completion Time not met. OR Two pressurizer safety valves inoperable.	B.1 Be in MODE 3. AND B.2 Be in MODE 4 with any RCS cold leg temperature < the LTOP enabling temperature specified in the PTLR.	6 hours 12 hours

L.1

B
PTLR

Insert 3.4.10-3:

SURVEILLANCE	FREQUENCY
SR 3.4.10.1 Verify each pressurizer safety valve is OPERABLE in accordance with the Inservice Testing Program. Following testing, lift settings shall be ≥ 2440.71 psig and ≤ 2551.25 psig	In accordance with the Inservice Testing Program

LA.1

Justification For Deviations - NUREG-1431 Section 3.04.10

01-Aug-00

JFD Number	JFD Text	
01 Rev. A	The brackets have been removed and the proper plant specific information has been provided. In some instances, even though the information was designated as being site specific information in the LCO (bracketed), the corresponding Bases information was not bracketed. These cases are self evident, corresponding to the bracketed information in the LCO, and the have had the appropriate site specific information provided.	
	ITS:	**NUREG:**
	B 3.04.10	B 3.04.10
	LCO 3.04.10	LCO 3.04.10
	LCO 3.04.10 NOTE	LCO 3.04.10 NOTE
02 Rev. A	ITS Specification 3.4.10 is modified to reflect a safety valve operability setpoint tolerance of +/- 3% to allow for drift, in accordance with Section III of the ASME Boiler and Pressure Vessel Code.	
	ITS:	**NUREG:**
	B 3.04.10	B 3.04.10
	LCO 3.04.10	LCO 3.04.10
03 Rev. B	Not Used.	
	ITS:	**NUREG:**
	N/A	N/A
04 Rev. B	The actual numerical values for an LTOP enabling temperature are replaced with a reference to the temperature specified in the PTLR. The LTOP enabling temperature will then be calculated and controlled by the licensee in accordance with the topical reports identified in the PTLR.	
	ITS:	**NUREG:**
	B 3.04.10	B 3.04.10
	LCO 3.04.10	LCO 3.04.10
	LCO 3.04.10 COND B RA B.2	LCO 3.04.10 COND B RA B.2
05 Rev. A	Consistent with the range specified in PBNP calculation 98-0096, as tested lift setting of the pressurizer safety valves (+2.67% / -1.78%), SR 3.4.10.1 is modified to specify a pressurizer safety valve lift setting of greater than or equal to 2440.71 psig and less than or equal to 2551.25 psig.	
	ITS:	**NUREG:**
	B 3.04.10	B 3.04.10
	SR 3.04.10.01	SR 3.04.10.01

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.10 Pressurizer Safety Valves.

Two 1

LCO 3.4.10 [Three] pressurizer safety valves shall be OPERABLE with lift settings ≥ [2460] psig and ≤ [2510] psig. 4

2410 2 2560 the LTOP enabling temperature specified in the PTLR

APPLICABILITY: MODES 1, 2, and 3,
MODE 4 with all RCS cold leg temperatures > ~~[275]°F~~. B PTLR

--------------------------NOTE--------------------------
The lift settings are not required to be within the LCO limits during MODES 3 and 4 for the purpose of setting the pressurizer safety valves under ambient (hot) conditions. This exception is allowed for [54] hours following entry into MODE 3 provided a preliminary cold setting was made prior to heatup.
----

36 1

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One pressurizer safety valve inoperable.	A.1 Restore valve to OPERABLE status.	15 minutes
B. Required Action and associated Completion Time not met. OR 1 Two ~~or more~~ pressurizer safety valves inoperable.	B.1 Be in MODE 3. AND B.2 Be in MODE 4 with any RCS cold leg temperatures ≤ ~~[275]°F~~.	6 hours 12 hours

the LTOP enabling temperature specified in the PTLR 4

B PTLR

B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.10 Pressurizer Safety Valves

BASES

BACKGROUND

The pressurizer safety valves provide, in conjunction with the Reactor Protection System, overpressure protection for the RCS. The pressurizer safety valves are totally enclosed pop type, spring loaded, self actuated valves with backpressure compensation. The safety valves are designed to prevent the system pressure from exceeding the system Safety Limit (SL), [2735] psig, which is 110% of the design pressure.

2734 ◄ 1

Because the safety valves are totally enclosed and self actuating, they are considered independent components. The relief capacity for each valve, [380,000] lb/hr, is based on postulated overpressure transient conditions resulting from a complete loss of steam flow to the turbine. This event results in the maximum surge rate into the pressurizer, which specifies the minimum relief capacity for the safety valves. The discharge flow from the pressurizer safety valves is directed to the pressurizer relief tank. This discharge flow is indicated by an increase in temperature downstream of the pressurizer safety valves or increase in the pressurizer relief tank temperature or level.

1 ► 288,000

4

the LTOP enabling temperature specified in the PTLR

Overpressure protection is required in MODES 1, 2, 3, 4, and 5; however, in MODE 4, with one or more RCS cold leg temperatures ≤ [275]°F, and MODE 5 and MODE 6 with the reactor vessel head on, overpressure protection is provided by operating procedures and by meeting the requirements of LCO 3.4.12, "Low Temperature Overpressure Protection (LTOP) System."

B
PTLR

The pressurizer safety valve setpoint is ± 3% for OPERABILITY; however, the valves are reset to +2.67% / -1.78% during surveillance to allow for drift and account for the ambient conditions associated with MODES 1, 2 and 3.

2 5

The upper and lower pressure limits are based on the ± 1% tolerance requirement (Ref. 1) for lifting pressures above 1000 psig. The lift setting is for the ambient conditions associated with MODES 1, 2, and 3. This requires either that the valves be set hot or that a correlation between hot and cold settings be established.

The pressurizer safety valves are part of the primary success path and mitigate the effects of postulated accidents. OPERABILITY of the safety valves ensures that the RCS pressure will be limited to 110% of design pressure.

BASES

LCO (continued)

The limit protected by this Specification is the reactor coolant pressure boundary (RCPB) SL of 110% of design pressure. Inoperability of one or more valves could result in exceeding the SL if a transient were to occur. The consequences of exceeding the ASME pressure limit could include damage to one or more RCS components, increased leakage, or additional stress analysis being required prior to resumption of reactor operation.

1

APPLICABILITY

4 enabling two

In MODES 1, 2, and 3, and portions of MODE 4 above the LTOP arming temperature, OPERABILITY of [three] valves is required because the combined capacity is required to keep reactor coolant pressure below 110% of its design value during certain accidents. MODE 3 and portions of MODE 4 are conservatively included, although the listed accidents may not require the safety valves for protection.

LTOP enabling temperature specified in the PTLR

4

The LCO is not applicable in MODE 4 when all RCS cold leg temperatures are ≤ ~~[275]°F~~ or in MODE 5 because LTOP is provided. Overpressure protection is not required in MODE 6 with reactor vessel head detensioned.

B PTLR

The Note allows entry into MODES 3 and 4 with the lift settings outside the LCO limits. This permits testing and examination of the safety valves at high pressure and temperature near their normal operating range, but only after the valves have had a preliminary cold setting. The cold setting gives assurance that the valves are OPERABLE near their design condition. Only one valve at a time will be removed from service for testing. The [54] hour exception is based on 18 hour outage time for each of the [three] valves. The 18 hour period is derived from operating experience that hot testing can be performed in this timeframe.

36

1

1 two

ACTIONS

A.1

With one pressurizer safety valve inoperable, restoration must take place within 15 minutes. The Completion Time of 15 minutes reflects the importance of maintaining the RCS Overpressure Protection System. An inoperable safety valve

BASES

ACTIONS

A.1 (continued)

coincident with an RCS overpressure event could challenge the integrity of the pressure boundary.

B.1 and B.2

1

If the Required Action of A.1 cannot be met within the required Completion Time or if two ~~or more~~ pressurizer safety valves are inoperable, the plant must be brought to a MODE in which the requirement does not apply. To achieve this status, the plant must be brought to at least MODE 3 within 6 hours and to MODE 4 with any RCS cold leg temperatures ≤ ~~[275]°F~~ within 12 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems. With any RCS cold leg temperatures at or below [275] °F, overpressure protection is provided by the LTOP System. The change from MODE 1, 2, or 3 to MODE 4 reduces the RCS energy (core power and pressure), lowers the potential for large pressurizer insurges, and thereby removes the need for overpressure protection by [three] pressurizer safety valves.

B PTLR

the LTOP enabling temperature specified in the PTLR

4

two

1

SURVEILLANCE REQUIREMENTS

SR 3.4.10.1

SRs are specified in the Inservice Testing Program. Pressurizer safety valves are to be tested in accordance with the requirements of Section XI of the ASME Code (Ref. 4), which provides the activities and Frequencies necessary to satisfy the SRs. No additional requirements are specified.

1

3

The pressurizer safety valve setpoint is ± [3]% for OPERABILITY; however, the valves are reset to ± 1% during the Surveillance to allow for drift.

5

+2.67% / -1.78%

REFERENCES

1. ASME, Boiler and Pressure Vessel Code, Section III.

2. FSAR, Chapter [15]

14

1

No Significant Hazards Considerations - NUREG-1431 Section 3.04.10

01-Aug-00

NSHC Number	NSHC Text
A Rev. A	In accordance with the criteria set forth in 10 CFR 50.92, PBNP has evaluated this proposed Technical Specifications change and determined it does not represent a significant hazards consideration. The following is provided in support of this conclusion. 1. Does the change involve a significant increase in the probability or consequences of an accident previously evaluated? The proposed change involves reformatting and rewording of the current Technical Specifications. The reformatting and rewording process involves no technical changes to existing requirements. As such, this change is administrative in nature and does not impact initiators of analyzed events or assumed mitigation of accident or transient events. Therefore, this change does not increase the probability or consequences of an accident previously evaluated. 2. Does the change create the possibility of a new or different kind of accident from any accident previously evaluated? The proposed change does not require a physical alteration of the plant (no new or different type of equipment will be installed) or changes in parameters governing normal plant operation. The proposed change will not impose any new or eliminate any old requirements. Thus, this change does not create the possibility of a new or different kind of accident from any accident previously evaluated. 3. Does this change involve a significant reduction in a margin of safety? The proposed change will not significantly reduce the margin of safety because it has no impact on any safety analysis assumptions. This change is administrative. As such, there is no technical change to the requirements and, therefore, there is no reduction in the margin of safety.

NSHC Number	NSHC Text
L.01 Rev. A	In accordance with the criteria set forth in 10 CFR 50.92, PBNP has evaluated this proposed Technical Specifications change and determined it does not represent a significant hazards consideration. The following is provided in support of this conclusion. 1. Does the change involve a significant increase in the probability or consequences of an accident previously evaluated? The proposed change 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 adopts ISTS LCO 3.4.10 Actions A and B. These actions result in extending time allowed to place the plant in a mode in which the requirement does not apply from 7 hours to 12 hours. The Completion Time of 12 hours is based on operating experience to reach the required plant condition from a full power condition in an orderly manner and without challenging plant systems. This relaxation is acceptable, based on the broader LCO Applicability adopted as part of ITS 3.4.10, and the increased time required to place the plant in a non-applicable mode from full power conditions. The Completion Time is consistent with the time allowed by ITS LCO 3.0.3 to bring the plant to a Hot Shutdown condition from full power operation. 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 pressurizer safety valves are properly maintained. Therefore, this change does not involve a 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. This change adopts LCO 3.4.10 NOTE, allowing the pressurizer safety valve lift settings to be outside the LCO limits for the purpose of setting the safety valves under ambient (hot) conditions. Only one valve at a time will be removed from service for testing. This NOTE allows a maximum of 36 hours following entry into MODE 3 for the exception, provided a preliminary cold setting was made prior to heatup. This assures that reactor decay heat is significantly below the assumptions in the applicable safety analyses for LCO 3.4.10. 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 pressurizer safety valves are properly maintained. Therefore, this change does not involve a reduction in a margin of safety.

NSHC Number	NSHC Text
LA Rev. A	In accordance with the criteria set forth in 10 CFR 50.92, PBNP has evaluated this proposed Technical Specifications change and determined it does not represent a significant hazards consideration. The following is provided in support of this conclusion. 1. Does the change involve a significant increase in the probability or consequences of an accident previously evaluated? The proposed change relocates requirements from the Technical Specifications to the Bases, FSAR, or other plant controlled documents. The Bases and FSAR will be maintained using the provisions of 10 CFR 50.59. In addition to 10 CFR 50.59 provisions, the Technical Specifications Bases are subject to the change process in the Administrative Controls Chapter of the ITS. Plant procedures and other plant controlled documents are subject to controls imposed by plant administrative procedures, which endorse applicable regulations and standards. Changes to the Bases, FSAR, or other plant controlled documents will be evaluated in accordance with the requirements of the Bases Control Program in Chapter 5.0 of the ITS, 10 CFR 50.59, or plant administrative processes. Therefore, no increase in the probability or consequences of an accident previously evaluated will be allowed. 2. Does the change create the possibility of a new or different kind of accident from any accident previously evaluated? The proposed change does not require a physical alteration of the plant (no new or different type of equipment will be installed) or changes in parameters governing normal plant operation. The proposed change will not impose any different requirements and adequate control of the information will be maintained. Thus, this change does not create the possibility of a new or different kind of accident from any accident previously evaluated. 3. Does this change involve a significant reduction in a margin of safety? The proposed change will not reduce a margin of safety because it has no impact on any safety analysis assumptions. In addition, the requirements to be moved from the Technical Specifications to the Bases, FSAR, or other plant controlled documents are as they currently exist. Future changes to the requirements in the Bases, FSAR, or other plant controlled documents will be evaluated in accordance with the requirements of 10 CFR 50.59, the Bases Control Program in Chapter 5.0 of the ITS, or the applicable plant process and no reduction in a margin of safety will be allowed.

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.
R Rev. B	Not Used.

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.10 Pressurizer Safety Valves

LCO 3.4.10 Two pressurizer safety valves shall be OPERABLE with lift settings $\geq$ 2410 psig and $\leq$ 2560 psig.

APPLICABILITY: MODES 1, 2, and 3,
MODE 4 with all RCS cold leg temperatures > the LTOP enabling temperature specified in the PTLR. B PTLR

------------------------------------NOTE------------------------------------
The lift settings are not required to be within the LCO limits during MODES 3 and 4 for the purpose of setting the pressurizer safety valves under ambient (hot) conditions. This exception is allowed for 36 hours following entry into MODE 3 provided a preliminary cold setting was made prior to heatup. B PTLR

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One pressurizer safety valve inoperable.	A.1 Restore valve to OPERABLE status.	15 minutes
B. Required Action and associated Completion Time not met. <u>OR</u> Two pressurizer safety valves inoperable.	B.1 Be in MODE 3. <u>AND</u> B.2 Be in MODE 4 with any RCS cold leg temperature $\leq$ the LTOP enabling temperature specified in the PTLR.	6 hours 12 hours

B PTLR

B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.10 Pressurizer Safety Valves

BASES

BACKGROUND

The pressurizer safety valves provide, in conjunction with the Reactor Protection System, overpressure protection for the RCS. The pressurizer safety valves are totally enclosed pop type, spring loaded, self actuated valves with backpressure compensation. The safety valves are designed to prevent the system pressure from exceeding the system Safety Limit (SL), 2734 psig, which is 110% of the design pressure.

Because the safety valves are totally enclosed and self actuating, they are considered independent components. The relief capacity for each valve, 288,000 lb/hr, is based on postulated overpressure transient conditions resulting from a complete loss of steam flow to the turbine. This event results in the maximum surge rate into the pressurizer, which specifies the minimum relief capacity for the safety valves. The discharge flow from the pressurizer safety valves is directed to the pressurizer relief tank. This discharge flow is indicated by an increase in temperature downstream of the pressurizer safety valves or increase in the pressurizer relief tank temperature or level.

Overpressure protection is required in MODES 1, 2, 3, 4, and 5; however, in MODE 4, with one or more RCS cold leg temperatures ≤ the LTOP enabling temperature specified in the PTLR, and MODE 5 and MODE 6 with the reactor vessel head on, overpressure protection is provided by operating procedures and by meeting the requirements of LCO 3.4.12, "Low Temperature Overpressure Protection (LTOP) System." B PTLR

The pressurizer safety valve setpoint is ± 3% for OPERABILITY; however, the valves are reset to +2.67%/-1.78% during surveillance to allow for drift and account for the ambient conditions associated with MODES 1, 2 and 3.

The pressurizer safety valves are part of the primary success path and mitigate the effects of postulated accidents. OPERABILITY of the safety valves ensures that the RCS pressure will be limited to 110% of design pressure.

The consequences of exceeding the American Society of Mechanical Engineers (ASME) pressure limit (Ref. 1) could include damage to RCS components, increased leakage, or a requirement to perform additional stress analyses prior to resumption of reactor operation.

BASES

APPLICABLE SAFETY ANALYSES

All accident and safety analyses in the FSAR (Ref. 2) that require safety valve actuation assume operation of two pressurizer safety valves to limit increases in RCS pressure. The overpressure protection analysis (Ref. 3) is also based on operation of two safety valves. Accidents that could result in overpressurization if not properly terminated include:

a. Uncontrolled rod withdrawal from full power;

b. Loss of reactor coolant flow;

c. Loss of external electrical load;

d. Loss of normal feedwater;

e. Loss of all AC power to station auxiliaries; and

f. Locked rotor.

Detailed analyses of the above transients are contained in Reference 2. Safety valve actuation is required in events c, d, and e (above) to limit the pressure increase. Compliance with this LCO is consistent with the design bases and accident analyses assumptions.

Pressurizer safety valves satisfy Criterion 3 of the NRC Policy Statement.

LCO

The two pressurizer safety valves are set to open at the RCS design pressure (2500 psia), and within the ASME specified tolerance, to avoid exceeding the maximum design pressure SL, to maintain accident analyses assumptions, and to comply with ASME requirements. The pressurizer safety valve setpoint is ± 3% for OPERABILITY; however, the valves are reset to +2.67%/-1.78% during surveillance to allow for drift. The limit protected by this Specification is the reactor coolant pressure boundary (RCPB) SL of 110% of design pressure. Inoperability of one or more valves could result in exceeding the SL if a transient were to occur. The consequences of exceeding the ASME pressure limit could include damage to one or more RCS components, increased leakage, or additional stress analysis being required prior to resumption of reactor operation.

APPLICABILITY

In MODES 1, 2, and 3, and portions of MODE 4 above the LTOP enabling temperature, OPERABILITY of two valves is required because the combined capacity is required to keep reactor coolant pressure below 110% of its design value during certain accidents. MODE 3 and portions of MODE 4 are conservatively included, although the listed accidents may not require the safety valves for protection.

B PTLR

The LCO is not applicable in MODE 4 when all RCS cold leg temperatures are ≤ the LTOP enabling temperature specified in the PTLR or in MODE 5 because LTOP is provided. Overpressure protection is not required in MODE 6 with reactor vessel head detensioned.

The Note allows entry into MODES 3 or 4 with the lift settings outside the LCO limits. This permits testing and examination of the safety valves at high pressure and temperature near their normal operating range, but only after the valves have had a preliminary cold setting. The cold setting gives assurance that the valves are OPERABLE near their design condition. Only one valve at a time will be removed from service for testing. The 36 hour exception is based on 18 hour outage time for each of the two valves. The 18 hour period is derived from operating experience that hot testing can be performed in this timeframe.

ACTIONS

A.1

With one pressurizer safety valve inoperable, restoration must take place within 15 minutes. The Completion Time of 15 minutes reflects the importance of maintaining the RCS Overpressure Protection System. An inoperable safety valve coincident with an RCS overpressure event could challenge the integrity of the pressure boundary.

B.1 and B.2

If the Required Action of A.1 cannot be met within the required Completion Time or if two pressurizer safety valves are inoperable, the plant must be brought to a MODE in which the requirement does not apply. To achieve this status, the plant must be brought to at least MODE 3 within 6 hours and to MODE 4 with any RCS cold leg temperatures at or below the LTOP enabling temperature specified in the PTLR within 12 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems. With any RCS cold leg temperatures at or below the LTOP enabling temperature specified in the PTLR,

B PTLR

BASES

ACTIONS (continued)	overpressure protection is provided by the LTOP System. The change from MODE 1, 2, or 3 to MODE 4 reduces the RCS energy (core power and pressure), lowers the potential for large pressurizer insurges, and thereby removes the need for overpressure protection by two pressurizer safety valves.
SURVEILLANCE REQUIREMENTS	<u>SR 3.4.10.1</u> SRs are specified in the Inservice Testing Program. Pressurizer safety valves are to be tested in accordance with the requirements of Section XI of the ASME Code (Ref. 4), which provides the activities and Frequencies necessary to satisfy the SRs. No additional requirements are specified. The pressurizer safety valve setpoint is ± 3% for OPERABILITY; however, the valves are reset to +2.67%/-1.78% during the Surveillance to allow for drift.
REFERENCES	1. ASME, Boiler and Pressure Vessel Code, Section III. 2. FSAR, Chapter 14. 3. WCAP-7769, Rev. 1, June 1972. 4. ASME, Boiler and Pressure Vessel Code, Section XI.

Description of Changes - NUREG-1431 Section 3.04.11

01-Aug-00

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

CTS:	ITS:
15.03.01.A.05	LCO 3.04.11
15.03.01.A.05.A.01	LCO 3.04.11 COND A LCO 3.04.11 COND A RA A.1
15.03.01.A.05.A.02	LCO 3.04.11 COND B LCO 3.04.11 COND B RA B.1 LCO 3.04.11 COND B RA B.2 LCO 3.04.11 COND B RA B.3 LCO 3.04.11 COND D
15.03.01.A.05.A.03	LCO 3.04.11 COND E LCO 3.04.11 COND E RA E.1 LCO 3.04.11 COND E RA E.2
15.03.01.A.05.A.04	LCO 3.04.11 COND C LCO 3.04.11 COND C RA C.1 LCO 3.04.11 COND C RA C.2 LCO 3.04.11 COND D
15.03.01.A.05.A.05	LCO 3.04.11 COND F LCO 3.04.11 COND F RA F.1 LCO 3.04.11 COND F RA F.2 LCO 3.04.11 COND G
15.04.01 T 15.04.01-02 21 (A)	SR 3.04.11.01
15.04.01 T 15.04.01-02 21 (A)(13)	SR 3.04.11.01 NOTE
15.04.01 T 15.04.01-02 27	SR 3.04.11.02 SR 3.04.11.03

DOC Number	DOC Text
A.02 Rev. A	CTS 15.3.1.A.5 contains a statement that informs the operator that if the unit is placed in HOT SHUTDOWN in accordance with specifications 15.3.1.A.5.a(1) through 15.3.1.A.5.a(5), then the RCS temperature should be maintained > 355 F to avoid entry into the applicability of 15.3.15, Low Temperature Overpressure Protection, unless required to restore the inoperable components. This statement is not being retained in ITS, because it does not provide any requirements.

CTS:	ITS:
15.03.01.A.05	N/A

Description of Changes - NUREG-1431 Section 3.04.11

01-Aug-00

DOC Number	DOC Text
L.01 Rev. B	CTS 15.4.1, Table 15.4.1-1, item 34, requires that PORV automatic actuation at normal operating conditions be verified through performance of a quarterly Channel Functional Test and a Channel Calibration performed each refueling outage. The CFT frequency is modified by Note (11), which specifies "Performance of a channel functional test is required, excluding valve operation." These surveillance requirements are being deleted from the Technical Specifications, because they do not verify a function assumed in accident analyses to mitigate a design basis accident or transient. As such, the surveillances are not required to be in the ITS to provide adequate protection to the public health and safety.

CTS:	ITS:
15.04.01 T 15.04.01-01 34	N/A
15.04.01 T 15.04.01-01 34 (11)	N/A

DOC Number	DOC Text
LA.01 Rev. B	Not Used.

CTS:	ITS:
N/A	N/A

DOC Number	DOC Text
LB.01 Rev. A	The requirement for testing the PORV and PORV block valve position indicators, as specified in CTS 15.4.1, Table 15.4.1-1, Items 33 and 35 have been deleted. The testing of these valves is incorporated in the IST program, which specifies the appropriate testing to be performed. Controls for inservice testing of ASME Code Class 1, 2, and 3 components are provided in proposed Specification 5.5.6, "Inservice Testing Program".

CTS:	ITS:
15.04.01 T 15.04.01-01 33	N/A
15.04.01 T 15.04.01-01 35	N/A
15.04.01 T 15.04.01-01 35 (21)	N/A

DOC Number	DOC Text
M.01 Rev. A	CTS 15.3.1.A.5.a requires two PORVs and their associated block valves to be operable. Proposed ITS LCO 3.4.11 requires each PORV and associated block valve to be operable in MODES 1, 2 and MODE 3 with Tavg greater than or equal to 500 F. CTS 15.3.1.A.5.a does not include an applicability statement, although the actions taken when the LCO is not met, allow unrestricted operation in HOT SHUTDOWN. Proposed ITS LCO 3.4.11 expands the applicability to encompass the conditions of unit operation where the PORVs and associated block valves are required for SGTR event mitigation. Therefore, this change is more restrictive.

CTS:	ITS:
15.03.01.A.05.A	LCO 3.04.11

TABLE 15.4.1-1 (continued)

NO.	CHANNEL DESCRIPTION	CHECK	CALIBRATE	TEST	PLANT CONDITIONS WHEN REQUIRED	
20.	Auxiliary Feedwater Flowrate	(13)	R	-	ALL	< See Section 3.3 >
21.	Boric Acid Control System	-	R	-	ALL	
22.	Boric Acid Tank Level	D	R	-	ALL	< See Section 3.5 >
23.	Charging Flow	-	R	-	ALL	
24.	Condensate Storage Tank Level	S(1)	R	-	ALL	< See Sections 3.3 and 3.7 >
25.	Containment High Range Radiation	M(1)	R(14)	-	ALL	
26.	Containment Hydrogen Monitor	D	-	-	ALL	
	-Gas Calibration	-	Q(15)	-	ALL	
	-Electronic Calibration	-	R	-	ALL	
27.	Containment Pressure	S	R	Q(1,3,9)	ALL	
28.	Containment Water Level	M	R	-	ALL	< See Section 3.3 >
29.	Emergency Plan Radiation Survey Instruments	Q	R	Q	ALL	
30.	Deleted					
31.	In-Core Thermocouples	M	R(14)	-	ALL	< See Section 3.3 >
32.	Low Temperature Overpressure Protection System	S(12)	R	(10)	ALL	< See LCO 3.4.12 >
33.	PORV Block Valve Position Indicator	Q	R	-	ALL	LB.1
34.	PORV Operability	-	R	Q(11)	ALL	L.1
35.	PORV Position Indicator	S(21)	R	R	ALL	LB.1

B

NOTES USED IN TABLE 15.4.1-1 (continued)

< See LCO 3.4.12 >

(10) When used for the Low Temperature Overpressure Protection System, each PORV shall be demonstrated operable by:
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

L.1

B

(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 .

(15) Sample gas for calibration at 2% and 6%.

< See Section 3.3 >

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

< See LCO 3.4.3 >

(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 3.1.5 >

(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.

(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.B must be met.

< See Section 3.3 >

(21) Check required only when the low temperature overpressure protection system is in operation .

LB.1

(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.

< See 3.1.5, 3.1.6, 3.1.7, 3.3.1 >

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

(24) Neutron detectors excluded from calibration.

< See Section 3.3 >

NSHC Number	NSHC Text
L.01 Rev. B	In accordance with the criteria set forth in 10 CFR 50.92, PBNP has evaluated this proposed Technical Specifications change and determined it does not represent a significant hazards consideration. The following is provided in support of this conclusion. 1. Does the change involve a significant increase in the probability or consequences of an accident previously evaluated? The proposed change does not involve any physical alteration of plant systems, structures or components, changes in parameters governing normal plant operation, or methods of operation. The proposed change deletes surveillances requirements of the PORV automatic actuation function. These surveillance requirements are being deleted from the Technical Specifications, because they do not verify a function assumed in accident analyses to mitigate a design basis accident or transient. As such, the surveillances are not required to be in the ITS to provide adequate protection to the public health and safety. Therefore, this change does not involve a significant increase in the probability or consequences of an accident previously evaluated. 2. Does the change create the possibility of a new or different kind of accident from any accident previously evaluated? The proposed change does not 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 surveillance requirements being deleted from the Technical Specifications do not verify a function assumed in accident analyses to mitigate a design basis accident or transient. Therefore, this change does not involve a significant reduction in a margin of safety.

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
F. Two block valves inoperable.	------------------NOTE------------------ Required Action F.1 does not apply when block valve is inoperable solely as a result of complying with Required Actions B.2 or E.2 --	
	F.1 Restore one block valve to OPERABLE status.	2 hours
G. Required Action and associated Completion Time of Condition F not met.	G.1 Be in MODE 3.	6 hours
	AND	
	G.2 Reduce T_{avg} to < 500°F.	12 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.4.11.1	---------------------------NOTE--------------------------- Not required to be met with block valve closed in accordance with the Required Action of Condition B or E. ---	
	Perform a complete cycle of each block valve.	92 days
SR 3.4.11.2	Perform a complete cycle of each PORV.	18 months

Description of Changes - NUREG-1431 Section 3.04.12

01-Aug-00

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

CTS:	ITS:
15.03.15	N/A
15.03.15.A	N/A
15.03.15.A.02	N/A
15.03.15.A.02.A	LCO 3.04.12 COND D RA D.1 LCO 3.04.12 COND F LCO 3.04.12 COND F RA F.1
15.03.15.A.02.B	LCO 3.04.12 COND E RA E.1 LCO 3.04.12 COND F LCO 3.04.12 COND F RA F.1
15.03.15.A.03	SR 3.04.12.03
15.03.15.B.01	LCO 3.04.12.A
15.04.01 T 15.04.01-01 32	SR 3.04.12.06
15.04.01 T 15.04.01-02 21 (B)	SR 3.04.12.04
15.04.01 T 15.04.01-02 21 (B)(14)	SR 3.04.12.04

DOC Number	DOC Text
A.02 Rev. A	CTS 15.3.15.A.1.b requires both PORV block valves be open for PORV operability. This requirement is retained in ITS in proposed SR 3.4.12.6. This surveillance requires the PORV block valve for each required PORV be verified open at a frequency of 72 hours. This verification ensures the flowpath for each required PORV is established and maintained for the conditions that the PORVs are required to be operable.

CTS:	ITS:
15.03.15.A.01.B	SR 3.04.12.04

DOC Number	DOC Text
A.03 Rev. B	Not Used.

CTS:	ITS:
N/A	N/A

Description of Changes - NUREG-1431 Section 3.04.12

01-Aug-00

DOC Number	DOC Text	
A.04 Rev. A	CTS 15.3.15.A.2.b provides actions for conditions where one PORV is inoperable while reactor coolant temperature is less than or equal to 200 F. Proposed ITS LCO 3.4.12, Condition E, provides Required Actions in the event one PORV is inoperable in MODES 5 or 6. LCO 3.4.12 applicability in MODE 6 is when the reactor vessel head in on. ITS MODE 5 has a temperature requirement of less than or equal to 200 F. Therefore, the actions of CTS 15.3.15.A.2.b and ITS LCO 3.4.12, Condition E, are required to be performed under the same set of plant conditions. This change is being made to adopt the terms and conventions utilized in NUREG-1431, and is, therefore, administrative in nature.	
	CTS:	**ITS:**
	15.03.15.A.02.B	LCO 3.04.12 COND E
	BASES	B 3.04.13
A.05 Rev. B	Not Used.	
	CTS:	**ITS:**
	N/A	N/A
A.06 Rev. B	CTS 15.3.15.A.3 requires verification of the RCS vent pathway when required per Specification 15.3.1.A.2.a, b, or c. The verification is required every 31 days when it is provided by a non-isolable pathway or by a valve(s) that is locked, sealed, or otherwise secured in the open position. The verification is required every 12 hours when it is provided by other means. Proposed ITS SR 3.4.12.3 requires verification of the RCS vent at a frequency of 31 days for non-isolable pathways and locked open vent valve(s), and at a frequency of 12 hours for unlocked open vent valve(s). The Bases for ITS SR 3.4.12.3 states the 31 day frequency is for a non-isolable pathway or a valve that is locked, sealed, or secured in position. The Bases also state a removed pressurizer safety valve fits this category. Therefore, proposed ITS SR 3.4.12.3 provides the same requirements as CTS 15.3.15.A.4, and this change is administrative.	
	CTS:	**ITS:**
	15.03.15.A.03	SR 3.04.12.03
A.07 Rev. A	The Bases of the current Technical Specifications for this section have been completely replaced by revised Bases that reflect the format and applicable content of PBNP ITS Chapter 3.4, consistent with the Standard Technical Specifications for Westinghouse Plants, NUREG-1431. The revised Bases are as shown in the PBNP ITS Bases.	
	CTS:	**ITS:**
	BASES	B 3.04.12

Description of Changes - NUREG-1431 Section 3.04.12

01-Aug-00

DOC Number	DOC Text
A.08 Rev. B	CTS 15.4.1, Table 15.4.1-1, Item 32, requires a Calibration and Test of the Low Temperature Overpressure Protection channels in ALL plant conditions. Implicit in this statement is the requirement to perform these surveillances when the associated LCO (15.3.15.A.1) is applicable. Proposed ITS SR 3.4.12.5 and SR 3.4.12.6 require the performance of a COT and a CHANNEL CALIBRATION on the Low Temperature Overpressure Protection channels. Per proposed SR 3.0.1, SRs shall be met during the MODES (or other specified conditions of Applicability) for individual LCOs, unless otherwise stated in the SR. Therefore, SR 3.4.12.5 and SR 3.4.12.6 are required to be performed when the associated LCO (3.4.12) is applicable. The performance of the CTS and the ITS surveillance requirements for Low Temperature Overpressure Protection instruments is dictated by the applicability of the respective LCO. This change is administrative.

CTS:	ITS:
15.04.01 T 15.04.01-01 32	SR 3.04.12.05
	SR 3.04.12.06

DOC Number	DOC Text
L.01 Rev. B	CTS 15.4.1, Table 15.4.1-1, item 32 requires a shiftly CHECK be performed on the LTOP System, 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. The provisional statement regarding the RCS being open to the atmosphere is not being retained in the ITS. This restriction is being changed to require pressure relief capabilities consistent with assumptions of the analysis. Proposed ITS SR 3.4.12.4 requires a 72 hour verification that the required trains of LTOP are enabled. Verifying the LTOP enabled lights are illuminated, verifies the PORV block valves are open and the LTOP enabling switches are in the correct position. This verification meets the same requirements as performing a CHECK of the LTOP System under CTS 15.4.1, Table 15.4.1-1, item 32. Reducing the frequency requirement from shiftly to 72 hours is less restrictive, but is adequate, considering the LTOP enabling indications are readily available to the operators in the control room, and any change in the LTOP enabling status would be easily identified.

CTS:	ITS:
15.04.01 T 15.04.01-01 32	SR 3.04.12.04
15.04.01 T 15.04.01-01 32 (12)	N/A

Description of Changes - NUREG-1431 Section 3.04.12

01-Aug-00

DOC Number	DOC Text
LA.01 Rev. A	The value of the LTOP enabling temperature and the pressurizer power operated relief valve setpoints are removed from the Specifications and placed in the Pressure Temperature Limits Report (PTLR). This information provides details of design or process that are not directly pertinent to the actual requirement, i.e., Limiting Condition for Operation or Surveillance Requirement, but rather describe frequently changing parameters of the specification. This detail is not necessary to adequately describe the actual regulatory requirement, and can be moved to licensee controlled documents without a significant impact on safety. Administrative controls are included in Section 5 of the proposed ITS to control revisions to these values.

CTS:	ITS:
15.03.15.A.01	LCO 3.04.12
15.03.15.A.01.A	LCO 3.04.12.C.1
15.03.15.A.02.A	LCO 3.04.12 COND D
15.03.15.A.02.C	LCO 3.04.12 COND F RA F.1

DOC Number	DOC Text
LA.02 Rev. A	CTS 15.3.15.B.1 provides information on the methods of verifying a maximum of one safety injection pump capable of injecting into the RCS. These details have been moved to the Bases. This information provides details which are not directly pertinent to the actual requirement, i.e., Limiting Condition of Operation or Surveillance Requirement, but rather describe an acceptable method of compliance. Since these details are not necessary to adequately describe the actual regulatory requirement, they can be moved to other documents without impact on safety. The Bases will be controlled by the Bases Control Process in Section 5 of the proposed ITS.

CTS:	ITS:
15.03.15.B.01	N/A

DOC Number	DOC Text
LB.01 Rev. A	CTS 15.4.1, Table 15.4.1-2, Item 27 requires the operation of the PORVs, PORV Solenoid Air Control Valves, and Air System Check each shutdown. This requirement is modified by Note 16, which states the test valve operation shall be in accordance with the inservice test requirements of the ASME Boiler and Pressure Vessel Code, Section XI. These details are not required to be in the ITS to provide adequate protection of public health and safety. This information is duplicated in 10CFR 50.55a; therefore, the requirements will continue to be applicable to Point Beach, and this change is an administrative relocation of information.

CTS:	ITS:
15.04.01 T 15.04.01-02 27 (16)	N/A

DOC Number	DOC Text
M.01 Rev. B	CTS 15.3.15.A.1 requires that the LTOP system be operable whenever the reactor coolant system is not open to the atmosphere and the temperature is less than the LTOP enable temperature. The provisional statement regarding the RCS being open to the atmosphere is not being retained in the ITS. This restriction is being changed to require pressure relief capabilities consistent with assumptions of the analysis. Therefore, this change is more restrictive.

CTS:	ITS:
15.03.15.A.01	LCO 3.04.12.C.2

Description of Changes - NUREG-1431 Section 3.04.12

01-Aug-00

DOC Number	DOC Text
M.02 Rev. B	CTS 15.3.15 is revised to adopt ITS LCO 3.4.12.b, LCO 3.4.12 Conditions B and C, and SR 3.4.12.2.

LCO 3.4.12.b requires the accumulators to be isolated when accumulator pressure is greater than or equal to the maximum RCS pressure for the existing RCS cold leg temperature allowed by the P/T limit curves provided in the PTLR. These restrictions are necessary to limit the coolant input capability consistent with assumptions of the analysis.

When an accumulator is not isolated, Required Action B.1 dictates the accumulator be isolated within one hour. This is only required when the accumulator pressure is more than the maximum RCS pressure for the existing temperature allowed by the P/T limit curves. If the Required Action and associated Completion Time of Condition B are not met, Required Action C.1 or C.2 must be performed in the next 12 hours. By increasing the RCS temperature to greater than the LTOP enabling temperature specified in the PTLR an accumulator pressure of 800 psig cannot exceed the LTOP limits, if the accumulators are fully injected. Depressurizing the accumulators to less than the maximum RCS pressure for the existing RCS cold leg temperature allowed in the LTOP, also gives this protection. The Completion Times are based on operating experience that these activities can be accomplished in these time periods and on engineering evaluations indicating that an event requiring LTOP is not likely in the allowed times.

To minimize the potential for a low temperature overpressure event by limiting the mass input capability, SR 3.4.12.2 requires the accumulator discharge isolation valves to be verified closed and locked out, when accumulator pressure is greater than or equal to the maximum RCS pressure for existing cold leg temperature allowed in the PTLR. The frequency of 12 hours is sufficient, considering other indications and alarms available to the operators in the control room, to verify the required status of the equipment.

CTS:	ITS:
NEW	LCO 3.04.12 COND B
	LCO 3.04.12 COND B RA B.1
	LCO 3.04.12 COND C
	LCO 3.04.12 COND C RA C.1
	LCO 3.04.12 COND C RA C.2
	LCO 3.04.12.B
	SR 3.04.12.02

Description of Changes - NUREG-1431 Section 3.04.12

01-Aug-00

DOC Number	DOC Text	
M.03 Rev. B	CTS 15.3.15.A.1 states the LTOP system is not required to be operable whenever the RCS is open to the atmosphere. Although "RCS is open to the atmosphere" is not defined, CTS Bases do define the RCS as "vented", if there is an opening in the RCS pressure boundary to atmosphere or the pressurizer relief tank that has an equivalent system pressure relieving capability as a PORV. "Venting the RCS" is an action specified in CTS 15.3.15.A, to be taken when the requirements of the LCO cannot be met. This results in placing the plant in a condition whereby the requirements of LCO 15.3.15 are not required. Proposed ITS LCO 3.4.12.c.2 allows RCS depressurization with a RCS vent path with a venting capability equivalent to or greater than a PORV, as an alternative RCS relief path to the PORVs. The vent path must be capable of relieving the flow resulting from the limiting LTOP mass or heat input transient, and maintaining pressure below the P/T limits. The required vent capacity may be provided by one or more vent paths. For an RCS vent to meet the flow capacity requirement, it requires removing a pressurizer safety valve, removing a PORV's internals, or similarly establishing a vent by opening an RCS vent valve or non-isolable pathway. The allowance for LTOP to be considered operable under depressurized and vented conditions (with the reactor vessel head on), per proposed ITS 3.4.12.c.2, places additional requirements on plant operation, and, therefore, is more restrictive.	
	CTS:	**ITS:**
	NEW	LCO 3.04.12.C.2
M.04 Rev. A	CTS 15.3.15.B.1 requires the second high pressure safety injection pump to be verified inoperable whenever LTOP is required to be enabled. Proposed ITS SR 3.4.12.1 requires verification that a maximum of one SI pump is capable of injecting into the RCS every 12 hours. The frequency of 12 hours is sufficient, considering other indications and alarms available to the operator in the control room, to verify the required status of the equipment. Requiring periodic verification that only one SI pump is capable of injecting into the RCS places additional requirements on plant operation is, therefore, more restrictive.	
	CTS:	**ITS:**
	15.03.15.B.01	SR 3.04.12.01
M.05 Rev. A	CTS 15.3.15 is revised to adopt ITS LCO 3.4.12, Condition A, to provide Required Actions in the event more than one SI pump is capable of injecting into the RCS. To immediately initiate action to restore restricted coolant input capability to the RCS reflects the urgency of removing the RCS from this condition. This change imposes additional requirements on plant operation and is therefore more restrictive.	
	CTS:	**ITS:**
	NEW	LCO 3.04.12 COND A LCO 3.04.12 COND A RA A.1

DOC Number	DOC Text
M.06 Rev. B	CTS 15.4.1, Table 15.4.1-1, Item 32, requires a Channel Functional Test (CFT) on the PORV actuation channel, excluding valve operation, within 31 days prior to entering a condition in which the PORV is required to be operable and at least 31 days thereafter when the PORV is required to be operable. Proposed ITS SR 3.4.12.5 requires the performance of a COT on each required PORV, excluding actuation, at a frequency of 31 days. However, the requirements of SR 3.0.4 are only applicable for entry into a MODE (or other specified condition of Applicability) in MODES 1, 2, 3 and 4. Therefore, during plant operation in MODE 6, when the reactor vessel head is being re-installed, SR 3.4.12.5 would not be required to be performed prior to entering the Applicability of LCO 3.4.12 (MODE 6 when the reactor vessel head is on). Therefore, a NOTE has been added to the ACTIONS to preclude an entry into the Applicability of LCO 3.4.12 without the requirements of the LCO being met. Preventing entry into MODE 6 with the reactor vessel head on, from MODE 6 with the reactor vessel head removed, if the requirements of LCO not met, places additional requirements on plant operation and is, therefore, more restrictive. **CTS:** 15.04.01 T 15.04.01-01 32 (10) **ITS:** LCO 3.04.12 ACTIONS NOTE SR 3.04.12.05
M.07 Rev. A	CTS 15.4.1, Table 15.4.1-2, Item 27 requires the operation of the PORVs, PORV Solenoid Air Control Valves, and Air System Check each shutdown. Proposed ITS SR 3.4.12.8 requires a complete cycle of each required PORV, and SR 3.4.12.7 require a complete cycle of each solenoid air control valve and check valve on the air accumulators in the PORV control systems. Both of these surveillances are required at a frequency of 18 months. The CTS requirement is the same as the proposed ITS with the exception of the specified frequency. The CTS does not define a specific frequency of performance for these Surveillance, but rather an evolution, which can vary significantly from shutdown to shutdown with no bounding limit. Accordingly, the adoption of a bounding frequency (18 months) is a more restrictive change. **CTS:** 15.04.01 T 15.04.01-02 27 **ITS:** SR 3.04.12.07 SR 3.04.12.08

15.3.15 LOW TEMPERATURE OVERPRESSURE PROTECTION SYSTEM

Applicability

Applies to operability of the low temperature overpressure protection (LTOP) system when the reactor coolant system temperature is < 355°F.

Objective

To specify functional requirements and limiting conditions for operation on the use of the pressurizer power operated relief valves when used as part of the LTOP system and to specify further limiting conditions for operation when the reactor coolant system is operated at low temperatures.

Specification

A. System Operability

M.1 LA.1

1. Except as specified in 15.3.15.A.2 below, the LTOP system shall be operable whenever the reactor coolant system is not open to the atmosphere and the temperature is < 355°F. Operability requirements are:

with lift settings within the limits specified in the PTLR.

LA.1

a. Both pressurizer power operated relief valves operable at a setpoint of ≤440 psig.

or RCS depressurized and an RCS vent path with venting capability equivalent to or greater than a PORV. M.3

b. Both power operated relief valve block valves are open. A.2

2. The requirements of 15.3.15.A.1 may be modified as specified below :

a. With one PORV inoperable while reactor coolant system temperature is >200°F but < 355°F, either restore the inoperable PORV to operable status within 7 days, or depressurize and vent reactor coolant system within the next 8 hours.

LTOP enable temperature specified in the PTLR.

LA.1

B PTLR

b. With one PORV inoperable while reactor coolant system temperature is ≤200°F, either restore the inoperable PORV to operable status within 24 hours, or depressurize and vent the reactor coolant system within a total of 32 hours.

A.4 A.4

c. Each accumulator isolated whose pressure is greater than or equal to the maximum RCS pressure for the existing RCS cold leg temperature allowed by the P/T limit curves in the PTLR.

B PTLR

M.2

LTOP enable temperature specified in the PTLR.

c. With both power operated relief valves inoperable while the reactor coolant system temperature is ≤ 355°F, the reactor coolant system must be depressurized and vented within 8 hours .

B
PTLR

LA.1

3. If the reactor coolant system is vented per Specification 15.3.15.A.2.a, b, or c, the pathway must be verified at least once every 31 days when it is provided by a non-isolable pathway or by a valve(s) that is locked, sealed, or otherwise secured in the open position; otherwise, verify the pathway every 12 hours. A.6

B. Additional Limitations

1. When LTOP is required to be enabled by Specification 15.3.15.A.1, no more than one high pressure safety injection pump shall be operable. The second high pressure safety injection pump shall be rendered inoperable whenever LTOP is required to be enabled by verifying that the motor circuit breakers have been removed from their electrical power supply circuits or by verifying that the discharge valves from the high pressure safety injection pumps to the reactor coolant system are shut and that power is removed from their operators.

M.4

LA.2

2. A reactor coolant pump shall not be started when the reactor coolant system temperature is < 355°F unless:

a. There is a pressure absorbing volume in the pressurizer or in the steam generator tubes or

b. The secondary water temperature of each steam generator is less than 50°F above the temperature of the reactor coolant system.

< See LCO 3.4.6 and LCO 3.4.7 >

Basis

The Low Temperature Overpressure Protection System consists of a redundant means of relieving pressure during periods of water solid operation and when the reactor coolant system temperature is < 355°F. This method of water

A.7

Add Condition A.
See Insert 3.4.12-1.

M.5

Add Conditions B and C.
See Insert 3.4.12-1.

M.2

Add SR 3.4.12.2.
See Insert 3.4.12-2.

M.2

B
PTLR

Section 3.4.12 CTS Markup Inserts

Insert 3.4.12-1:

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Two SI pumps capable of injecting into the RCS.	A.1 Initiate action to verify a maximum of one SI pump is capable of injecting into the RCS.	Immediately
B. An accumulator not isolated when the accumulator pressure is greater than or equal to the maximum RCS pressure for existing cold leg temperature allowed in the PTLR.	B.1 Isolate affected accumulator.	1 hour
C. Required Action and associated Completion Time of Condition B not met.	C.1 Increase RCS cold leg temperature to > LTOP enabling temperature specified in the PTLR. OR C.2 Depressurize affected accumulator to less than the maximum RCS pressure for existing cold leg temperature allowed in the PTLR.	12 hours 12 hours

B PTLR

B RAI 3.4.12-1 PTLR

Insert 3.4.12-2:

SURVEILLANCE		FREQUENCY
SR 3.4.12.2	-------------------NOTE------------------ Only required when accumulator pressure is ≥ the maximum RCS pressure for existing cold leg temperature allowed by the P/T limit curves provided in the PTLR. --- Verify each accumulator is isolated.	12 hours

B

RAI 3.4.12-2

PTLR

Justification For Deviations - NUREG-1431 Section 3.04.12

01-Aug-00

JFD Number	JFD Text	
01 Rev. B	Not Used.	
	ITS:	**NUREG:**
	N/A	N/A
02 Rev. A	With the deletion of non-plant specific information from the NUREG, LCO 3.4.12 is arranged into a format which more clearly delineates the requirements for LTOP. This format is consistent with TSTF-280, Rev. 1.	
	ITS:	**NUREG:**
	B 3.04.12	B 3.04.12
	LCO 3.04.12.A	LCO 3.04.12
	LCO 3.04.12.B	LCO 3.04.12
	LCO 3.04.12.C.1	LCO 3.04.12 A.1
	LCO 3.04.12.C.2	LCO 3.04.12 B

Justification For Deviations - NUREG-1431 Section 3.04.12

01-Aug-00

JFD Number	JFD Text
03 Rev. B	The brackets have been removed and the proper plant specific information has been provided. In some instances, even though the information was designated as plant specific information in the LCO (bracketed), the corresponding Bases information was not bracketed. These cases are self evident, corresponding to the bracketed information in the LCO, and have had the appropriate site specific information provided. NUREG-1431 SR 3.4.12.2, SR 3.4.12.4, SR 3.4.12.7 and Condition B were not used for Point Beach, leading to the renumbering of subsequent surveillance requirements and Actions. Additionally, TSTF-285, Rev. 1 removed and modified the note in LCO 3.4.12 Condition B and reinserted it into a note in the LCO. This note and TSTF were not adopted in the Point Beach conversion because it was unnecessary due to Point Beach design.

ITS:	NUREG:
B 3.04.12	B 3.04.12
LCO 3.04.12 COND A	LCO 3.04.12 COND A
LCO 3.04.12 COND A RA A.1	LCO 3.04.12 COND A RA A.1
LCO 3.04.12 COND B	LCO 3.04.12 COND C
LCO 3.04.12 COND B RA B.1	LCO 3.04.12 COND C RA C.1
LCO 3.04.12 COND C	LCO 3.04.12 COND D
LCO 3.04.12 COND C RA C.1	LCO 3.04.12 COND D RA D.1
LCO 3.04.12 COND C RA C.2	LCO 3.04.12 COND D RA D.2
LCO 3.04.12 COND D	LCO 3.04.12 COND E
LCO 3.04.12 COND D RA D.1	LCO 3.04.12 COND E RA E.1
LCO 3.04.12 COND E	LCO 3.04.12 COND F
LCO 3.04.12 COND E RA E.1	LCO 3.04.12 COND F RA F.1
LCO 3.04.12 COND F	LCO 3.04.12 COND G
LCO 3.04.12 COND F RA F.1	LCO 3.04.12 COND G RA G.1
N/A	LCO 3.04.12 A.2 LCO 3.04.12 A.3 LCO 3.04.12 COND B LCO 3.04.12 COND B RA B.1 LCO 3.04.12 COND B RA B.1 NOTE SR 3.04.12.02 SR 3.04.12.04 SR 3.04.12.07
SR 3.04.12.01	SR 3.04.12.01

JFD Number	JFD Text	
	SR 3.04.12.02	SR 3.04.12.03
	SR 3.04.12.03	SR 3.04.12.05
	SR 3.04.12.04	SR 3.04.12.06
	SR 3.04.12.05	SR 3.04.12.08
	SR 3.04.12.06	SR 3.04.12.09
04 Rev. B	Not Used.	
	ITS:	**NUREG:**
	N/A	N/A
05 Rev. A	The Applicability of NUREG-1431, LCO 3.4.12, is corrected to state, "MODE 4 when any RCS cold leg temperature is ...", instead of "MODE 4 when all RCS cold leg temperature is ..." The Bases state that the Applicability is "MODE 4 when any cold leg temperature is ..." This change is consistent with the Pressurizer Safety Valve requirements of LCO 3.4.10, which is applicable in MODE 4 with all RCS cold leg temperatures > [275 F]. This change is made in accordance with TSTF-243, Rev. 0.	
	ITS:	**NUREG:**
	LCO 3.04.12	LCO 3.04.12
06 Rev. A	The actual numerical values for LTOP enabling temperature are replaced with a reference to the temperature specified in the PTLR. The LTOP enabling temperature will then be calculated and controlled by the licensee in accordance with the topical reports identified in the PTLR.	
	ITS:	**NUREG:**
	B 3.04.12	B 3.04.12
	LCO 3.04.12	LCO 3.04.12
07 Rev. A	NUREG LCO 3.4.12 Applicability NOTE is moved to ITS LCO 3.4.12.c. The Note modifies the LCO statement by requiring accumulator isolation only when accumulator pressure is greater than or equal to the maximum RCS pressure for the existing RCS cold leg temperature allowed by the P/T limit curves provided in the PTLR. The Note does not provide modifying information for the Applicability of the LCO. TSTF-285, Rev. 1 also moved this note to the LCO.	
	ITS:	**NUREG:**
	B 3.04.12	B 3.04.12
	LCO 3.04.12.B	LCO 3.04.12 APPL NOTE

Justification For Deviations - NUREG-1431 Section 3.04.12

01-Aug-00

JFD Number	JFD Text	
08 Rev. A	NUREG LCO 3.4.12 ACTIONS have been modified with the addition of a Note stating that while the LCO is not met, entry into MODE 6, with the reactor vessel head on, from MODE 6, with the reactor vessel head removed, is not permitted. This Note prevents entry into the MODES of applicability for LTOP without the requirements of LCO 3.4.12 being met. This Note is necessary, because LCO 3.0.4 is only applicable for entry into a MODE or other specified condition in the Applicability in MODES 1, 2, 3 and 4.	
	ITS:	**NUREG:**
	B 3.04.12	B 3.04.12
	LCO 3.04.12 ACTIONS NOTE	N/A
09 Rev. B	Not Used.	
	ITS:	**NUREG:**
	B 3.04.12	B 3.04.12
	N/A	N/A N/A
10 Rev. B	Not used.	
	ITS:	**NUREG:**
	N/A	N/A
11 Rev. A	NUREG-1431, LCO 3.4.12, Conditions E and F references to "RCS relief valve(s)," has been modified to "PORV(s)". NUREG-1431, LCO 3.4.12 was written for plants which may utilize RHR suction relief valves to meet LTOP requirements. Point Beach current licensing basis does not credit RHR suction relief valves in the mitigation of low temperature overpressure events. Therefore, to clarify Conditions E and F, the RCS relief valve(s) will be referred to as PORV(s).	
	ITS:	**NUREG:**
	B 3.04.12	B 3.04.12
	LCO 3.04.12 COND D	LCO 3.04.12 COND E
	LCO 3.04.12 COND D RA D.1	LCO 3.04.12 COND E RA E.1
	LCO 3.04.12 COND E	LCO 3.04.12 COND F
	LCO 3.04.12 COND E RA E.1	LCO 3.04.12 COND F RA F.1

Justification For Deviations - NUREG-1431 Section 3.04.12

01-Aug-00

JFD Number	JFD Text	
12 Rev. B	NUREG-1431, SR 3.4.12.3 is being modified by a Note to only require verification that each accumulator is isolated when accumulator pressure is greater than or equal to the maximum RCS pressure for existing cold leg temperatures allowed in the PTLR. This change allows the performance of SR 3.4.12.3 to be consistent with the requirements of isolating the accumulators per LCO 3.4.12 and the required actions of Condition C.	
	ITS:	**NUREG:**
	B 3.04.12	B 3.04.12
	SR 3.04.12.02	SR 3.04.12.03
13 Rev. A	The NUREG-1431 LCO 3.4.12 Bases Background discussion of the RCS Vent Requirements was replaced with a discussion of the applicable Point Beach Licensing Bases attributes contained in the Point Beach CTS Bases discussion of LTOP.	
	ITS:	**NUREG:**
	B 3.04.12	B 3.04.12
14 Rev. A	NUREG-1431, SR 3.4.12.5 is modified to reflect a frequency for verifying other vent path(s) utilized to meet the LTOP requirement. Point Beach will retain the option of meeting LTOP requirements by depressurizing the RCS and providing an RCS vent path equivalent to the relief capacity of the PORVs. Included as a viable RCS vent pathway at Point Beach are the SG and Pressurizer manways. The frequency of 31 days for verification of these vents is consistent with the frequency of verifying a removed pressurizer safety valve. This change is in accordance with TSTF-271, Rev. 1. Additional Bases changes to SR 3.4.12.5 were also made in accordance with TSTF-271, Rev.1.	
	ITS:	**NUREG:**
	B 3.04.12	B 3.04.12
	SR 3.04.12.03	SR 3.04.12.05
15 Rev. B	NUREG-1431, SR 3.4.12.6 is modified from, "Verify PORV block valve open for each required PORV", to "Verify required trains of LTOP enabled." This is more consistent with the CTS 15.4.1, Table 15.4.1-1, item 32, requirement to perform a CHECK of the LTOP System. Verifying both LTOP trains are armed, not only verifies the PORV block valves are open, but also verifies the LTOP enabling switches are in the correct position. Only verifying the PORV block valves are open, doesn't ensure the LTOP System is available to protect the RCPB.	
	ITS:	**NUREG:**
	B 3.04.12	B 3.04.12
	SR 3.04.12.04	SR 3.04.12.06

Justification For Deviations - NUREG-1431 Section 3.04.12

01-Aug-00

JFD Number	JFD Text	
16 Rev. A	The NUREG Note modifying NUREG SR 3.4.12.8 is deleted. Performance of a COT on the LTOP instrumentation does not require the plant to be operating in the LTOP MODES. Therefore this exemption from the requirements of LCO 3.0.4 is unwarranted. Therefore, incorporation of TSTF-233, Rev. 0 change to this note was also not necessary.	
	ITS:	**NUREG:**
	B 3.04.12	B 3.04.12
	N/A	SR 3.04.12.08 NOTE
17 Rev. A	NUREG LCO 3.4.12 is modified by the addition of SR 3.4.12.9 and SR 3.4.12.10. These surveillances provide for the operation of the PORVs, the solenoid air control valves and the check valves on the air accumulator to ensure the PORVs and PORV control systems will actuate properly when called upon. The surveillances are consistent with the requirements of LCO 3.4.11, which also requires these components be periodically operated to ensure their operability.	
	ITS:	**NUREG:**
	B 3.04.12	B 3.04.12
	SR 3.04.12.07	N/A
	SR 3.04.12.08	N/A
18 Rev. A	Plant specific information regarding LTOP configuration has been provided to replace generic LTOP configuration information contained in the Background section of the Bases. This information has been replaced in order to provide a more accurate description of LTOP operation at Point Beach.	
	ITS:	**NUREG:**
	B 3.04.12	B 3.04.12

Justification For Deviations - NUREG-1431 Section 3.04.12

01-Aug-00

JFD Number	JFD Text
19 Rev. B	NUREG-1431, LCO 3.4.12, requires either two RCS relief valves or the RCS depressurized and a RCS vent of x square inches, to meet LTOP requirements. Point Beach does not currently have a calculation that establishes the number of square inches necessary to provide an adequate vent path to meet the LTOP requirements. However, Technical Specifications 15.3.15 Bases states that the reactor coolant system is vented if there is an opening in the reactor coolant system pressure boundary to atmosphere or the PRT that has an equivalent system relieving capability as a PORV. Some examples of such openings include an open or removed PORV, open steam generator or pressurizer manways, a removed pressurizer safety valve, and the top of the reactor vessel when the reactor vessel head has been unbolted or removed. Therefore NUREG-1431, LCO 3.4.12 and SR 3.4.12.5 have been revised to require a RCS vent path with venting capability equivalent or greater than a PORV. Furthermore, the associated Bases includes examples of RCS openings that meet the LTOP requirement. TSTF 280, Rev. 1 deleted the note in SR 3.4.12.5 and added the word "required" before RCS vent. Point Beach staff determined this note was useful, and therefore did not adopt this part of TSTF 280, Rev. 1.

ITS:	NUREG:
B 3.04.12	B 3.04.12
LCO 3.04.12 COND F RA F.1	LCO 3.04.12 COND G RA G.1
LCO 3.04.12.C.2	LCO 3.04.12 B
SR 3.04.12.03	SR 3.04.12.05

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.12 Low Temperature Overpressure Protection (LTOP) System

B
RAI 3.4.12-3

LCO 3.4.12 An LTOP System shall be OPERABLE with ~~a maximum of [one] [high pressure injection (HPI)] pump [and one charging pump] capable of injecting into the RCS and the accumulators isolated and either a or b below.~~ 2 One of the following pressure relief capabilities:

Insert "A" 2

c a Two RCS relief valves, as follows: 2

1. Two power operated relief valves (PORVs) with lift settings within the limits specified in the PTLR, or

3 ~~[2. Two residual heat removal (RHR) suction relief valves with setpoints ≥ [436.5] psig and ≤ [463.5] psig, or]~~

~~[3. One PORV with a lift setting within the limits specified in the PTLR and one RHR suction relief valve with a setpoint ≥ [436.5] psig and ≤ [463.5] psig].~~

2 2 b The RCS depressurized and an RCS vent ~~of ≥ [2.07] square inches.~~ 19 path with venting capability equivalent to or greater than a PORV. 19

APPLICABILITY: MODE 4 when ~~all~~ any 5 RCS cold leg temperature is ≤ ~~[275]°F~~ 6 LTOP enabling temperature specified in the PTLR,
MODE 5,
MODE 6 when the reactor vessel head is on.

B
PTLR

~~NOTE~~
~~Accumulator isolation is only required when accumulator pressure is greater than or equal to the maximum RCS pressure for the existing RCS cold leg temperature allowed by the P/T limit curves provided in the PTLR.~~

7

8 Insert "B"

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
3 SI A. Two or more [HPI] pumps capable of injecting into the RCS.	A.1 Initiate action to verify a maximum of [one] [HPI] pump is capable of injecting into the RCS. 3 one SI	Immediately
3 ~~B. Two or more charging pumps capable of injecting into the RCS.~~	~~B.1 --------NOTE--------- Two charging pumps may be capable of injecting into the RCS during pump swap operation for ≤ 15 minutes. -------------------~~ ~~Initiate action to verify a maximum of [one] charging pump is capable of injecting into the RCS.~~	~~Immediately~~
C. An accumulator not isolated when the accumulator pressure is greater than or equal to the maximum RCS pressure for existing cold leg temperature allowed in the PTLR. B 3	C.1 Isolate affected accumulator. B 3	1 hour

B
PTLR

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
D. Required Action and associated Completion Time of Condition [C] not met.	D.1 Increase RCS cold leg temperature to > [275]°F. LTOP enabling temperature specified in the PTLR. OR D.2 Depressurize affected accumulator to less than the maximum RCS pressure for existing cold leg temperature allowed in the PTLR.	12 hours 12 hours
E. One required RCS relief valve PORV inoperable in MODE 4.	E.1 Restore required ~~RCS~~ relief valve PORV to OPERABLE status.	7 days
F. One required RCS relief valve PORV inoperable in MODE 5 or 6.	F.1 Restore required ~~RCS~~ relief valve PORV to OPERABLE status.	24 hours

B
RAI 3.4.12-1
PTLR

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
3 F G. Two required RCS relief valves PORVs 11 inoperable. OR Required Action and associated Completion Time of Condition A, 3 ~~[B,]~~ D, E, or F C, D or E 3 not met. OR LTOP System inoperable for any reason other than Condition A, [B,] B, 3 C, D, E, or F 3 or E	3 F G.1 Depressurize RCS and establish RCS vent of ≥ [2.07] square inches. path with venting capability equivalent to or greater than a PORV. 19	8 hours

B
RAI 3.4.12-3
PTLR

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.4.12.1	Verify a maximum of 3 one SI [one] [HPI] pump is capable of injecting into the RCS.	12 hours
3 ~~SR 3.4.12.2~~	~~Verify a maximum of one charging pump is capable of injecting into the RCS.~~	~~12 hours~~
SR 3.4.12.3 3 2	Verify each accumulator is isolated.	12 hours

(continued)

12 ---NOTE---
Only required when accumulator pressure is ≥ the maximum RCS pressure for existing cold leg temperature allowed by the P/T limit curves provided in the PTLR.

B
RAI 3.4.12-2

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE	FREQUENCY
3 ~~SR 3.4.12.4 Verify RHR suction valve is open for each required RHR suction relief valve.~~	~~12 hours~~
SR 3.4.12.~~5~~ 3 3 ------------------NOTE------------------ Only required to be performed when complying with LCO 3.4.12.~~b~~ c.2 2 -- Verify RCS vent ~~≥ [2.07] square inches open.~~ path with venting capability equivalent to or greater than a PORV 19	12 hours for unlocked open vent valve(s) AND 31 days for ~~locked open vent valve(s)~~ other vent path(s) 14 B PTLR
SR 3.4.12.~~6~~ 3 4 Verify ~~PORV block valve is open for each required PORV.~~ required trains of LTOP enabled. 15	72 hours B PTLR
3 ~~SR 3.4.12.7 Verify associated RHR suction isolation valve is locked open with operator power removed for each required RHR suction relief valve.~~	~~31 days~~
SR 3.4.12.~~8~~ 3 5 ~~------------------NOTE------------------~~ ~~Not required to be met until 12 hours after decreasing RCS cold leg temperature to ≤ [275] °F.~~ 16 ~~--~~ Perform a COT on each required PORV, excluding actuation.	31 days B PTLR

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.4.12.9 6 3	Perform CHANNEL CALIBRATION for each required PORV actuation channel.	[18] months 18 3

B PTLR

SR 3.4.12.7	Perform a complete cycle of each required PORV solenoid air control valve and check valve on the nitrogen gas bottles.	18 months
SR 3.4.12.8	Perform a complete cycle of each required PORV.	18 months

B PTLR

B PTLR

17

Insert "A"

a. A maximum of one Safety Injection (SI) pump capable of injecting into the RCS; and

b. Each accumulator isolated, whose pressure is ≥ the maximum RCS pressure for the existing RCS cold leg temperature allowed by the P/T limit curves provided in the PTLR. B PTLR

Insert "B"

------------------------------NOTE------------------------------
While this LCO is not met, entry into MODE 6, with the reactor vessel head on, from MODE 6, with the reactor vessel head removed, is not permitted.

Insert "C"

Not Used.

Insert "D"

Not Used.

B PTLR

B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.12 Low Temperature Overpressure Protection (LTOP) System

BASES

B RAI 3.4.12-3

BACKGROUND

The LTOP System controls RCS pressure at low temperatures so the integrity of the reactor coolant pressure boundary (RCPB) is not compromised by violating the pressure and temperature (P/T) limits of 10 CFR 50, Appendix G (Ref. 1). The reactor vessel is the limiting RCPB component for demonstrating such protection. The PTLR provides the maximum allowable actuation logic setpoints for the power operated relief valves (PORVs) and the maximum RCS pressure for the existing RCS cold leg temperature during cooldown, shutdown, and heatup to meet the Reference 1 requirements during the LTOP MODES.

The reactor vessel material is less tough at low temperatures than at normal operating temperature. As the vessel neutron exposure accumulates, the material toughness decreases and becomes less resistant to pressure stress at low temperatures (Ref. 2). RCS pressure, therefore, is maintained low at low temperatures and is increased only as temperature is increased.

The potential for vessel overpr essurization is most acute when the RCS is water solid, occurring only while shutdown; a pressure fluctuation can occur more quickly than an operator can react to relieve the condition. Exceeding the RCS P/T limits by a significant amount could cause brittle cracking of the reactor vessel. LCO 3.4.3, "RCS Pressure and Temperature (P/T) Limits," requires administrative control of RCS pressure and temperature during heatup and cooldown to prevent exceeding the PTLR limits.

3 one Safety Injection (SI)

This LCO provides RCS overpressure protection by having a minimum coolant input capability and having adequate pressure relief capacity. Limiting coolant input capability requires all but [one] [high pressure injection (HPI)] pump 3 ~~[and one charging pump]~~ incapable of injection into the RCS and isolating the accumulators. The pressure relief capacity requires either two redundant RCS relief valves or a depressurized RCS and an RCS vent of sufficient size. One 11 PORV RCS relief valve or the open RCS vent is the overpressure protection device that acts to terminate an increasing pressure event.

B PTLR

PORVs 11

BACKGROUND (continued)

With minimum coolant input capability, the ability to provide core coolant addition is restricted. The LCO does not require the makeup control system deactivated or the safety injection (SI) actuation circuits blocked. Due to the lower pressures in the LTOP MODES and the expected core decay heat levels, the makeup system can provide adequate flow via the makeup control valve. If conditions require the use of more than one [HPI or] charging pump for makeup in the event of loss of inventory, then pumps can be made available through manual actions.

3 SI pump

B RAI 3.4.12-3 PTLR

The LTOP System for pressure relief consists of two PORVs with reduced lift settings, ~~or two residual heat removal (RHR) suction relief valves, or one PORV and one RHR suction relief valve,~~ or a depressurized RCS and an RCS vent of sufficient size. Two RCS relief valves are required for redundancy. One RCS relief valve has adequate relieving capability to keep from overpressurization for the required coolant input capability.

3

PORV

11

PORV Requirements

Insert "C"

As designed for the LTOP System, each PORV is signaled to open if the RCS pressure approaches a limit determined by the LTOP actuation logic. The LTOP actuation logic monitors both RCS temperature and RCS pressure and determines when a condition not acceptable in the PTLR limits is approached. The wide range RCS temperature indications are auctioneered to select the lowest temperature signal.

The lowest temperature signal is processed through a function generator that calculates a pressure limit for that temperature. The calculated pressure limit is then compared with the indicated RCS pressure from a wide range pressure channel. If the indicated pressure meets or exceeds the calculated value, a PORV is signaled to open.

18

The PTLR presents the PORV setpoints for LTOP. ~~The setpoints are normally staggered so only one valve opens during a low temperature overpressure transient.~~ Having the setpoints of both valves within the limits in the PTLR ensures that the Reference 1 limits will not be exceeded in any analyzed event.

B
PTLR

APPLICABLE SAFETY ANALYSES

Safety analyses (Ref. 4) demonstrate that the reactor vessel is adequately protected against exceeding the Reference 1 P/T limits. In MODES 1, 2, and 3, and in MODE 4 with RCS ~~cold leg temperature exceeding [275] °F~~ the LTOP enabling temperature specified in the PTLR the pressurizer safety valves will prevent RCS pressure from exceeding the ~~Reference 1 limits. At about [275] °F~~ and below, overpressure prevention falls to two OPERABLE RCS relief PORVs ~~valves~~ or to a depressurized RCS and a sufficient sized RCS vent. Each of these means has a limited overpressure relief capability.

6

11

11

The actual temperature at which the pressure in the P/T limit curve falls below the pressurizer safety valve setpoint increases as the reactor vessel material toughness decreases due to neutron embrittlement. Each time the PTLR curves are revised, the LTOP System must be re -evaluated to ensure its functional requirements can still be met using the ~~RCS relief valve~~ PORV method or the depressurized and vented RCS condition.

11

B
RAI 3.4.12-3

The PTLR contains the acceptance limits that define the LTOP requirements. Any change to the RCS must be evaluated against the Reference 4 analyses to determine the impact of the change on the LTOP acceptance limits.

Transients that are capable of overpressurizing the RCS are categorized as either mass or heat input transients, examples of which follow:

Mass Input Type Transients

a. Inadvertent safety injection; or

b. Charging/letdown flow mismatch.

Heat Input Type Transients

a. Inadvertent actuation of pressurizer heaters;

b. Loss of RHR cooling; or

c. Reactor coolant pump (RCP) startup with temperature asymmetry within the RCS or between the RCS and steam generators.

APPLICABLE SAFETY ANALYSIS (continued)

The following are required during the LTOP MODES to ensure that mass and heat input transients do not occur, which either of the LTOP overpressure protection means cannot handle:

3 → one SI

3

a. Rendering all but [one] [HPI] pump ~~[and one charging pump]~~ incapable of injection;

3

B PTLR

b. Deactivating the accumulator discharge isolation valves in their closed positions; and

c. Disallowing start of an RCP if secondary temperature is more than [50] °F above primary temperature in any one loop. LCO 3.4.6, "RCS Loops - MODE 4," and LCO 3.4.7, "RCS Loops - MODE 5, Loops Filled," provide this protection.

3 → 50

11

3

11 → PORV

3

The Reference 4 analyses demonstrate that either one RCS ~~relief valve~~ or the depressurized RCS and RCS vent can maintain RCS pressure below limits when only one [HPI] pump ~~[and one charging pump are]~~ is ~~[are]~~ actuated. Thus, the LCO allows only [one] [HPI] pump ~~[and one charging pump]~~ OPERABLE during the LTOP MODES. Since neither one ~~RCS~~ relief valve nor the RCS vent can handle the pressure transient need from accumulator injection, when RCS temperature is low, the LCO also requires the accumulators isolation when accumulator pressure is greater than or equal to the maximum RCS pressure for the existing RCS cold leg temperature allowed in the PTLR.

SI

3

3 → one SI

11

11 → PORV

B PTLR

The isolated accumulators must have their discharge valves closed and the valve power supply breakers fixed in their open positions. The analyses show the effect of accumulator discharge is over a narrower RCS temperature range ([175] °F and below) than that of the LCO ([275] °F and below).

3

270

approximately 265

Fracture mechanics analyses established the temperature of LTOP Applicability at [275] °F.

3

The consequences of a small break loss of coolant accident (LOCA) in LTOP MODE 4 conform to 10 CFR 50.46 and 10 CFR 50, Appendix K (Refs. 5 and 6), requirements by having a maximum of [one] [HPI] pump ~~[and one charging pump]~~ OPERABLE and SI actuation enabled.

3 → one SI

3

B PTLR

APPLICABLE SAFETY ANALYSIS (continued)

PORV Performance

The fracture mechanics analyses show that the vessel is protected when the PORVs are set to open at or below the limit shown in the PTLR. The setpoints are derived by analyses that model the performance of the LTOP System, assuming the limiting LTOP transient of [one] [HPI] pump ~~[and one charging pump]~~ injecting into the RCS. These analyses consider pressure overshoot and undershoot beyond the PORV opening and closing, resulting from signal processing and valve stroke times. The PORV setpoints at or below the derived limit ensures the Reference 1 P/T limits will be met.

B PTLR

3

one SI

3

B RAI 3.4.12-3

The PORV setpoints in the PTLR will be up dated when the revised P/T limits conflict with the LTOP analysis limits. The P/T limits are periodically modified as the reactor vessel material toughness decreases due to neutron embrittlement caused by neutron irradiation. Revised limits are determined using neutron fluence projections and the results of examinations of the reactor vessel material irradiation surveillance specimens. The Bases for LCO 3.4.3, "RCS Pressure and Temperature (P/T) Limits," discuss these examinations.

The PORVs are considered active components. Thus, the failure of one PORV is assumed to represent the worst case, single active failure.

3

RHR Suction Relief Valve Performance

The RHR suction relief valves do not have variable pressure and temperature lift setpoints like the PORVs. Analyses must show that one RHR suction relief valve with a setpoint at or between [436.5] psig and [463.5] psig will pass flow greater than that required for the limiting LTOP transient while maintaining RCS pressure less than the P/T limit curve. Assuming all relief flow requirements during the limiting LTOP event, an RHR suction relief valve will maintain RCS pressure to within the valve rated lift setpoint, plus an accumulation £ 10% of the rated lift setpoint.

Although each RHR suction relief valve may itself meet single failure criteria, its inclusion and location within the RHR System does not allow it to meet single failure criteria when spurious RHR suction isolation valve closure

APPLICABLE SAFETY ANALYSIS (continued)

3

~~is postulated. Also, as the RCS P/T limits are decreased to reflect the loss of toughness in the reactor vessel materials due to neutron embrittlement, the RHR suction relief valves must be analyzed to still accommodate the design basis transients for LTOP.~~

~~The RHR suction relief valves are considered active components. Thus, the failure of one valve is assumed to represent the worst case single active failure.~~

19

RCS Vent Performance

path with venting capability equivalent to or greater than a PORV

With the RCS depressurized, analyses show a vent ~~size of~~ 2.07 square inches is capable of mitigating the allowed LTOP overpressure transient. The capacity of a vent this size is greater than the flow of the limiting transient for the LTOP configuration, [one] HPI pump [and one charging pump] OPERABLE, maintaining RCS pressure less than the maximum pressure on the P/T limit curve.

one SI pump

3

B PTLR

The RCS vent size will be re-evaluated for compliance each time the P/T limit curves are revised based on the results of the vessel material surveillance.

The RCS vent is passive and is not subject to active failure.

The LTOP System satisfies Criterion 2 of the NRC Policy Statement.

B RAI 3.4.12-3

LCO

This LCO requires that the LTOP System is OPERABLE. The LTOP System is OPERABLE when the minimum coolant input and pressure relief capabilities are OPERABLE. Violation of this LCO could lead to the loss of low temperature overpressure mitigation and violation of the Reference 1 limits as a result of an operational transient.

3

a maximum of one SI pump

To limit the coolant input capability, the LCO requires [one] [HPI] pump [and one charging pump] capable of injecting into the RCS and all accumulator discharge isolation valves closed and immobilized. When accumulator pressure is greater than or equal to the maximum RCS pressure for the existing RCS cold leg temperature allowed in the PTLR.

B PTLR

LCO (continued)

The elements of the LCO that provide low temperature overpressure mitigation through pressure relief are:

One of the following pressure relief capabilities

2

a. Two RCS relief valves, as follows:

1. Two OPERABLE PORVs; or

A PORV is OPERABLE for LTOP when its block valve is open, its lift setpoint is set to the limit required by the PTLR and testing proves its ability to open at this setpoint, and motive power is available to the two valves and their control circuits.

3

~~2. Two OPERABLE RHR suction relief valves; or~~

~~An RHR suction relief valve is OPERABLE for LTOP when its RHR suction isolation valve and its RHR suction valve are open, its setpoint is at or between [436.5] psig and [463.5] psig, and testing has proven its ability to open at this setpoint.~~

~~3. One OPERABLE PORV and one OPERABLE RHR suction relief valve; or~~

2 2 b A depressurized RCS and an RCS vent.

19 An RCS vent is OPERABLE when open with ~~an area of ≥ [2.07] square inches.~~

a venting capability equivalent to or greater than a PORV.

19

Each of these methods of overpressure prevention is capable of mitigating the limiting LTOP transient.

B PTLR

APPLICABILITY

This LCO is applicable in MODE 4 when any RCS cold leg temperature is ≤ [275]°F, in MODE 5, and in MODE 6 when the reactor vessel head is on. The pressurizer safety valves provide overpressure protection that meets the Reference 1 P/T limits above [275]°F. When the reactor vessel head is off, overpressurization cannot occur.

the LTOP enabling temperature specified in the PTLR

6

LCO 3.4.3 provides the operational P/T limits for all MODES. LCO 3.4.10, "Pressurizer Safety Valves," requires the OPERABILITY of the pressurizer safety valves that provide overpressure protection during MODES 1, 2, and 3, and MODE 4 above ~~[275]°F~~.

B PTLR

APPLICABILITY (continued)

Low temperature overpressure prevention is most critical during shutdown when the RCS is water solid, and a mass or heat input transient can cause a very rapid increase in RCS pressure when little or no time allows operator action to mitigate the event.

7 ~~The Applicability is modified by a Note stating that accumulator isolation is only required when the accumulator pressure is more than or at the maximum RCS pressure for the existing temperature, as allowed by the P/T limit curves. This Note permits the accumulator discharge isolation valve Surveillance to be performed only under these pressure and temperature conditions.~~

ACTIONS Insert "D" 8

A.1 ~~[and B.1]~~ 3 SI pumps

With two or more HPI pumps capable of injecting into the RCS, RCS overpressurization is possible.

To immediately initiate action to restore restricted coolant input capability to the RCS reflects the urgency of removing the RCS from this condition.

B PTLR

3 ~~Required Action B.1 is modified by a Note that permits two charging pumps capable of RCS injection for ≤ 15 minutes to allow for pump swaps.~~

C.1, D.1, and D.2 B.1, C.1 and C.2 3

An unisolated accumulator requires isolation within 1 hour. This is only required when the accumulator pressure is at or more than the maximum RCS pressure for the existing temperature allowed by the P/T limit curves.

3 C

If isolation is needed and cannot be accomplished in 1 hour, Required Action D.1 and Required Action D.2 provide two options, either of which must be performed in the next 12 hours. By increasing the RCS temperature to > [275]°F, an accumulator pressure of [600] psig cannot exceed the LTOP limits if the accumulators are fully injected. Depressurizing the accumulators below the LTOP limit from the PTLR also gives this protection.

LTOP enabling temperature specified in the PTLR 6

3 800

B RAI 3.4.12-1 PTLR

ACTIONS (continued)

The Completion Times are based on operating experience that these activities can be accomplished in these time periods and on engineering evaluations indicating that an event requiring LTOP is not likely in the allowed times.

6 → LTOP enabling temperature specified in the PTLR

3 → D → E 1

B PTLR

11 → PORV

11 → PORVs

In MODE 4 when any RCS cold leg temperature is ≤ [275]°F with one required RCS relief valve inoperable, the ~~RCS~~ relief valve must be restored to OPERABLE status within a Completion Time of 7 days. Two RCS relief valves ~~[in any combination of the PORVS and the RHR suction relief valves]~~ are required to provide low temperature overpressure mitigation while withstanding a single failure of an active component.

11 → 3

11 → PORVs

The Completion Time considers the facts that only one of the RCS relief valves is required to mitigate an overpressure transient and that the likelihood of an active failure of the remaining valve path during this time period is very low.

3 → E → F 1

B PTLR

11 → PORVs

The consequences of operational events that will overpressurize the RCS are more severe at lower temperature (Ref. 7). Thus, with one of the two RCS relief valves inoperable in MODE 5 or in MODE 6 with the head on, the Completion Time to restore two valves to OPERABLE status is 24 hours.

11 → PORV

The Completion Time represents a reasonable time to investigate and repair several types of relief valve failures without exposure to a lengthy period with only one OPERABLE RCS relief valve to protect against overpressure events.

3 → F → G 1

B PTLR

The RCS must be depressurized and a vent must be established within 8 hours when:

PORVs ← 11

a. Both required RCS relief valves are inoperable; or

b. A Required Action and associated Completion Time of Condition A, ~~[B,]~~ D, E, or F is not met; or

3 C, D or E ← 3

B PTLR

ACTIONS (continued)

3

c. The LTOP System is inoperable for any reason other than Condition A, [B,] C, D, E, or F.

or E

B,

3

B
RAI 3.4.12-3
PTLR

19

path must have a venting capability equivalent to or greater than a PORV

The vent must be sized ≥ [2.07] square inches to ensure that the flow capacity is greater than that required for the worst case mass input transient reasonable during the applicable MODES. This action is needed to protect the RCPB from a low temperature overpressure event and a possible brittle failure of the reactor vessel.

The Completion Time considers the time required to place the plant in this Condition and the relatively low probability of an overpressure event during this time period due to increased operator awareness of administrative control requirements.

3

2

SURVEILLANCE REQUIREMENTS

3

SR 3.4.12.1, ~~[SR 3.4.12.2,]~~ and SR 3.4.12.3

B
PTLR

one SI is verified capable

To minimize the potential for a low temperature overpressure event by limiting the mass input capability, a maximum of [one] [HPI] pump [and a maximum of one charging pump] are verified incapable of injecting into the RCS and the accumulator discharge isolation valves are verified closed and locked out.

12

Insert "G"

The SI pump is rendered incapable of injecting into the RCS through removing the power from the pump by racking the breaker out under administrative control.

The [HPI] pump[s] and charging pump[s] are rendered incapable of injecting into the RCS through removing the power from the pumps by racking the breakers out under administrative control. An alternate method of LTOP control may be employed using at least two independent means to prevent a pump start such that a single failure or single action will not result in an injection into the RCS. This may be accomplished through the pump control switch being placed in [pull to lock] and at least one valve in the discharge flow path being closed.

B
PTLR

3

pull to lock

The Frequency of 12 hours is sufficient, considering other indications and alarms available to the operator in the control room, to verify the required status of the equipment.

SURVEILLANCE REQUIREMENTS (continued)

B PTLR

3

~~SR 3.4.12.4~~

~~Each required RHR suction relief valve shall be demonstrated OPERABLE by verifying its RHR suction valve and RHR suction isolation valves are open and by testing it in accordance with the Inservice Testing Program. (Refer to SR 3.4.12.7 for the RHR suction isolation valve Surveillance.) This Surveillance is only required to be performed if the RHR suction relief valve is being used to meet this LCO.~~

~~The RHR suction valve is verified to be opened every 12 hours. The Frequency is considered adequate in view of other administrative controls such as valve status indications available to the operator in the control room that verify the RHR suction valve remains open.~~

~~The ASME Code, Section XI (Ref. 8), test per Inservice Testing Program verifies OPERABILITY by proving proper relief valve mechanical motion and by measuring and, if required, adjusting the lift setpoint.~~

19

path with venting capability equivalent to or greater than a PORV

SR 3.4.12.5 3 3

B PTLR

The RCS vent of ≥ [2.07] square inches is proven OPERABLE by verifying its open condition either:

is not

(valves that are sealed or secured in the open position are considered "locked" in this context)

a. Once every 12 hours for a valve that ~~cannot be~~ locked.

b. Once every 31 days for a valve that is locked, sealed, or secured in position. A removed pressurizer safety valve fits this category.

other vent path(s) (e.g., a vent

or open manway also

The passive vent arrangement must only be open to be OPERABLE. This Surveillance is required to be performed if the vent is being used to satisfy the pressure relief requirements of the LCO 3.4.12b.

14

path

c.2 2

SR 3.4.12.6 4 3

15

The required trains of LTOP must be verified enabled

The PORV block valve must be verified open every 72 hours to provide the flow path for each required PORV to perform its function when actuated. The valve must be remotely verified open in the main control room. ~~[This Surveillance is performed if the PORV satisfies the LCO.]~~ 3

B PTLR

A LTOP train is verified enabled by ensuring its enabling switch is in the correct position and that the associated PORV Block Valve is open.

SURVEILLANCE REQUIREMENTS (continued)

The block valve is a remotely controlled, m otor operated valve. The power to the valve operator is not required removed, and the manual operator is not required locked in the inactive position. Thus, the block valve can be closed in the event the PORV develops excessive leakage or does not close (sticks open) after relieving an overpressure situation.

The 72 hour Frequency is considered adequate in view of other administrative controls available to the operator in the control room, such as valve position indication, that verify that the PORV block valve remains open.

~~SR 3.4.12.7~~

~~Each required RHR suction relief valve shall be demonstrated OPERABLE by verifying its RHR suction valve and RHR suction isolation valve are open and by testing it in accordance with the Inservice Testing Program. (Refer to SR 3.4.12.4 for the RHR suction valve Surveillance and for a description of the requirements of the Inservice Testing Program.) This Surveillance is only performed if the RHR suction relief valve is being used to satisfy this LCO.~~

~~Every 31 days the RHR suction isolation valve is verified locked open, with power to the valve operator removed, to ensure that accidental closure will not occur. The "locked open" valve must be locally verified in its open position with the manual actuator locked in its inactive position. The 31 day Frequency is based on engineering judgment, is consistent with the procedural controls governing valve operation, and ensures correct valve position.~~

SR 3.4.12.8 5 3

B PTLR

Approved TSTF-205 R.3

Insert K

16

Performance of a COT is required ~~within 12 hours after decreasing RCS temperature to ≤ [275]°F and~~ every 31 days on each required PORV to verify and, as necessary, adjust its lift setpoint. The COT will verify the setpoint is within the PTLR allowed maximum limits in the PTLR. PORV actuation could depressurize the RCS and is not required.

~~The 12 hour Frequency considers the unlikelihood of a low temperature overpressure event during this time.~~

16

~~A Note has been added indicating that this SR is required to be met 12 hours after decreasing RCS cold leg temperature to~~

SURVEILLANCE REQUIREMENTS (continued)

16

~~≤ [275]°F. The COT cannot be performed until in the LTOP MODES when the PORV lift setpoint can be reduced to the LTOP setting. The test must be performed within 12 hours after entering the LTOP MODES.~~

SR 3.4.12.9 6 3 18

B PTLR

Performance of a CHANNEL CALIBRATION on each required PORV actuation channel is required every [18] months to adjust the whole channel so that it responds and the valve opens within the required range and accuracy to known input.

17 Insert "H"

REFERENCES

1. 10 CFR 50, Appendix G.
2. Generic Letter 88-11.
3. ASME, Boiler and Pressure Vessel Code, Section III.
4. FSAR, Chapter [15] 14 3
5. 10 CFR 50, Section 50.46.
6. 10 CFR 50, Appendix K.
7. Generic Letter 90-06.
8. ASME, Boiler and Pressure Vessel Code, Section XI.

Insert "C"

The Low Temperature Overpressure Protection System consists of two control trains. The trains incorporate two key-operated enabling switches and two valve control switches in the control room. Signals from pressurizer pressure instrumentation and reactor coolant Loop A hot leg pressure instrumentation are used to control the PORVS. The pressurizer pressure instrumentation controls one PORV, while the reactor coolant pressure instrumentation controls the other PORV. B RAI 3.4.12-3

The protection circuits are enabled by turning the key switches to the enabled position. When the circuit is enabled and the PORV block valves are fully open, a red light above the respective key switch illuminates, signifying the circuits are armed. With both circuits properly armed, each PORV with its valve control switch in the Auto position will open, if system pressure increases to the lift setpoint.

Insert "D"

The ACTIONS are modified by a Note stating that while the LCO is not met, entry into MODE 6, with the reactor vessel head on, from MODE 6, with the reactor vessel head removed, is not permitted. This Note prevents entry into the MODES of applicability for LTOP without the requirements of LCO 3.4.12 being met. This Note is necessary, because LCO 3.0.4 is only applicable for entry into a MODE or other specified condition in the Applicability in MODES 1, 2, 3 and 4.

Insert "E"

Not Used.

B PTLR

Insert "F"

Not Used.

B

RAI 3.4.12-1

Insert "G"

when accumulator pressure is ≥ the maximum RCS pressure for existing cold leg temperature allowed by the P/T limit curves provided in the PTLR.

Insert "H"

SR 3.4.12.7 and SR 3.4.12.8

B

PTLR

Operating the PORVs, the solenoid air control valves and the check valves on the nitrogen gas bottles ensures the PORVs and PORV control system will actuate properly when called upon. The Frequency of 18 months is based on a typical refueling cycle and the frequency of other surveillances used to demonstrate PORV OPERABILITY.

Insert "I"

Not Used.

B

PTLR

Insert "J"

The reactor coolant system is defined as vented if there is an opening in the reactor coolant system pressure boundary to atmosphere or the pressurizer relief tank that has an equivalent system pressure relieving capability as a PORV. Some examples of such openings include an open or removed PORV, open steam generator or pressurizer manways, a removed pressurizer safety valve, and the top of the reactor vessel when the reactor vessel head has been unbolted or removed.

Insert "K"

A successful test of the required contact(s) of a channel relay may be performed by the verification of the change of state of a single contact of the relay. This clarifies what is an acceptable CHANNEL OPERATIONAL TEST of a relay. This is acceptable because all of the other required contacts of the relay are verified by other Technical Specifications and non-Technical Specifications tests at least once per refueling interval with applicable extensions.

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.12 Low Temperature Overpressure Protection (LTOP) System

B
RAI 3.4.12-3

LCO 3.4.12 An LTOP System shall be OPERABLE with:

a. A maximum of one Safety Injection (SI) pump capable of injecting into the RCS;

b. Each accumulator isolated, whose pressure is ≥ the maximum RCS pressure for the existing RCS cold leg temperature allowed by the P/T limit curves provided in the PTLR, and

B
PTLR

c. One of the following pressure relief capabilities:

1. Two power operated relief valves (PORVs) with lift settings within the limits specified in the PTLR, or

2. The RCS depressurized and an RCS vent path with venting capability equivalent to or greater than a PORV.

APPLICABILITY: MODE 4 when any RCS cold leg temperature is ≤ LTOP enabling temperature specified in the PTLR,
MODE 5,
MODE 6 when the reactor vessel head is on.

B
PTLR

ACTIONS

-----NOTE-----
While this LCO is not met, entry into MODE 6, with the reactor vessel head on, from MODE 6, with the reactor vessel head removed, is not permitted.

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. Two SI pumps capable of injecting into the RCS.	A.1	Initiate action to verify a maximum of one SI pump is capable of injecting into the RCS.	Immediately
B. An accumulator not isolated when the accumulator pressure is greater than or equal to the maximum RCS pressure for existing cold leg temperature allowed in the PTLR.	B.1	Isolate affected accumulator.	1 hour
C. Required Action and associated Completion Time of Condition B not met.	C.1	Increase RCS cold leg temperature to > LTOP enabling temperature specified in the PTLR.	12 hours
	<u>OR</u>		
	C.2	Depressurize affected accumulator to less than the maximum RCS pressure for existing cold leg temperature allowed in the PTLR.	12 hours

B
RAI 3.4.12-1
PTLR

(continued)

CONDITION	REQUIRED ACTION		COMPLETION TIME
D. One required PORV inoperable in MODE 4.	D.1	Restore required PORV to OPERABLE status.	7 days
E. One required PORV inoperable in MODE 5 or 6.	E.1	Restore required PORV to OPERABLE status.	24 hours
F. Two required PORVs inoperable. <u>OR</u> Required Action and associated Completion Time of Condition A, C, D or E not met. <u>OR</u> LTOP System inoperable for any reason other than Condition A, B, C, D or E.	F.1	Depressurize RCS and establish RCS vent path with venting capability equivalent to or greater than a PORV.	8 hours

B PTLR

B PTLR

B PTLR

B RAI 3.4.12-3 PTLR

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.4.12.1	Verify a maximum of one SI pump is capable of injecting into the RCS.	12 hours
SR 3.4.12.2	-----NOTE----- Only required when accumulator pressure is ≥ the maximum RCS pressure for existing cold leg temperature allowed by the P/T limit curves provided in the PTLR. ----- Verify each accumulator is isolated.	12 hours
SR 3.4.12.3	-----NOTE----- Only required to be performed when complying with LCO 3.4.12.c.2. ----- Verify required RCS vent path with venting capability equivalent to or greater than a PORV.	12 hours for unlocked open vent valve(s) AND 31 days for other vent path(s)
SR 3.4.12.4	Verify required trains of LTOP armed.	72 hours
SR 3.4.12.5	Perform a COT on each required PORV, excluding actuation.	31 days

B RAI 3.4.12-2 PTLR

B PTLR

B PTLR

B PTLR

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY	
SR 3.4.12.6	Perform CHANNEL CALIBRATION for each required PORV actuation channel.	18 months	B PTLR
SR 3.4.12.7	Perform a complete cycle of each required PORV solenoid air control valve and check valve on the nitrogen gas bottles.	18 months	B PTLR
SR 3.4.12.8	Perform a complete cycle of each required PORV.	18 months	B PTLR

B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.12 Low Temperature Overpressure Protection (LTOP) System

BASES

BACKGROUND

The LTOP System controls RCS pressure at low temperatures so the integrity of the reactor coolant pressure boundary (RCPB) is not compromised by violating the pressure and temperature (P/T) limits of 10 CFR 50, Appendix G (Ref. 1). The reactor vessel is the limiting RCPB component for demonstrating such protection. The PTLR provides the maximum allowable actuation logic setpoints for the power operated relief valves (PORVs) and the maximum RCS pressure for the existing RCS cold leg temperature during cooldown, shutdown, and heatup to meet the Reference 1 requirements during the LTOP MODES. B RAI 3.4.12-3

The reactor vessel material is less tough at low temperatures than at normal operating temperature. As the vessel neutron exposure accumulates, the material toughness decreases and becomes less resistant to pressure stress at low temperatures (Ref. 2). RCS pressure, therefore, is maintained low at low temperatures and is increased only as temperature is increased.

The potential for vessel overpressurization is most acute when the RCS is water solid, occurring only while shutdown; a pressure fluctuation can occur more quickly than an operator can react to relieve the condition. Exceeding the RCS P/T limits by a significant amount could cause brittle cracking of the reactor vessel. LCO 3.4.3, "RCS Pressure and Temperature (P/T) Limits," requires administrative control of RCS pressure and temperature during heatup and cooldown to prevent exceeding the PTLR limits.

This LCO provides RCS overpressure protection by having a minimum coolant input capability and having adequate pressure relief capacity. Limiting coolant input capability requires all but one Safety Injection (SI) pump incapable of injection into the RCS and isolating the accumulators. The pressure relief capacity requires either two redundant PORVs or a depressurized RCS and an RCS vent of sufficient size. One PORV or the open RCS vent is the overpressure protection device that acts to terminate an increasing pressure event. B PTLR

With minimum coolant input capability, the ability to provide core coolant addition is restricted. The LCO does not require the makeup control system deactivated or the safety injection (SI) actuation circuits blocked. Due to the lower pressures in the LTOP MODES and the expected core decay heat levels, the makeup system can provide

BASES

BACKGROUND (continued)

adequate flow via the makeup control valve. If conditions require the use of more than one SI pump for makeup in the event of loss of inventory, then pumps can be made available through manual actions. B PTLR

The LTOP System for pressure relief consists of two PORVs with reduced lift settings, or a depressurized RCS and an RCS vent of sufficient size. Two PORVs are required for redundancy. One PORV has adequate relieving capability to keep from overpressurization for the required coolant input capability. B RAI 3.4.12-3

PORV Requirements

The Low Temperature Overpressure Protection System consists of two control trains. The trains incorporate two key-operated enabling switches and two valve control switches in the control room. Signals from pressurizer pressure instrumentation and reactor coolant Loop A hot leg pressure instrumentation are used to control the PORVS. The pressurizer pressure instrumentation controls one PORV, while the reactor coolant pressure instrumentation controls the other PORV. B RAI 3.4.12-3

The protection circuits are enabled by turning the key switches to the enabled position. When the circuit is enabled and the PORV block valves are fully open, a red light above the respective key switch illuminates, signifying the circuits are armed. With both circuits properly armed, each PORV with its valve control switch in the Auto position will open, if system pressure increases to the lift setpoint.

The PTLR presents the PORV setpoints for LTOP. Having the setpoints of both valves within the limits in the PTLR ensures that the Reference 1 limits will not be exceeded in any analyzed event.

When a PORV is opened in an increasing pressure transient, the release of coolant will cause the pressure increase to slow and reverse. As the PORV releases coolant, the RCS pressure decreases until a reset pressure is reached and the valve is signaled to close. The pressure continues to decrease below the reset pressure as the valve closes.

RCS Vent Requirements

Once the RCS is depressurized, a vent exposed to the containment atmosphere will maintain the RCS at containment ambient pressure in an RCS overpressure transient, if the relieving requirements of the transient do not exceed the capabilities of the vent. Thus, the vent path must be capable of relieving the flow resulting from the limiting LTOP mass or heat input transient, and maintaining pressure below the P/T limits.

BACKGROUND (continued)

The RCS is defined as vented if there is an opening in the reactor coolant system pressure boundary to atmosphere or the pressurizer relief tank that has an equivalent system pressure relieving capability as a PORV. Some examples of such openings include an open or removed PORV, open steam generator or pressurizer manways, a removed pressurizer safety valve, and the top of the reactor vessel when the reactor vessel head has been unbolted or removed. The vent path(s) must be above the level of reactor coolant, so as not to drain the RCS when open.

The required vent capacity may be provided by one or more vent paths.

APPLICABLE SAFETY ANALYSES

Safety analyses (Ref. 4) demonstrate that the reactor vessel is adequately protected against exceeding the Reference 1 P/T limits. In MODES 1, 2, 3 and in MODE 4 with RCS cold leg temperature exceeding the LTOP enabling temperature specified in the PTLR, the pressurizer safety valves will prevent RCS pressure from exceeding the Reference 1 limits. At the LTOP arming temperature specified in the PTLR and below, overpressure prevention falls to two OPERABLE PORVs or to a depressurized RCS and a sufficient sized RCS vent. Each of these means has a limited overpressure relief capability. B PTLR

The actual temperature at which the pressure in the P/T limit curve falls below the pressurizer safety valve setpoint increases as the reactor vessel material toughness decreases due to neutron embrittlement. Each time the PTLR curves are revised, the LTOP System must be re-evaluated to ensure its functional requirements can still be met using the PORV method or the depressurized and vented RCS condition. B RAI 3.4.12-3

The PTLR contains the acceptance limits that define the LTOP requirements. Any change to the RCS must be evaluated against the Reference 4 analyses to determine the impact of the change on the LTOP acceptance limits.

Transients that are capable of overpressurizing the RCS are categorized as either mass or heat input transients, examples of which follow:

<u>Mass Input Type Transients</u>

a. Inadvertent safety injection; or

b. Charging/letdown flow mismatch.

BASES

APPLICABLE SAFETY ANALYSES (continued)

Heat Input Type Transients

a. Inadvertent actuation of pressurizer heaters;

b. Loss of RHR cooling; or

c. Reactor coolant pump (RCP) startup with temperature asymmetry within the RCS or between the RCS and steam generators.

The following are required during the LTOP MODES to ensure that mass and heat input transients do not occur, which either of the LTOP overpressure protection means cannot handle:

a. Rendering all but one SI pump incapable of injection;

b. Deactivating the accumulator discharge isolation valves in their closed positions; and

B PTLR

c. Disallowing start of an RCP if secondary temperature is more than 50°F above primary temperature in any one loop. LCO 3.4.6, "RCS Loops — MODE 4," and LCO 3.4.7, "RCS Loops — MODE 5, Loops Filled," provide this protection.

The Reference 4 analyses demonstrate that either one PORV or the depressurized RCS and RCS vent can maintain RCS pressure below limits when only one SI pump is actuated. Thus, the LCO allows only one SI pump OPERABLE during the LTOP MODES. Since neither one PORV nor the RCS vent can handle the pressure transient need from accumulator injection, when RCS temperature is low, the LCO also requires the accumulators isolation when accumulator pressure is greater than or equal to the maximum RCS pressure for the existing RCS cold leg temperature allowed in the PTLR.

B PTLR

The isolated accumulators must have their discharge valves closed and the valve power supply breakers fixed in their open positions. The analyses show the effect of accumulator discharge is over a narrower RCS temperature range (approximately 265°F and below) than that of the LCO (270°F and below).

Fracture mechanics analyses established the temperature of LTOP Applicability at 270°F.

B PTLR

BASES

APPLICABLE SAFETY ANALYSES (continued)

The consequences of a small break loss of coolant accident (LOCA) in LTOP MODE 4 conform to 10 CFR 50.46 and 10 CFR 50, Appendix K (Refs. 5 and 6), requirements by having a maximum of one SI pump OPERABLE and SI actuation enabled. B PTLR

PORV Performance

The fracture mechanics analyses show that the vessel is protected when the PORVs are set to open at or below the limit shown in the PTLR. The setpoints are derived by analyses that model the performance of the LTOP System, assuming the limiting LTOP transient of one SI pump injecting into the RCS. These analyses consider pressure overshoot and undershoot beyond the PORV opening and closing, resulting from signal processing and valve stroke times. The PORV setpoints at or below the derived limit ensures the Reference 1 P/T limits will be met. B RAI 3.4.12-3 PTLR

The PORV setpoints in the PTLR will be updated when the revised P/T limits conflict with the LTOP analysis limits. The P/T limits are periodically modified as the reactor vessel material toughness decreases due to neutron embrittlement caused by neutron irradiation. Revised limits are determined using neutron fluence projections and the results of examinations of the reactor vessel material irradiation surveillance specimens. The Bases for LCO 3.4.3, "RCS Pressure and Temperature (P/T) Limits," discuss these examinations.

The PORVs are considered active components. Thus, the failure of one PORV is assumed to represent the worst case, single active failure.

RCS Vent Performance

With the RCS depressurized, analyses show a vent path with venting capability equivalent to or greater than a PORV is capable of mitigating the allowed LTOP overpressure transient. The capacity of a vent this size is greater than the flow of the limiting transient for the LTOP configuration, one SI pump OPERABLE, maintaining RCS pressure less than the maximum pressure on the P/T limit curve. B PTLR

The RCS vent size will be re-evaluated for compliance each time the P/T limit curves are revised based on the results of the vessel material surveillance.

The RCS vent is passive and is not subject to active failure.

The LTOP System satisfies Criterion 2 of the NRC Policy Statement.

BASES

LCO

This LCO requires that the LTOP System is OPERABLE. The LTOP System is OPERABLE when the minimum coolant input and pressure relief capabilities are OPERABLE. Violation of this LCO could lead to the loss of low temperature overpressure mitigation and violation of the Reference 1 limits as a result of an operational transient. B RAI 3.4.12-3

To limit the coolant input capability, the LCO requires a maximum of one SI pump capable of injecting into the RCS and all accumulator discharge isolation valves closed and immobilized. When accumulator pressure is greater than or equal to the maximum RCS pressure for the existing RCS cold leg temperature allowed in the PTLR. B PTLR

The elements of the LCO that provide low temperature overpressure mitigation through pressure relief are:

a. One of the following pressure relief capabilities:

   1. Two OPERABLE PORVs; or

      A PORV is OPERABLE for LTOP when its block valve is open, its lift setpoint is set to the limit required by the PTLR and testing proves its ability to open at this setpoint, and motive power is available to the two valves and their control circuits.

   2. A depressurized RCS and an RCS vent.

An RCS vent is OPERABLE when open with a venting capability equivalent to or greater than a PORV.

Each of these methods of overpressure prevention is capable of mitigating the limiting LTOP transient.

APPLICABILITY

This LCO is applicable in MODE 4 when any RCS cold leg temperature is ≤ the LTOP enabling temperature specified in the PTLR, in MODE 5, and in MODE 6 when the reactor vessel head is on. The pressurizer safety valves provide overpressure protection that meets the Reference 1 P/T limits above the LTOP enabling temperature specified in the PTLR. When the reactor vessel head is off, overpressurization cannot occur. B PTLR

LCO 3.4.3 provides the operational P/T limits for all MODES. LCO 3.4.10, "Pressurizer Safety Valves," requires the OPERABILITY of the pressurizer safety valves that provide overpressure protection during MODES 1, 2, and 3 and MODE 4 above the LTOP enabling temperature specified in the PTLR. B PTLR

BASES

APPLICABILITY (continued)	Low temperature overpressure prevention is most critical during shutdown when the RCS is water solid, and a mass or heat input transient can cause a very rapid increase in RCS pressure when little or no time allows operator action to mitigate the event.

ACTIONS

The ACTIONS are modified by a Note stating that while the LCO is not met, entry into MODE 6, with the reactor vessel head on, from MODE 6, with the reactor vessel head removed, is not permitted. This Note prevents entry into the MODES of applicability for LTOP without the requirements of LCO 3.4.12 being met. This Note is necessary, because LCO 3.0.4 is only applicable for entry into a MODE or other specified condition in the Applicability in MODES 1, 2, 3 and 4.

A.1

With two SI pumps capable of injecting into the RCS, RCS overpressurization is possible. B PTLR

To immediately initiate action to restore restricted coolant input capability to the RCS reflects the urgency of removing the RCS from this condition.

B.1, C.1 and C.2 B RAI 3.4.12-1 PTLR

An unisolated accumulator requires isolation within 1 hour. This is only required when the accumulator pressure is at or more than the maximum RCS pressure for the existing temperature allowed by the P/T limit curves.

If isolation is needed and cannot be accomplished in 1 hour, Required Action C.1 and Required Action C.2 provide two options, either of which must be performed in the next 12 hours. By increasing the RCS temperature to > LTOP enabling temperature specified in the PTLR, an accumulator pressure of 800 psig cannot exceed the LTOP limits if the accumulators are fully injected. Depressurizing the accumulators below the LTOP limit from the PTLR also gives this protection. B RAI 3.4.12-1 PTLR

The Completion Times are based on operating experience that these activities can be accomplished in these time periods and on engineering evaluations indicating that an event requiring LTOP is not likely in the allowed times.

BASES

ACTIONS (continued) D.1

In MODE 4 when any RCS cold leg temperature is ≤ LTOP enabling temperature specified in the PTLR, with one required PORV inoperable, the PORV must be restored to OPERABLE status within a Completion Time of 7 days. Two PORVs are required to provide low temperature overpressure mitigation while withstanding a single failure of an active component.

B PTLR

The Completion Time considers the facts that only one of the PORVs is required to mitigate an overpressure transient and that the likelihood of an active failure of the remaining valve path during this time period is very low.

E.1

B PTLR

The consequences of operational events that will overpressurize the RCS are more severe at lower temperature (Ref. 7). Thus, with one of the two PORVs inoperable in MODE 5 or in MODE 6 with the head on, the Completion Time to restore two valves to OPERABLE status is 24 hours.

The Completion Time represents a reasonable time to investigate and repair several types of relief valve failures without exposure to a lengthy period with only one OPERABLE PORV to protect against overpressure events.

F.1

B PTLR

The RCS must be depressurized and a vent must be established within 8 hours when:

a. Both required PORVs are inoperable; or

b. A Required Action and associated Completion Time of Condition A, C, D or E is not met; or

c. The LTOP System is inoperable for any reason other than Condition A, B, C, D or E.

B RAI 3.4.12-3 PTLR

The vent path must have a venting capability equivalent to or greater than a PORV to ensure that the flow capacity is greater than that required for the worst case mass input transient reasonable during the applicable MODES. This action is needed to protect the RCPB from a low temperature overpressure event and a possible brittle failure of the reactor vessel.

BASES

ACTIONS (continued)

The Completion Time considers the time required to place the plant in this Condition and the relatively low probability of an overpressure event during this time period due to increased operator awareness of administrative control requirements.

SURVEILLANCE REQUIREMENTS

SR 3.4.12.1 and SR 3.4.12.2 B PTLR

To minimize the potential for a low temperature overpressure event by limiting the mass input capability, a maximum of one SI pump is verified capable of injecting into the RCS and the accumulator discharge isolation valves are verified closed and locked out when accumulator pressure is ≥ the maximum RCS pressure for existing cold leg temperature allowed by the P/T limit curves provided in the PTLR. B PTLR

The SI pump is rendered incapable of injecting into the RCS through removing the power from the pump by racking the breaker out under administrative control. An alternate method of LTOP control may be employed using at least two independent means to prevent a pump start such that a single failure or single action will not result in an injection into the RCS. This may be accomplished through the pump control switch being placed in pull to lock and at least one valve in the discharge flow path being closed. B PTLR

The Frequency of 12 hours is sufficient, considering other indications and alarms available to the operator in the control room, to verify the required status of the equipment.

SR 3.4.12.3 B PTLR

The RCS vent path with a venting capability equivalent or greater than a PORV is proven OPERABLE by verifying its open condition either:

a. Once every 12 hours for a valve that is not locked (valves that are sealed or secured in the open position are considered "locked" in this context).

b. Once every 31 days for other vent path(s) (e.g., a vent or a valve that is locked, sealed, or secured in position). A removed pressurizer safety valve or open manway also fits this category.

The passive vent path arrangement must only be open when required to be OPERABLE. This Surveillance is required to be performed if the vent is being used to satisfy the pressure relief requirements of the LCO 3.4.12.c.2.

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.4.12.4

The required trains of LTOP must be verified enabled every 72 hours to provide the flow path for each required PORV to perform its function when actuated. A LTOP train is verified enabled by ensuring its enabling switch is in the correct position and that the associated PORV Block Valve is open.

B PTLR

The block valve is a remotely controlled, motor operated valve. The power to the valve operator is not required removed, and the manual operator is not required locked in the inactive position. Thus, the block valve can be closed in the event the PORV develops excessive leakage or does not close (sticks open) after relieving an overpressure situation.

The 72 hour Frequency is considered adequate in view of other administrative controls available to the operator in the control room, such as valve position indication, that verify that the PORV block valve remains open.

SR 3.4.12.5

B PTLR

Performance of a COT is required every 31 days on each required PORV to verify and, as necessary, adjust its lift setpoint. A successful test of the required contact(s) of a channel relay may be performed by the verification of the change of state of a single contact of the relay. This clarifies what is an acceptable CHANNEL OPERATIONAL TEST of a relay. This is acceptable because all of the other required contacts of the relay are verified by other Technical Specifications and non-Technical Specifications tests at least once per refueling interval with applicable extensions. The COT will verify the setpoint is within the PTLR allowed maximum limits in the PTLR. PORV actuation could depressurize the RCS and is not required.

SR 3.4.12.6

B PTLR

Performance of a CHANNEL CALIBRATION on each required PORV actuation channel is required every 18 months to adjust the whole channel so that it responds and the valve opens within the required range and accuracy to known input.

SR 3.4.12.7 and SR 3.4.12.8

B PTLR

Operating the PORVs, the solenoid air control valves and the check valves on the nitrogen gas bottles ensures the PORVs and PORV control system will actuate properly when called upon. The Frequency of 18 months is based on a typical refueling cycle and the frequency of other surveillances used to demonstrate PORV OPERABILITY.

BASES

REFERENCES

1. 10 CFR 50, Appendix G.
2. Generic Letter 88-11.
3. ASME, Boiler and Pressure Vessel Code, Section III.
4. FSAR, Chapter 14
5. 10 CFR 50, Section 50.46.
6. 10 CFR 50, Appendix K.
7. Generic Letter 90-06.
8. ASME, Boiler and Pressure Vessel Code, Section XI.

Description of Changes - NUREG-1431 Section 3.04.13

03-Aug-00

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

CTS:	ITS:
15.03.01.D.01	LCO 3.04.13 COND A
15.03.01.D.02	LCO 3.04.13 C
15.03.01.D.04	LCO 3.04.13 COND B
	LCO 3.04.13 D
15.03.01.D.05	LCO 3.04.13 A
	LCO 3.04.13 COND B

DOC Number	DOC Text
A.02 Rev. A	CTS 15.3.1.D.1 specifies that a follow-up evaluation of the safety implications shall be initiated as soon as practicable, but no later than within 4 hours, if leakage of reactor coolant from the RCS is indicated to exceed 1 gpm. CTS 15.3.1.D.2 requires a reactor shutdown be initiated as soon as practical, but no later than 24 hours after the leak was detected. Proposed ITS LCO 3.4.13, Condition A, requires RCS leakage that is not within the limits (other than pressure boundary leakage), be reduced to within the limits within 24 hours. This allows time to verify leakage rates and either identify unidentified leakage or reduce leakage to within the limits before the reactor must be shutdown. Requiring these actions be completed within 24 hours is consistent with the CTS requirements.

CTS:	ITS:
15.03.01.D.01	LCO 3.04.13 COND A

DOC Number	DOC Text
A.03 Rev. A	CTS 15.3.1.D is revised to adopt NUREG-1431 SR 3.4.13.2, which requires verification of the SG Tube Surveillance Program. This surveillance requirement emphasizes the importance of SG Tube integrity. This change is administrative, because the SG Tube Surveillance Program already exists in CTS 15.4.2.A, and proposed SR 3.4.13.2 does not impose any new requirements.

CTS:	ITS:
15.04.02.A	SR 3.04.13.02

Description of Changes - NUREG-1431 Section 3.04.13

03-Aug-00

DOC Number	DOC Text	
A.04 Rev. A	CTS 15.3.1.D.6 requires that the reactor not be restarted until the leak is repaired or until the problem is otherwise corrected. Proposed LCO 3.0.4 states when an LCO is not met, entry into a MODE in the Applicability shall not be made except when the associated ACTIONS to be entered permit continued operation in the MODE in the Applicability for an unlimited period of time. Proposed LCO 3.4.13 has Applicability in MODES 1, 2, 3 and 4, and the ACTIONS of LCO 3.4.13 do not permit continued operation in any of these MODES for an unlimited period of time. Therefore, the statement of CTS 15.3.1.D.6 is not required, and is not retained in ITS.	
	CTS:	**ITS:**
	15.03.01.D.06	N/A
A.05 Rev. B	The Bases of the current Technical Specifications for this section have been completely replaced by revised Bases that reflect the format and applicable content of PBNP ITS Chapter 3.4, consistent with the Standard Technical Specifications for Westinghouse Plants, NUREG-1431. The revised Bases are as shown in the PBNP ITS Bases.	
	CTS:	**ITS:**
	BASES	B 3.04.13
L.01 Rev. A	CTS 15.3.1.D.4 requires the reactor be shutdown and the plant be placed in cold shutdown within 30 hours of detection of exceeding primary to secondary leakage limits. Proposed ITS LCO 3.4.13, Conditions A and B require that the leakage be returned to within limits in 4 hours, or be in MODE 3 in 6 hours and in MODE 5 in 36 hours. This is a relaxation of requirements and is less restrictive, but is acceptable. The proposed time requirement has been shown to be a reasonable time, based on industry experience, to reach MODE 5 from full power conditions in an orderly manner without challenging plant systems. Additional consideration has shown that there is a low probability of further degradation of the RCPB in the additional time interval.	
	CTS:	**ITS:**
	15.03.01.D.04	LCO 3.04.13 COND B RA B.1 LCO 3.04.13 COND B RA B.2
L.02 Rev. A	CTS 15.4.1, Table 15.4.1-2, Item 16, Primary System Leakage Evaluation, is modified by Note (6), which states the surveillance is not required during periods of refueling shutdown. Per ITS SR 3.0.1, surveillance requirements shall be met during the MODES in the Applicability for individual LCOs. Therefore, SR 3.4.13.1 is required to be met during MODES 1, 2, 3 and 4. Deleting this note and adopting the applicability of ITS LCO 3.4.13 is less restrictive, but is acceptable because these are the conditions where the RCS is pressurized. In MODES 5 and 6, leakage limits are not required because the reactor coolant pressure is far lower, resulting in lower stresses and reduced potentials for leakage. Furthermore, adopting the Applicability of ITS LCO 3.4.13, establishes consistency with the requirements of CTS 15.3.1.D.	
	CTS:	**ITS:**
	15.04.01 T 15.04.01-02 16 (6)	LCO 3.04.13

DOC Number	DOC Text	
L.03 Rev. A	CTS 15.4.1, Table 15.4.1-2, Item 16, Primary System Leakage Evaluation, is revised to adopt a Note which states, the performance of the surveillance requirement is not required in MODES 3 or 4 until 12 hours of steady state operation. This change is less restrictive, but is acceptable because steady state operation is required to perform a proper RCS water inventory balance. These calculations include data dependent on RCS pressure, temperature, power level, pressurizer and makeup tank levels, makeup and letdown, and RCP seal injection and return flows. Changes occurring in these parameters during maneuvering invalidate the data, making the calculations useless. Therefore, this surveillance is not required to be performed in MODES 3 or 4 until 12 hours of steady state operation near operating pressure have been established.	
	CTS:	**ITS:**
	NEW	SR 3.04.13.01 NOTE
L.04 Rev. B	CTS 15.3.1.D.1 and 15.3.1.D.3 require a follow-up evaluation of the safety implications of the RCS leakage and provide information to be considered and contained in the evaluation concerning plant shutdown and exposure to offsite personnel. The requirement to perform an evaluation and the details of the information to be included are not being retained in the ITS, because they are not required to provide adequate protection of the public health and safety. The purpose of ITS LCO 3.4.13 is to limit system operation in the presence of LEAKAGE from these sources to amounts that do not compromise safety. ITS LCO 3.4.13 provides leakage limits for pressure boundary leakage, identified leakage, unidentified leakage, and primary to secondary leakage through a SG. Separating the identified LEAKAGE from the unidentified LEAKAGE is necessary to provide quantitative information to the operators, allowing them to take corrective action should a leak occur that is detrimental to the safety of the facility and the public. Therefore, specifying a requirement to perform an evaluation is unnecessary.	
	CTS:	**ITS:**
	15.03.01.D.01	N/A
	15.03.01.D.03	N/A
LA.01 Rev. B	CTS 15.3.1.D.1 provides means by which leakage of reactor coolant from the RCS can be indicated. These details are being deleted from Technical Specifications, and are moved to the Bases. This information provides details which are not directly pertinent to the actual requirement, and are not required to be in the ITS to provide adequate protection to the public health and safety. Changes to these details will be controlled in accordance with the provisions of the Bases Control Program described in Chapter 5 of the Improved Technical Specifications and the 50.59 process as applicable.	
	CTS:	**ITS:**
	15.03.01.D.01	N/A

Description of Changes - NUREG-1431 Section 3.04.13

03-Aug-00

DOC Number	DOC Text	
LA.02 Rev. A	CTS 15.3.1.D.1 states any identified leakage shall be considered to be real leakage until it is determined that either (1) a safety problem does not exist or (2) that the indicated leak cannot be substantiated by direct observation or other indication. These details are being deleted from Technical Specifications and are moved to licensee control. These details are not required to be in the ITS to provide adequate protection to the public health and safety. Changes to plant procedures and other plant controlled documents are subject to controls imposed by plant administrative procedures, which endorse applicable regulations and standards.	
	CTS:	**ITS:**
	15.03.01.D.01	N/A
LA.03 Rev. B	Not Used.	
	CTS:	**ITS:**
	N/A	N/A
M.01 Rev. A	CTS 15.3.1.D.2 provides limits on continued plant operation, if reactor coolant leakage is substantiated and is not evaluated as safe or is determined to exceed 10 gpm. Proposed ITS LCO 3.4.13 includes the following RCS operational leakage requirements; 1 gpm unidentified leakage, and 10 gpm identified leakage. Limiting unidentified leakage to 1 gpm is a reasonable minimum detectable amount that the containment air monitoring and containment sump level monitoring equipment can detect within a reasonable time period. Adopting this limit places additional requirements on plant operation and is, therefore, more restrictive.	
	CTS:	**ITS:**
	15.03.01.D.01	LCO 3.04.13 COND A RA A.1
M.02 Rev. A	CTS 15.3.1.D.2 requires a reactor shutdown be initiated as soon as practicable, but no later than within 24 hours after the leak was first detected, if the indicated leakage is not evaluated as safe or exceeds 10 gpm. Proposed ITS LCO 3.4.13 requires the leakage be returned to within limits in 24 hours, or be in MODE 3 in 6 hours and in MODE 5 in 36 hours. Adopting the requirement to place the plant in MODE 5 in 36 hours lowers the likelihood of further deterioration. These proposed actions place additional requirements on plant operation and are, therefore, more restrictive.	
	CTS:	**ITS:**
	15.03.01.D.02	LCO 3.04.13 B LCO 3.04.13 COND B

DOC Number	DOC Text	
M.03 Rev. A	CTS 15.3.1.D does not specifically state the plant conditions for which the requirements apply. However, the Actions contained in CTS 15.3.1.D.2 require the plant to be shutdown, if leakage is unsafe or exceeds 10 gpm, and CTS 15.3.1.D.4 and 15.3.1.D.5 require the plant to be shutdown and cooled down to the cold shutdown condition, when primary to secondary SG leakage exceeds 500 gpd in either SG, or leakage exists from the RCPB, respectively. These actions imply the requirements are applicable when the plant is above the cold shutdown condition (MODE 5). Proposed ITS 3.4.13 has Applicability of MODES 1, 2, 3 and 4. Adopting the Applicability statement of LCO 3.4.13 places additional requirements on plant operation and is, therefore, more restrictive.	
	CTS:	**ITS:**
	15.03.01.D.02	LCO 3.04.13
		LCO 3.04.13
M.04 Rev. A	CTS 15.3.1.D.5 requires a reactor shutdown and cooldown to the cold shutdown condition be initiated within 24 hours of detection, if reactor coolant leakage exists through a non-isolable fault in a reactor coolant system component. Proposed ITS LCO 3.4.13 requires that the unit be in MODE 3 in 6 hours and in MODE 5 in 36 hours, if pressure boundary leakage exists. Requiring the plant be in MODE 3 in 6 hours and in MODE 5 in 36 hours, is more restrictive than requiring "shutdown and cooldown to the cold shutdown condition" be initiated within 24 hours.	
	CTS:	**ITS:**
	15.03.01.D.05	LCO 3.04.13 COND B RA B.1
		LCO 3.04.13 COND B RA B.2
M.05 Rev. A	CTS 15.4.1, Table 15.4.1-2, Item 16, requires a monthly evaluation of primary system leakage. Proposed ITS SR 3.4.13.1 requires the performance of a RCS water inventory balance at a frequency of 72 hours during steady state operation. The 72 hour frequency is more restrictive, but is a reasonable interval to trend leakage and recognizes the importance of early leakage detection in the prevention of accidents.	
	CTS:	**ITS:**
	15.04.01 T 15.04.01-02 16	SR 3.04.13.01

DOC Number	DOC Text
R.01 Rev. B	Wisconsin Electric Power Company has utilized the selection criteria provided in the 10 CFR 50.36.ii, and has concluded that the Primary System Testing LCO can be relocated to licensee control. The basis for this conclusion is as follows:

Primary system testing is used to verify the integrity of the primary system after the system is closed following normal opening, modification or repair. This surveillance is not used continuously and does not provide any automatic protection functions. The RCS system integrity requirements (leakage limits and surveillances) are provided as specific requirements in NUREG-1431, Specifications 3.4.13, 3.4.14, and 3.4.15. Therefore, this surveillance requirement is not needed.

Comparison to Screening Criteria:

1. The associated primary system testing is not used for detecting a significant abnormal degradation of the reactor coolant pressure boundary prior to a design basis accident (DBA).

2. The associated primary system testing is not used to indicate status of, or monitor a process variable, design feature, or operating restriction that is an initial condition of a DBA or transient.

3. The associated primary system testing is not part of a primary success path in the mitigation of a DBA or transient.

4. The risk contribution of primary system testing to core damage frequency is not evaluated in WCAP-11618 or the Point Beach PRA.

Conclusion:

The Primary System Testing LCO may be relocated to other plant controlled documents outside Technical Specifications because the screening criteria have not been satisfied, and the appropriate RCS integrity requirements will continue to be maintained within the appropriate Technical Specifications.

CTS:	ITS:
15.04.03	TRM 3.07.03
15.04.03.A	TRM 3.07.03
15.04.03.A.01	TRM 3.07.03
15.04.03.A.02	TRM 3.07.03
15.04.03.B	TRM 3.07.03
15.04.03.C	TRM 3.07.03

D. LEAKAGE OF REACTOR COOLANT

Specification:

1. If leakage of reactor coolant from the reactor coolant system is indicated to exceed 1 gpm ~~by the means available such as water inventory balances, monitoring equipment or direct observation,~~ a follow-up evaluation of the safety implications shall be initiated as soon as practicable but no later than within 4 hours. ~~Any indicated leak shall be considered to be a real leak until it is determined that either (1) a safety problem does not exist or (2) that the indicated leak cannot be substantiated by direct observation or other indication.~~ LA.1 A.2 L.4 B RAI 3.4.13-2 LA.2

2. If the indicated reactor coolant leakage is substantiated and is not evaluated as safe or is determined to exceed 10 gpm, reactor shutdown shall be initiated as soon as practicable, but no later than within 24 hours after the leak was first detected. M.1 M.2

3. ~~The nature of the leak as well as the magnitude of the leak shall be considered in the safety evaluation. If plant shutdown is necessary per specification 2 above, the rate of shutdown and the conditions of shutdown shall be determined by the safety evaluation for each case and justified in writing as soon thereafter as practicable. The safety evaluation shall assure that the exposure of offsite personnel to radiation from the primary system coolant activity is within the guidelines of 10 CFR 20.~~ L.4 B RAI 3.4.13-2

Add "APPLICABILITY: MODES 1, 2, 3, and 4." M.3

Add:

SURVEILLANCE		FREQUENCY
SR 3.4.13.2	Verify steam generator tube integrity is in accordance with the Steam Generator Tube Surveillance Program.	In accordance with the Steam Generator Tube Surveillance Program

A.3

No Significant Hazards Considerations - NUREG-1431 Section 3.04.13

01-Aug-00

NSHC Number	NSHC Text
A Rev. A	In accordance with the criteria set forth in 10 CFR 50.92, PBNP has evaluated this proposed Technical Specifications change and determined it does not represent a significant hazards consideration. The following is provided in support of this conclusion. 1. Does the change involve a significant increase in the probability or consequences of an accident previously evaluated? The proposed change involves reformatting and rewording of the current Technical Specifications. The reformatting and rewording process involves no technical changes to existing requirements. As such, this change is administrative in nature and does not impact initiators of analyzed events or assumed mitigation of accident or transient events. Therefore, this change does not increase the probability or consequences of an accident previously evaluated. 2. Does the change create the possibility of a new or different kind of accident from any accident previously evaluated? The proposed change does not require a physical alteration of the plant (no new or different type of equipment will be installed) or changes in parameters governing normal plant operation. The proposed change will not impose any new or eliminate any old requirements. Thus, this change does not create the possibility of a new or different kind of accident from any accident previously evaluated. 3. Does this change involve a significant reduction in a margin of safety? The proposed change will not significantly reduce the margin of safety because it has no impact on any safety analysis assumptions. This change is administrative. As such, there is no technical change to the requirements and, therefore, there is no reduction in the margin of safety.

No Significant Hazards Considerations - NUREG-1431 Section 3.04.13

01-Aug-00

NSHC Number	NSHC Text
L.01 Rev. A	In accordance with the criteria set forth in 10 CFR 50.92, PBNP has evaluated this proposed Technical Specifications change and determined it does not represent a significant hazards consideration. The following is provided in support of this conclusion. 1. Does the change involve a significant increase in the probability or consequences of an accident previously evaluated? 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 increases the time allowed to place the plant in MODE 5, when the primary to secondary leakage in either SG exceeds 500 gpd. The proposed time requirement has been shown to be a reasonable time, based on industry experience, to reach MODE 5 from full power conditions in an orderly manner without challenging plant systems. 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. This change increases the time allowed to place the plant in MODE 5, when the primary to secondary leakage in either SG exceeds 500 gpd. Additional consideration has shown that there is a low probability of further degradation of the RCPB associated with the increased time interval. Therefore, this change does not involve a 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. This change results in the deletion of a note modifying the Primary Leakage System Evaluation surveillance frequency, stating it was not required during periods of refueling shutdown, and adopts an applicability of MODES 1, 2, 3, and 4. This is acceptable because these are the conditions where the RCS is pressurized. In MODES 5 and 6, leakage limits are not required because the reactor coolant pressure is far lower, resulting in lower stresses and reduced potentials for leakage. 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 RCS Operational Leakage are properly maintained. Therefore, this change does not involve a 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 Primary System Leakage Evaluation surveillance is revised to adopt a Note which states, the performance of the surveillance requirement is not required in MODES 3 or 4 until 12 hours of steady state operation. This change is acceptable because steady state operation is required to perform a proper RCS water inventory balance. These calculations include data dependent on RCS pressure, temperature, power level, pressurizer and makeup tank levels, makeup and letdown, and RCP seal injection and return flows. Plant changes that affect these parameters invalidate the data, making the calculations useless. 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 RCS Operational Leakage are properly maintained. Therefore, this change does not involve a reduction in a margin of safety.

NSHC Number	NSHC Text
L.04 Rev. B	In accordance with the criteria set forth in 10 CFR 50.92, PBNP has evaluated this proposed Technical Specifications change and determined it does not represent a significant hazards consideration. The following is provided in support of this conclusion. 1. Does the change involve a significant increase in the probability or consequences of an accident previously evaluated? The proposed change does not involve any physical alteration of plant systems, structures or components, changes in parameters governing normal plant operation, or methods of operation. This change results in the deletion of the requirement to perform an evaluation and the details of the information to be considered and contained in the evaluation concerning plant shutdown and exposure to offsite personnel. This information provides details which are not directly pertinent to the actual requirement, and are not required to provide adequate protection of the 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. 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 RCS Operational Leakage are properly maintained. Therefore, this change does not involve a reduction in a margin of safety.

NSHC Number	NSHC Text
LA Rev. A	In accordance with the criteria set forth in 10 CFR 50.92, PBNP has evaluated this proposed Technical Specifications change and determined it does not represent a significant hazards consideration. The following is provided in support of this conclusion. 1. Does the change involve a significant increase in the probability or consequences of an accident previously evaluated? The proposed change relocates requirements from the Technical Specifications to the Bases, FSAR, or other plant controlled documents. The Bases and FSAR will be maintained using the provisions of 10 CFR 50.59. In addition to 10 CFR 50.59 provisions, the Technical Specifications Bases are subject to the change process in the Administrative Controls Chapter of the ITS. Plant procedures and other plant controlled documents are subject to controls imposed by plant administrative procedures, which endorse applicable regulations and standards. Changes to the Bases, FSAR, or other plant controlled documents will be evaluated in accordance with the requirements of the Bases Control Program in Chapter 5.0 of the ITS, 10 CFR 50.59, or plant administrative processes. Therefore, no increase in the probability or consequences of an accident previously evaluated will be allowed. 2. Does the change create the possibility of a new or different kind of accident from any accident previously evaluated? The proposed change does not require a physical alteration of the plant (no new or different type of equipment will be installed) or changes in parameters governing normal plant operation. The proposed change will not impose any different requirements and adequate control of the information will be maintained. Thus, this change does not create the possibility of a new or different kind of accident from any accident previously evaluated. 3. Does this change involve a significant reduction in a margin of safety? The proposed change will not reduce a margin of safety because it has no impact on any safety analysis assumptions. In addition, the requirements to be moved from the Technical Specifications to the Bases, FSAR, or other plant controlled documents are as they currently exist. Future changes to the requirements in the Bases, FSAR, or other plant controlled documents will be evaluated in accordance with the requirements of 10 CFR 50.59, the Bases Control Program in Chapter 5.0 of the ITS, or the applicable plant process and no reduction in a margin of safety will be allowed.

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. B	In accordance with the criteria set forth in 10 CFR 50.92, PBNP has evaluated this proposed Technical Specifications change and determined it does not represent a significant hazards consideration. The following is provided in support of this conclusion. 1. Does the change involve a significant increase in the probability or consequences of an accident previously evaluated? The proposed change 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 significant reduction in a margin of safety.

BACKGROUND (Continued)

PIVs are provided to isolate the RCS from the following typically connected systems:

1

a. Residual Heat Removal (RHR) System;

b. Safety Injection System; and Technical Requirements Manual

B

4

~~c. Chemical and Volume Control System.~~

The PIVs are listed in the FSAR, Section [ ] (Ref. 6).

Violation of this LCO could result in continued degradation of a PIV, which could lead to overpressurization of a low pressure system and the loss of the integrity of a fission product barrier.

APPLICABLE SAFETY ANALYSES

Reference 4 identified potential intersystem LOCAs as a significant contributor to the risk of core melt. The dominant accident sequence in the intersystem LOCA category is the failure of the low pressure portion of the RHR System outside of containment. The accident is the result of a postulated failure of the PIVs, which are part of the RCPB, and the subsequent pressurization of the RHR System downstream of the PIVs from the RCS. Because the low pressure portion of the RHR System is typically designed for 600 psig, overpressurization failure of the RHR low pressure line would result in a LOCA outside containment and subsequent risk of core melt.

3

~~Reference 5 evaluated various PIV configurations, leakage testing of the valves, and operational changes to determine the effect on the probability of intersystem LOCAs. This study concluded that periodic leakage testing of the PIVs can substantially reduce the probability of an intersystem LOCA.~~

RCS PIV leakage satisfies Criterion 2 of the NRC Policy Statement.

LCO

RCS PIV leakage is identified LEAKAGE into closed systems connected to the RCS. Isolation valve leakage is usually on the order of drops per minute. Leakage that increases

SURVEILLANCE REQUIREMENTS (continued)

1

~~opened is set so the actual RCS pressure must be < [425] psig to open the valves. This setpoint ensures the RHR design pressure will not be exceeded and the RHR relief valves will not lift. The [18] month Frequency is based on the need to perform the Surveillance under conditions that apply during a plant outage. The [18] month Frequency is also acceptable based on consideration of the design reliability (and confirming operatin g experience) of the equipment.~~

~~These SRs are modified by Notes allowing the RHR autoclosure function to be disabled when using the RHR System suction relief valves for cold overpressure protection in accordance with SR 3.4.12.7.~~

REFERENCES

1. 10 CFR 50.2.
2. 10 CFR 50.55a(c).
3. 10 CFR 50, Appendix A, Section V, GDC 55 .

Event V Order, April 20, 1981.

3

4. WASH-1400 (NUREG-75/014), Appendix V, October 1975.
5. NUREG-0677, May 1980.

Technical Requirements Manual

6. [Document containing list of PIVs.]

1

B

7. ASME, Boiler and Pressure Vessel Code, Section XI.
8. 10 CFR 50.55a(g).

B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.14 RCS Pressure Isolation Valve (PIV) Leakage

BASES

BACKGROUND

Event V order, issued April 20, 1981, specifies certain PIVs which are required to be leak tested periodically. During their lives, these valves can produce varying amounts of reactor coolant leakage through either normal operational wear or mechanical deterioration. The RCS PIV Leakage LCO allows RCS high pressure operation when leakage through these valves exists in amounts that do not compromise safety.

The PIV leakage limit applies to each individual valve. Leakage through both series PIVs in a line must be included as part of the identified LEAKAGE, governed by LCO 3.4.13, "RCS Operational LEAKAGE." This is true during operation only when the loss of RCS mass through two series valves is determined by a water inventory balance (SR 3.4.13.1). A known component of the identified LEAKAGE before operation begins is the least of the two individual leak rates determined for leaking series PIVs during the required surveillance testing; leakage measured through one PIV in a line is not RCS operational LEAKAGE if the other is leaktight.

Although this specification provides a limit on allowable PIV leakage rate, its main purpose is to prevent overpressure failure of the low pressure portions of connecting systems. The leakage limit is an indication that the PIVs between the RCS and the connecting systems are degraded or degrading. PIV leakage could lead to overpressure of the low pressure piping or components. Failure consequences could be a loss of coolant accident (LOCA) outside of containment, an unanalyzed accident, that could degrade the ability for low pressure injection.

PIVs are provided to isolate the RCS from the following typically connected systems:

a. Residual Heat Removal (RHR) System;

b. Safety Injection System; and

The PIVs are listed in the Technical Requirements Manual (Ref. 6). B

Violation of this LCO could result in continued degradation of a PIV, which could lead to overpressurization of a low pressure system and the loss of the integrity of a fission product barrier.

BASES

APPLICABLE SAFETY ANALYSES

Reference 4 identified potential intersystem LOCAs as a significant contributor to the risk of core melt. The dominant accident sequence in the intersystem LOCA category is the failure of the low pressure portion of the RHR System outside of containment. The accident is the result of a postulated failure of the PIVs, which are part of the RCPB, and the subsequent pressurization of the RHR System downstream of the PIVs from the RCS. Because the low pressure portion of the RHR System is typically designed for 600 psig, overpressurization failure of the RHR low pressure line would result in a LOCA outside containment and subsequent risk of core melt.

RCS PIV leakage satisfies Criterion 2 of the NRC Policy Statement.

LCO

RCS PIV leakage is identified LEAKAGE into closed systems connected to the RCS. Isolation valve leakage is usually on the order of drops per minute. Leakage that increases significantly suggests that something is operationally wrong and corrective action must be taken.

Leakage rates ≤ 1.0 gpm are acceptable. Leakage rates > 1.0 gpm, but ≤ 5.0 gpm are 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. Leakage rates > 1.0 gpm ≤ 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. Leakage rates > 5.0 gpm are considered unacceptable.

Reference 7 permits leakage testing at a lower pressure differential than between the specified maximum RCS pressure and the normal pressure of the connected system during RCS operation (the maximum pressure differential) in those types of valves in which the higher service pressure will tend to diminish the overall leakage channel opening. In such cases, the observed rate may be adjusted to the maximum pressure differential by assuming leakage is directly proportional to the pressure differential to the one half power.

BASES

APPLICABILITY

In MODES 1, 2, 3, and 4, this LCO applies because the PIV leakage potential is greatest when the RCS is pressurized. In MODE 4, valves in the RHR flow path are not required to meet the requirements of this LCO when in, or during the transition to or from, the RHR mode of operation.

In MODES 5 and 6, leakage limits are not provided because the lower reactor coolant pressure results in a reduced potential for leakage and for a LOCA outside the containment.

ACTIONS

The Actions are modified by two Notes. Note 1 provides clarification that each flow path allows separate entry into a Condition. This is allowed based upon the functional independence of the flow path. Note 2 requires an evaluation of affected systems if a PIV is inoperable. The leakage may have affected system operability, or isolation of a leaking flow path with an alternate valve may have degraded the ability of the interconnected system to perform its safety function.

A.1 and A.2

The flow path must be isolated by two valves. Required Actions A.1 and A.2 are modified by a Note that the valves used for isolation must meet the same leakage requirements as the PIVs and must be within the RCPB or the high pressure portion of the system.

Required Action A.1 requires that the isolation with one valve must be performed within 4 hours. Four hours provides time to reduce leakage in excess of the allowable limit and to isolate the affected system if leakage cannot be reduced.

The 4 hour Completion Time allows the actions and restricts the operation with leaking isolation valves.

Required Action A.2 specifies that the double isolation barrier of two valves be restored by closing some other valve qualified for isolation or restoring one leaking PIV. The 72 hour Completion Time after exceeding the limit considers the time required to complete the Action and the low probability of a second valve failing during this time period.

BASES

ACTIONS (continued)

B.1 and B.2

If leakage cannot be reduced, the system isolated, or the other Required Actions accomplished, the plant must be brought to a MODE in which the requirement does not apply. To achieve this status, the plant must be brought to MODE 3 within 6 hours and MODE 5 within 36 hours. This Action may reduce the leakage and also reduces the potential for a LOCA outside the containment. The allowed Completion Times are reasonable based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.4.14.1

Performance of leakage testing on each RCS PIV or isolation valve used to satisfy Required Action A.1 and Required Action A.2 is required to verify that leakage is below the limit contained in the PIV Leakage Program and to identify each leaking valve. Leakage testing requires a stable pressure condition.

For the two PIVs in series, the leakage requirement applies to each valve individually and not to the combined leakage across both valves. If the PIVs are not individually leakage tested, one valve may have failed completely and not be detected if the other valve in series meets the leakage requirement. In this situation, the protection provided by redundant valves would be lost.

Testing is to be performed every 18 months, a typical refueling cycle, if the plant does not go into MODE 5 for at least 7 days. The 18 month Frequency is consistent with 10 CFR 50.55a(g) (Ref. 8) as contained in the Inservice Testing Program, is within frequency allowed by the American Society of Mechanical Engineers (ASME) Code, Section XI (Ref. 7), and is based on the need to perform such surveillances under the conditions that apply during an outage and the potential for an unplanned transient if the Surveillance were performed with the reactor at power.

In addition, testing must be performed once after the valve has been opened by flow or exercised to ensure tight reseating. PIVs disturbed in the performance of this Surveillance should also be tested unless documentation shows that an infinite testing loop cannot practically be avoided. Testing must be performed within 24 hours after the valve has been reseated. Within 24 hours is a reasonable and practical time limit for performing this test after opening or reseating a valve.

BASES

SURVEILLANCE REQUIREMENTS (continued)	The leakage limit is to be met at the RCS pressure associated with MODES 1 and 2. This permits leakage testing at high differential pressures with stable conditions not possible in the MODES with lower pressures. Entry into MODES 3 and 4 is allowed to establish the necessary differential pressures and stable conditions to allow for performance of this Surveillance. The Note that allows this provision is complementary to the Frequency of prior to entry into MODE 2 whenever the unit has been in MODE 5 for 7 days or more, if leakage testing has not been performed in the previous 9 months. In addition, this Surveillance is not required to be performed on the RHR System when the RHR System is aligned to the RCS in the shutdown cooling mode of operation. PIVs contained in the RHR shutdown cooling flow path must be leakage rate tested after RHR is secured and stable unit conditions and the necessary differential pressures are established.
REFERENCES	1. 10 CFR 50.2. 2. 10 CFR 50.55a(c). 3. Event V Order, April 20, 1981. 4. WASH-1400 (NUREG-75/014), Appendix V, October 1975. 5. NUREG-0677, May 1980. 6. Technical Requirements Manual. B 7. ASME, Boiler and Pressure Vessel Code, Section XI. 8. 10 CFR 50.55a(g).

DOC Number	DOC Text	
L.04 Rev. B	CTS 15.4.1, Table 15.4.1-2, Item 1, requires reactor coolant samples be analyzed for gross activity at a frequency of 5/week. This surveillance requirement is modified by Note (7), which states "At least once per week during periods of refueling shutdown." The requirement to sample reactor coolant when the unit is shutdown and RCS average temperature is less than 500 F is not being retained in ITS. This is a relaxation of requirements and is less restrictive. This change is acceptable, because the LCO limit for gross specific activity when operating in MODES 1 and 2, and in MODE 3 with RCS average temperature greater than or equal to 500 F, is necessary to contain the potential consequences of a steam generator tube rupture (SGTR) to within acceptable site boundary dose values. When the unit is operating with RCS average temperature less than 500 F, the release of radioactivity in the event of a SGTR is unlikely, because the saturation pressure of the reactor coolant is below the lift pressure settings of the main steam safety valves.	
	CTS:	**ITS:**
	15.04.01 T 15.04.01-02 01.A (7)	N/A
LA.01 Rev. B	Not Used.	
	CTS:	**ITS:**
	N/A	N/A
LA.02 Rev. A	CTS 15.4.1, Table 15.4.1-2, Item 1, requires the reactor coolant sample be analyzed for tritium activity monthly. This requirement is not being retained in ITS, but is being moved to licensee controlled documents. This specification is not required to be in the ITS to provide adequate protection to the public health and safety, because ITS still retains the RCS specific activity limitations. This approach provides for an effective level of regulatory control and provides for a more appropriate change control process. The level of safety of facility operation is unaffected by the change, because there is no change in the overall operational requirements.	
	CTS:	**ITS:**
	15.04.01 T 15.04.01-02 01.B	TRM 3.04.01
LA.03 Rev. A	CTS 15.4.1, Table 15.4.1-2, Item 1, requires the determination of radiochemical E-bar (E) semiannually. This requirement is modified by Note (2) which states E determination will be started when the gross activity analysis of a filtered sample indicates greater than or equal to 10 microcuries/cc and will be redetermined if the primary coolant gross radioactivity of a filtered sample increases by more than 10 microcuries/cc. This note is not being retained in ITS, but is being moved to licensee controlled documents. This specification is not required to be in the ITS to provide adequate protection to the public health and safety, because ITS still retains the RCS specific activity limitations. This approach provides for an effective level of regulatory control and provides for a more appropriate change control process. The level of safety of facility operation is unaffected by the change, because there is no change in the overall operational requirements.	
	CTS:	**ITS:**
	15.04.01 T 15.04.01-02 01.C (2)	TRM 3.04.01

A.1

TABLE 15. 4.1-2

MINIMUM FREQUENCIES FOR EQUIPMENT AND SAMPLING TESTS

			Test	Frequency	
	1.	Reactor Coolant Samples	Gross Beta-gamma activity (excluding tritium)	5/week(7)	L.2, L.4, B
			Tritium activity	Monthly	LA.2
SR 3.4.16.3			Radiochemical $\overline{E}$ Determination	Semiannually(2)(10)	LA.3, M.1
SR 3.4.16.2			Isotopic Analysis for Dose Equivalent I-131 Concentration	Every two weeks(1)	L.3
COND A			Isotopic Analysis for Iodine including I-131, I-133, and I-135	a.) Once per 4 hours whenever the specific activity exceeds 0.8 μCi/ gram Dose Equivalent I-131 or 100/E μCi/g(6)	M.2, L.1, M.3
SR 3.4.16.2				b.) One sample between 2 and 6 hours following a thermal power change exceeding 15% of rated power in a one-hour period.	M.3
			Chloride Concentration	5/week(8)	R.1
			Diss. Oxygen Conc.	5/week(6)	R.1
			Fluoride Conc.	Weekly	R.1
	2.	Reactor Coolant Boron	Boron Concentration	Twice/week	< See LCO 3.9.1 >
	3.	Refueling Water Storage Tank Water Sample	Boron Concentration	Weekly(6)	< See LCO 3.5.4 >
	4.	Boric Acid Tanks	Boron Concentration	Twice/week and after each BAST concentration change when they are being relied upon as a source of borated water .	< See LCO 3.5.2 >
	5.	Spray Additive Tank	NaOH Concentration	Monthly	< See LCO 3.6 >
	6.	Accumulator	Boron Concentration	Monthly	< See LCO 3.5.1 >

TABLE 15.4.1-2 (Continued)

30.	Pressurizer Heaters	Verify that 100 KW of heaters are available.	Quarterly
31.	CVCS Charging Pumps	Verify operability pumps.(17)	Quarterly
32.	Potential Dilution in Progress Alarm	Verify operability of alarm.	Prior to placing plant in cold shutdown.
33.	Core Power Distribution	Perform power distribution maps using movable incore detector system to confirm hot channel factors.	Monthly (20)
34.	Shutdown Margin	Perform shutdown margin calculation	Daily (21)

< See LCO 3.4.9 >

< See LCO 3.5.2 >

< See LCO 3.3.9 >

< See LCOs 3.2.1 and 3.2.2 >

Associated Specification removed with Amendment 176/180.

< See Section 3.1 >

(1) Required only during periods of power operation. L.3

(2) E determination will be started when the gross activity analysis of a filtered sample indicates ≥10μCi/cc and will be redetermined if the primary coolant gross radioactivity of a filtered sample increases by more than 10μCi/cc. LA.3

(3) Drop test shall be conducted at rated reactor coolant flow. Rods shall be dropped under both cold and hot condition, but cold drop tests need not be timed. < See LCO 3.1.5 >

(4) Drop tests will be conducted in the hot condition for rods on which maintenance was performed.

(5) As accessible without disassembly of rotor.

(6) Not required during periods of refueling shutdown . L.1

(7) At least once per week during periods of refueling shutdown L.4

(8) At least three times per week (with maximum time of 72 hours between samples) during periods of refueling shutdown. R.1

B

(9) Not required during periods of cold or refueling shutdown, but must be performed prior to exceeding 200°F if it has not been performed during the previous surveillance period. < See LCOs 3.3.1, 3.6.3 >

(10) Sample to be taken after a minimum of 2 EFPD and 20 days power operation since the reactor was last subcritical for 48 hours or longer.

(11) An approximately equal number of valves shall be tested each refueling outage such that all valves will be tested within a five year period. If any valve fails its tests, an additional number of valves equal to the number originally tested shall be tested. If any of the additional tested valves fail, all remaining valves shall be tested.

(12) The specified buses shall be determined energized in the required manner at least once per shift by verifying correct static transfer switch alignment and indicated voltage on the buses. < See Section 3.8 >

(13) Not required if the block valve is shut to isolate a PORV that is inoperable for reasons other than excessive seat leakage.

< See LCO 3.4.11 > M.1

(14) Only applicable when the overpressure mitigation system is in service.

(15) Required to be performed only if conditions will be established, as defined in Specification 15.3.15, where the PORVs are used for low temperature overpressure protection. The test must be performed prior to establishing these conditions .

< See LCO 3.4.12 >

< See LCO 3.4.10 and 3.7.1 >

No Significant Hazards Considerations - NUREG-1431 Section 3.04.16

03-Aug-00

NSHC Number	NSHC Text
A Rev. A	In accordance with the criteria set forth in 10 CFR 50.92, PBNP has evaluated this proposed Technical Specifications change and determined it does not represent a significant hazards consideration. The following is provided in support of this conclusion. 1. Does the change involve a significant increase in the probability or consequences of an accident previously evaluated? The proposed change involves reformatting and rewording of the current Technical Specifications. The reformatting and rewording process involves no technical changes to existing requirements. As such, this change is administrative in nature and does not impact initiators of analyzed events or assumed mitigation of accident or transient events. Therefore, this change does not increase the probability or consequences of an accident previously evaluated. 2. Does the change create the possibility of a new or different kind of accident from any accident previously evaluated? The proposed change does not require a physical alteration of the plant (no new or different type of equipment will be installed) or changes in parameters governing normal plant operation. The proposed change will not impose any new or eliminate any old requirements. Thus, this change does not create the possibility of a new or different kind of accident from any accident previously evaluated. 3. Does this change involve a significant reduction in a margin of safety? The proposed change will not significantly reduce the margin of safety because it has no impact on any safety analysis assumptions. This change is administrative. As such, there is no technical change to the requirements and, therefore, there is no reduction in the margin of safety.

NSHC Number	NSHC Text
L.01 Rev. A	In accordance with the criteria set forth in 10 CFR 50.92, PBNP has evaluated this proposed Technical Specifications change and determined it does not represent a significant hazards consideration. The following is provided in support of this conclusion. 1. Does the change involve a significant increase in the probability or consequences of an accident previously evaluated? The proposed change 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 deletes the requirement to perform an isotopic analysis for Iodine once per 4 hours when reactor coolant activity exceeds 100/E microcuries/gram. If reactor coolant exceeds 100/E micocuries/gram, the reactor is required to be shutdown and cooled down to < 500 F in 6 hours, thereby placing the unit in a condition where the limits do not apply, and the analysis would not be required to be performed. 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 the 2 hour site boundary dose levels during the DBA are within acceptable limits. Therefore, this change does not involve a 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. This change reduces the frequency of performing gross activity analyses on the reactor coolant from 5/week to once per 7 days. This surveillance provides an indication of any increase in gross specific activity and trending the results of these analyses allows for proper remedial action to be taken before reaching the LCO limit under normal operating conditions. Relaxation of the frequency considers the unlikelihood of a gross fuel failure during the extended 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 the 2 hour site boundary dose levels during the DBA are within acceptable limits. Therefore, this change does not involve a 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. This change results in a reduction in the plant conditions under which verification of reactor coolant Dose Equivalent I-131 activity is required. The production of iodine activity in the reactor coolant is reduced between 2% and 5% of rated power, and is bounded by the assumptions made in the analysis of the SGTR accident. 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 the 2 hour site boundary dose levels during the DBA are within acceptable limits. Therefore, this change does not involve a reduction in a margin of safety.

NSHC Number	NSHC Text
L.04 Rev. B	In accordance with the criteria set forth in 10 CFR 50.92, PBNP has evaluated this proposed Technical Specifications change and determined it does not represent a significant hazards consideration. The following is provided in support of this conclusion.

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

The proposed change does not involve any physical alteration of plant systems, structures or components, changes in parameters governing normal plant operation, or methods of operation. This change results in a deletion of the requirement to sample reactor coolant when the unit is shutdown and RCS average temperature is less than 500 F. The LCO limit for gross specific activity when operating in MODES 1 and 2, and in MODE 3 with RCS average temperature greater than or equal to 500 F, is necessary to contain the potential consequences of a steam generator tube rupture (SGTR) to within acceptable site boundary dose values. When the unit is operating with RCS average temperature less than 500 F, the release of radioactivity in the event of a SGTR is unlikely, because the saturation pressure of the reactor coolant is below the lift pressure settings of the main steam safety valves. 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 the site boundary dose levels during the DBA are within acceptable limits. Therefore, this 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
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.

Justification For Deviations - NUREG-1431 Section 3.09.01

26-Jun-00

JFD Number	JFD Text	
01 Rev. A	Reference to the General Design Criteria (GDC) of 10 CFR 50 Appendix A has been deleted from the Bases of the Technical Specifications. Point Beach was constructed and licensed prior to the GDC being issued. The Point Beach construction permit was issued prior to the GDCs being issued in 1971. Point Beach was designed and constructed utilizing the 1967 proposed GDCs. Accordingly, reference has been provided to the appropriate criteria and section of the Point Beach FSAR which provides explanation of Point Beach's design basis.	
	ITS:	**NUREG:**
	B 3.09.01	B 3.09.01
02 Rev. B	Not used.	
	ITS:	**NUREG:**
	N/A	N/A
03 Rev. A	LCO 3.9.2 "Unborated Water Source Isolation Valves" was not adopted, based on the Point Beach design. Accordingly, the references to LCO 3.9.5 and 6 within the Bases for LCO 3.9.1 have been revised to reflect the renumbering that has occurred in Section 3.9 of the ITS.	
	ITS:	**NUREG:**
	B 3.09.01	B 3.09.01
04 Rev. A	With the incorporation of TSTF-9 (relocation of SDM to COLR), the differences between LCO 3.1.1 and LCO 3.1.2 are removed and LCO 3.1.2 is incorporated into LCO 3.1.1. This change eliminates the reference to LCO 3.1.2 from LCO 3.9.1 Bases. This change is consistent with TSTF 136, which has been approved for incorporation into revision two of NUREG 1431.	
	ITS:	**NUREG:**
	B 3.09.01	B 3.09.01
05 Rev. A	Requiring the verification of the boron concentration of the coolant in each volume is inconsistent with the other statements in the Bases. The Background and Applicable Safety Analysis sections state that the RCS, refueling cavity and refueling canal volumes are mixed and form a single mass. Therefore requiring sampling and analysis at more than one location is redundant and unnecessary.	
	ITS:	**NUREG:**
	B 3.09.01	B 3.09.01

B 3.9 REFUELING OPERATIONS

B 3.9.1 Boron Concentration

BASES

BACKGROUND

The limit on the boron concentrations of the Reactor Coolant System (RCS), the refueling canal, and the refueling cavity during refueling ensures that the reactor remains subcritical during MODE 6. Refueling boron concentration is the soluble boron concentration in the coolant in each of these volumes having direct access to the reactor core during refueling.

The soluble boron concentration offsets the core reactivity and is measured by chemical analysis of a representative sample of the coolant in each of the volumes. The refueling boron concentration limit is specified in the COLR. Plant procedures ensure the specified boron concentration in order to maintain an overall core reactivity of $k_{eff} \leq 0.95$ during fuel handling, with control rods and fuel assemblies assumed to be in the most adverse configuration (least negative reactivity) allowed by plant procedures.

1

Point Beach design criteria

GDC 26 of 10 CFR 50, Appendix A, require that two independent reactivity control systems of different design principles be provided (Ref. 1). One of these systems must be capable of holding the reactor core subcritical under cold conditions. The Chemical and Volume Control System (CVCS) is the system capable of maintaining the reactor subcritical in cold conditions by maintaining the boron concentration.

The reactor is brought to shutdown conditions before beginning operations to open the reactor vessel for refueling. After the RCS is cooled and depressurized and the vessel head is unbolted, the head is slowly removed to form the refueling cavity. The refueling canal and the refueling cavity are then flooded with borated water from the refueling water storage tank through the open reactor vessel by gravity feeding or by the use of the Residual Heat Removal (RHR) System pumps.

B

The pumping action of the RHR System in the RCS and the natural circulation due to thermal driving heads in the reactor vessel and refueling cavity mix the added concentrated boric acid with the water in the refueling

(continued)

BASES

ACTIONS

A.3 (continued)

Once actions have been initiated, they must be continued until the boron concentration is restored. The restoration time depends on the amount of boron that must be injected to reach the required concentration.

SURVEILLANCE REQUIREMENTS

SR 3.9.1.1

5

This SR ensures that the coolant boron concentra tion in the RCS, the refueling canal, and the refueling cavity is within the COLR limits. The boron concentration ~~of the coolant in each volume~~ is determined periodically by chemical analysis. B

of a representative sample of the interconnected volumes

5

A minimum Frequency of once every 72 hours is a reasonable amount of time to verify the boron concentration of representative samples. The Frequency is based on operating experience, which has shown 72 hours to be adequate.

REFERENCES

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

FSAR Sections 1.3.5, 3.1 and 9.3.

1

2. FSAR, Chapter [15]

14.1.4

6

B 3.9 REFUELING OPERATIONS

B 3.9.1 Boron Concentration

BASES

BACKGROUND

The limit on the boron concentrations of the Reactor Coolant System (RCS), the refueling canal, and the refueling cavity during refueling ensures that the reactor remains subcritical during MODE 6. Refueling boron concentration is the soluble boron concentration in the coolant in each of these volumes having direct access to the reactor core during refueling.

The soluble boron concentration offsets the core reactivity and is measured by chemical analysis of a representative sample of the coolant in each of the volumes. The refueling boron concentration limit is specified in the COLR. Plant procedures ensure the specified boron concentration in order to maintain an overall core reactivity of $k_{eff} \leq 0.95$ during fuel handling, with control rods and fuel assemblies assumed to be in the most adverse configuration (least negative reactivity) allowed by plant procedures.

Point Beach design criteria require that two independent reactivity control systems of different design principles be provided (Ref. 1). One of these systems must be capable of holding the reactor core subcritical under cold conditions. The Chemical and Volume Control System (CVCS) is the system capable of maintaining the reactor subcritical in cold conditions by maintaining the boron concentration.

The reactor is brought to shutdown conditions before beginning operations to open the reactor vessel for refueling. After the RCS is cooled and depressurized and the vessel head is unbolted, the head is slowly removed to form the refueling cavity. The refueling canal and the refueling cavity are then flooded with borated water from the refueling water storage tank through the open reactor vessel by gravity feeding or by use of the Residual Heat Removal (RHR) System pumps. B

The pumping action of the RHR System in the RCS and the natural circulation due to thermal driving heads in the reactor vessel and refueling cavity mix the added concentrated boric acid with the water in the refueling canal. The RHR System is in operation during refueling (see LCO 3.9.4, "Residual Heat Removal (RHR) and Coolant Circulation-High Water Level," and LCO 3.9.5, "Residual Heat Removal (RHR) and Coolant Circulation-Low Water Level") to provide forced circulation in the RCS and assist in maintaining the boron concentrations in the RCS, the refueling canal, and the refueling cavity above the COLR limit.

BASES

APPLICABLE SAFETY ANALYSES	During refueling operations, the reactivity condition of the core is consistent with the initial conditions assumed for the boron dilution accident in the accident analysis and is conservative for MODE 6. The boron concentration limit specified in the COLR is based on the core reactivity at the beginning of each fuel cycle (the end of refueling) and includes an uncertainty allowance. The required boron concentration and the plant refueling procedures that verify the correct fuel loading plan (including full core mapping) ensure that the k_{eff} of the core will remain ≤ 0.95 during the refueling operation. Hence, at least a 5% $\Delta k/k$ margin of safety is established during refueling. During refueling, the water volume in the spent fuel pool, the transfer canal, the refueling canal, the refueling cavity, and the reactor vessel form a single mass. As a result, the soluble boron concentration is relatively the same in each of these volumes. The limiting boron dilution accident analyzed occurs in MODE 5 (Ref. 2). A detailed discussion of this event is provided in Bases B 3.1.1, "SHUTDOWN MARGIN (SDM)." The RCS boron concentration satisfies Criterion 2 of the NRC Policy Statement.
LCO	The LCO requires that a minimum boron concentration be maintained in the RCS, the refueling canal, and the refueling cavity while in MODE 6. The boron concentration limit specified in the COLR ensures that a core k_{eff} of ≤ 0.95 is maintained during fuel handling operations. Violation of the LCO could lead to an inadvertent criticality during MODE 6.
APPLICABILITY	This LCO is applicable in MODE 6 to ensure that the fuel in the reactor vessel will remain subcritical. The required boron concentration ensures a $k_{eff} \leq 0.95$. Above MODE 6, LCO 3.1.1, "SHUTDOWN MARGIN (SDM)," ensure that an adequate amount of negative reactivity is available to shut down the reactor and maintain it subcritical.
ACTIONS	A.1 and A.2 Continuation of CORE ALTERATIONS or positive reactivity additions (including actions to reduce boron concentration) is contingent upon maintaining the unit in compliance with the LCO. If the boron concentration of any coolant volume in the RCS, the refueling canal, or

BASES

ACTIONS (continued)

the refueling cavity is less than its limit, all operations involving CORE ALTERATIONS or positive reactivity additions must be suspended immediately.

Suspension of CORE ALTERATIONS and positive reactivity additions shall not preclude moving a component to a safe position.

A.3

In addition to immediately suspending CORE ALTERATIONS or positive reactivity additions, boration to restore the concentration must be initiated immediately.

In determining the required combination of boration flow rate and concentration, no unique Design Basis Event must be satisfied. The only requirement is to restore the boron concentration to its required value as soon as possible. In order to raise the boron concentration as soon as possible, the operator should begin boration with the best source available for unit conditions.

Once actions have been initiated, they must be continued until the boron concentration is restored. The restoration time depends on the amount of boron that must be injected to reach the required concentration.

SURVEILLANCE REQUIREMENTS

SR 3.9.1.1

This SR ensures that the coolant boron concentration in the RCS, the refueling canal, and the refueling cavity is within the COLR limits. The boron concentration is determined periodically by chemical analysis of a representative sample of the interconnected volumes. B

A minimum Frequency of once every 72 hours is a reasonable amount of time to verify the boron concentration of representative samples. The Frequency is based on operating experience, which has shown 72 hours to be adequate.

REFERENCES

1. FSAR. Sections 1.3.5, 3.1, and 9.3.
2. FSAR. Chapter 14.1.4.

Description of Changes - NUREG-1431 Section 3.09.04

30-Jun-00

DOC Number	DOC Text	
A.01 Rev. A	In the conversion of Point Beach current Technical Specifications (CTS) to the proposed plant specific Improved Technical Specifications (ITS), certain wording preferences or conventions are adopted which do not result in technical changes (either actual or interpretational). Editorial changes, reformatting, and revised numbering are adopted to make the ITS consistent with the Standard Technical Specifications, Westinghouse Plants, NUREG-1431, Revision 1 (i.e., Improved Standard Technical Specifications (ISTS)).	
	CTS:	**ITS:**
	15.03.08.01	LCO 3.09.03
A.02 Rev. B	Not used.	
	CTS:	**ITS:**
	N/A	N/A
A.03 Rev. A	CTS 15.3.8.7 requires the Containment Purge and Vent System be operable. Proposed ITS 3.9.3.c requires Containment Purge and Exhaust System penetrations providing direct access from the containment atmosphere to the outside atmosphere be either: 1. closed by a manual or automatic isolation valve, blind flange, or equivalent, or 2. capable of being closed by an OPERABLE Containment Purge and Exhaust Isolation System. Proposed ITS LCO 3.9.3.c requires Containment Purge and Exhaust penetrations that are not capable of being closed by an OPERABLE isolation system to be isolated. This is consistent with CTS 15.3.8.8, which requires the closure of the Containment Purge and Exhaust System penetrations, if the Containment Purge and Exhaust System is inoperable. Therefore, CTS 15.3.8.7 and ITS 3.9.3.c both allow continued refueling operations with the isolation of any required penetrations that are inoperable.	
	CTS:	**ITS:**
	15.03.08.07	LCO 3.09.03 C LCO 3.09.03 C.1 LCO 3.09.03 C.2
L.01 Rev. A	CTS 15.3.8.7 requires the Containment Purge and Vent System be demonstrated operable within 4 days prior to the start of and at least once per 7 days during refueling operations by verifying that Containment Purge and Vent isolation occurs on manual initiation and on high radiation test signal. Proposed ITS SR 3.9.3.2 requires verification of each containment purge and exhaust valve actuates to the isolation position on an actual or simulated actuation signal once per 18 months. Adopting a less restrictive frequency for verification that each containment purge and exhaust valve actuates to the isolation position on an actual or simulated actuation signal is acceptable. The frequency of 18 months for SR 3.9.3.2 is consistent with other similar valve actuation tests.	
	CTS:	**ITS:**
	15.03.08.07 SR.02	SR 3.09.03.02

DOC Number	DOC Text
L.02 Rev. A	CTS 15.3.8.9 specifies that in the event the limiting condition for the equipment hatch and personnel locks is not met, refueling of the reactor shall cease. Additionally, work shall be initiated to correct the violated condition so that the specified limit is met, and no operations which may increase the reactivity of the core shall be made. Proposed ITS 3.9.3, Condition A, Required Actions A.1 and A.2 specify to immediately suspend Core Alterations and the movement of irradiated fuel assemblies within containment. This is a relaxation of requirements and is less restrictive. However, this change is acceptable since performing ITS 3.9.3 Required Actions A.1 and A.2 places the plant in a condition whereby LCO 3.9.3 is no longer applicable.

CTS:	ITS:
15.03.08.09	LCO 3.09.03 COND A
	LCO 3.09.03 COND A RA A.1
	LCO 3.09.03 COND A RA A.2

DOC Number	DOC Text
L.03 Rev. A	CTS 15.3.8.1 requires the personnel locks be capable of being closed during refueling operations. CTS 15.3.8.1 also requires a temporary third door on the outside of the personnel lock to be in place whenever both doors in a personnel lock are open (except for initial core loading.) ITS LCO 3.9.3.b requires one door in each airlock to be capable of being closed, during CORE ALTERATIONS, and during the movement of irradiated fuel assemblies within containment. This is consistent with NUREG 1431, LCO 3.9.4. The allowance to have containment personnel airlocks open during fuel movements and CORE ALTERATIONS is based on the Point Beach confirmatory dose calculation of a fuel handling accident. This calculation assumes a ground level release with acceptable radiological consequences. The personnel airlocks are not assumed to be closed during the fuel handling accident, nor are the airlocks assumed to be closed within any amount of time following the fuel handling accident. Although this change results in a relaxation of the current requirements, it is acceptable. Adopting the requirements of NUREG 1431 does not result in a significant reduction in the margin of safety, because the closure of the personnel airlock doors is not assumed to mitigate the radiological consequences of the fuel handling accident.

CTS:	ITS:
15.03.08.01	LCO 3.09.03 B
	N/A

Description of Changes - NUREG-1431 Section 3.09.04

10-Jul-00

DOC Number	DOC Text
L.04 Rev. B	CTS 15.3.8.1 and 15.3.8.7 establish requirements for the closure of containment penetrations during refueling operations. The requirements for containment penetration closure are conveyed in proposed ITS LCO 3.9.3, which are applicable during Core Alterations and during movement of irradiated fuel assemblies within containment. The CTS definition of Refueling Operation is any operation that involves the movement of core components that could affect the reactivity of the core within the containment when the vessel head is removed. Core components which could affect the reactivity are considered to be control rods and fuel assemblies. The ITS definition of Core Alterations is "the movement of any fuel, sources, or reactivity control components, within the reactor vessel with the vessel head removed and fuel in the vessel." Since proposed ITS LCO 3.9.3 applicability also includes the movement of irradiated fuel inside the containment, the combination of the defined term and specified applicability is equivalent to the CTS 15.3.8.1 and 15.3.8.7 applicabilities, with the exception of the movement of components other than irradiated fuel within containment. Although this change results in a relaxation of the current requirements, it is acceptable. The requirements for containment penetration closure are based on the Point Beach confirmatory dose calculation of a fuel handling accident. This calculation assumes a ground level release with acceptable radiological consequences. The containment penetrations are not assumed to be closed during a fuel handling accident, nor are the containment penetrations assumed to be closed within any amount of time following the fuel handling accident. Therefore, adopting the requirements of NUREG 1431 does not result in a significant reduction in the margin of safety, because the closure of containment penetrations is not assumed to mitigate the radiological consequences of the fuel handling accident. **CTS:** 15.03.08.01 — **ITS:** LCO 3.09.03
M.01 Rev. A	CTS 15.3.8.7 is revised to adopt ITS SR 3.9.3.1, which requires a weekly verification that each required containment penetration is in the required status. This surveillance demonstrates that each of the containment purge and exhaust system penetrations that are not capable of being closed by an OPERABLE Containment Purge and Exhaust Isolation System is isolated. Since this change imposes new requirements, it is more restrictive and has no adverse impact on safety. **CTS:** 15.03.08.07 SR.01 — **ITS:** SR 3.09.03.01

DOC Number	DOC Text
M.02 Rev. A	CTS 15.3.8.8 requires that if the Containment Purge and Vent System is inoperable, the Purge and Vent containment penetrations shall be closed, with no further actions specified if this condition can not be met. This allows the continuation of refueling operations, with or without the isolation of the containment penetrations. Proposed ITS LCO 3.9.3 is met if the Containment Purge and Vent penetrations are isolated or capable of being isolated. ITS 3.9.3, Condition A, is entered when one or more containment penetrations are not in the required status. ITS 3.9.3, Action A.1, requires Core Alterations to be suspended and Action A.2 requires the movement of irradiated fuel assemblies to be suspended. This change is more restrictive in that it introduces additional requirements on plant operation.

CTS:	ITS:
15.03.08.08	LCO 3.09.03 COND A
	LCO 3.09.03 COND A RA A.1
	LCO 3.09.03 COND A RA A.2

DOC Number	DOC Text
M.03 Rev. A	CTS 15.3.8.1 states, "The equipment hatch shall be closed." ITS 3.9.3.a specifies the equipment hatch closure with "...held in place with all bolts." Specifying the equipment hatch be held in place with all bolts places additional requirements on unit operation and is therefore more restrictive. This change is necessary to ensure the equipment hatch will be sufficiently secured in place to minimize the escape of fission product radioactivity to the environment that may be released from the reactor core following an accident.

CTS:	ITS:
15.03.08.01	LCO 3.09.03 A

A.1

15.3.8 REFUELING

Applicability:

Applies to operating limitations during refueling operations.

Objective:

To ensure that no incident could occur during refueling operations that would affect public health and safety.

Specifications :

L.3
L.4
M.3

B

Replace with Insert 3.9.4-1, LCO 3.9.3

During refueling operations :

1. The equipment hatch shall be closed and the personnel locks shall be capable of being closed. ~~A temporary third door on the outside of the personnel lock shall be in place whenever both doors in a personnel lock are open (except for initial core loading).~~ L.3

2. Radiation levels in fuel handling areas, the containment and spent fuel storage pool shall be monitored continuously. < See 3.9.1 >

< See 3.9.3 >

3. Core subcritical neutron flux shall be continuously monitored by at least two neutron monitors, each with continuous visual indication in the control room and one with audible indication in the containment available whenever core geometry is being changed. When core geometry is not being changed, at least one neutron flux monitor shall be in service.

< See 3.9.5 >

4. At least one residual heat removal loop shall be in operation. However, if refueling operations are affected by the residual heat removal loop flow, the operating residual heat removal loop may be removed from operation for up to one hour per eight hour period.

5. During reactor vessel head removal and while loading and unloading fuel from the reactor, a minimum boron concentration of 2100 ppm* shall be maintained in the primary coolant system.

* This boron concentration value is in effect following U1R25 for Unit 1 and following U2R23 for Unit 2; and takes effect prior to loading fuel for those outages. Prior to U1R25, the Unit 1 boron concentration value of this specification is 1800 ppm. Prior to U2R23, the Unit 2 boron concentration value of this specification is 1800 ppm. < See 3.9.1 >

A.1

Insert 3.9.4-1

M.3

LCO 3.9.3 The containment penetrations shall be in the following status:

a. The equipment hatch closed and held in place with all bolts;

L.3 b. One door in each air lock is capable of being closed;

c. Each Containment Purge and Exhaust System pe netration either:

1. closed by a manual or automatic isolation valve, blind flange, or equivalent, or

A.3 2. capable of being closed by an OPERABLE Containment Purge and Exhaust Isolation System.

APPLICABILITY: During CORE ALTERATIONS,
During movement of irradiated fuel assemblies within containment.

L.4 B

Insert 3.9.4-2

SR 3.9.3.2	------------------NOTE------------------ Not applicable to containment purge and exhaust valve(s) in penetrations closed to comply with LCO 3.9.3.c.1. --------------------------------------- Verify each required containment purge and exhaust valve actuates to the isolation position on an actual or simulated actuation signal.	18 months

L.1

Justification For Deviations - NUREG-1431 Section 3.09.04

21-Jun-00

JFD Number	JFD Text	
08 Rev. A	The containment equipment hatch at Point Beach is required to be held in place with all bolts in order to effect an adequate seal. As a result, "good engineering practice" to ensure the in-place bolts are equally spaced, is not an issue and can be deleted from the bases.	
	ITS:	**NUREG:**
	B 3.09.03	B 3.09.04
	LCO 3.09.03 A	LCO 3.09.04 A
09 Rev. B	The Bases have been modified by the addition of a statement that provides basis for allowing containment personnel airlocks to remain open during fuel movements and core alterations. Point Beach confirmatory dose calculations for a fuel handling accident assume a ground level release with acceptable radiological consequences.	
	ITS:	**NUREG:**
	B 3.09.03	B 3.09.04

1 → 3

LCO (continued)

Approved TSTF 68

Insert B3.9.3-1.

9 → Insert B3.9.3-2.

closure ~~times~~ specified in the FSAR can be achieved ~~and, therefore, meet the assumptions used in the safety analysis to ensure that releases through the valves are terminated, such that radiological doses are within the acceptance limit~~.

2

B RAI 3.9.3-1

APPLICABILITY

The containment penetration requirements are applicable during CORE ALTERATIONS or movement of irradiated fuel assemblies within containment because this is when there is a potential for a fuel handling accident. In MODES 1, 2, 3, and 4, containment penetration requirements are addressed by LCO 3.6.1. In MODES 5 and 6, when CORE ALTERATIONS or movement of irradiated fuel assemblies within containment are not being conducted, the potential for a fuel handling accident does not exist. Therefore, under these conditions no requirements are placed on containment penetration status.

ACTIONS

A.1 and A.2

3

Purpose and Exhaust System

3

If the containment equipment hatch, air locks, or any containment penetration ~~that provides direct access from the containment atmosphere to the outside atmosphere~~ is not in the required status, including the Containment Purge and Exhaust Isolation System not capable of automatic actuation when the purge and exhaust valves are open, the unit must be placed in a condition where the isolation function is not needed. This is accomplished by immediately suspending CORE ALTERATIONS and movement of irradiated fuel assemblies within containment. Performance of these actions shall not preclude completion of movement of a component to a safe position.

SURVEILLANCE REQUIREMENTS

SR 3.9.4.1

3 ← 1

This Surveillance demonstrates that each of the containment penetrations required to be in its closed position is in that position. The Surveillance on the open purge and exhaust valves will demonstrate that the valves are not blocked from closing. Also the Surveillance will

1 → 3

REFERENCES

~~1. GPU Nuclear Safety Evaluation SE-0002000-001, Rev. 0, May 20, 1988.~~ 3

3 1 2 FSAR, Section [15.4.5] 14.2.1 7

2 3 NUREG-0800, Section 15.7.4, Rev. 1, July 1981.

Insert B3.9.3-1:

The containment personnel airlock doors may be open during movement of irradiated fuel in the containment and during CORE ALTERATIONS provided that one door is capable of being closed in the event of a fuel handling accident. Should a fuel handling accident occur inside containment, one personnel airlock door will be closed following an evacuation of containment.

Insert B3.9.3-2:

The allowance to have containment personnel airlocks open during fuel movements and CORE ALTERATIONS is based on the Point Beach confirmatory dose calculation of a fuel handling accident. This calculation assumes a ground level release with acceptable radiological consequences. The personnel airlocks are not assumed to be closed during the fuel handling accident, nor are the airlocks assumed to be closed within any amount of time following the fuel handling accident. B RAI 3.9.3-1

No Significant Hazards Considerations - NUREG-1431 Section 3.09.04

30-Jun-00

NSHC Number	NSHC Text
A Rev. A	In accordance with the criteria set forth in 10 CFR 50.92, PBNP has evaluated this proposed Technical Specifications change and determined it does not represent a significant hazards consideration. The following is provided in support of this conclusion. 1. Does the change involve a significant increase in the probability or consequences of an accident previously evaluated? The proposed change involves reformatting and rewording of the current Technical Specifications. The reformatting and rewording process involves no technical changes to existing requirements. As such, this change is administrative in nature and does not impact initiators of analyzed events or assumed mitigation of accident or transient events. Therefore, this change does not increase the probability or consequences of an accident previously evaluated. 2. Does the change create the possibility of a new or different kind of accident from any accident previously evaluated? The proposed change does not require a physical alteration of the plant (no new or different type of equipment will be installed) or changes in parameters governing normal plant operation. The proposed change will not impose any new or eliminate any old requirements. Thus, this change does not create the possibility of a new or different kind of accident from any accident previously evaluated. 3. Does this change involve a significant reduction in a margin of safety? The proposed change will not significantly reduce the margin of safety because it has no impact on any safety analysis assumptions. This change is administrative. As such, there is no technical change to the requirements and, therefore, there is no reduction in the margin of safety.

30-Jun-00

NSHC Number	NSHC Text
L.01 Rev. A	In accordance with the criteria set forth in 10 CFR 50.92, PBNP has evaluated this proposed Technical Specifications change and determined it does not represent a significant hazards consideration. The following is provided in support of this conclusion. 1. Does the change involve a significant increase in the probability or consequences of an accident previously evaluated? 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 adopts a less restrictive frequency for verifying containment purge and exhaust valves actuate to the isolation position on an actual or simulated actuation signal. The logic associated with this function is adequately tested per LCO 3.3.6, and the proposed frequency of 18 months is consistent with other similar valve actuation tests. Therefore, this change does not involve a significant increase in the probability or consequences of an accident previously evaluated. 2. Does the change create the possibility of a new or different kind of accident from any accident previously evaluated? The proposed change does not 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 refueling are properly maintained. Therefore, this 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. This change limits the actions required in the event the equipment hatch and/or personnel locks are not in the required status, as specified in ITS LCO 3.9.3. However, this change is acceptable since performing ITS 3.9.3 Required Actions A.1 and A.2 places the plant in a condition whereby LCO 3.9.3 is no longer applicable. 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 refueling are properly maintained. Therefore, this change does not involve a 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. CTS 15.3.8.1 requires the personnel locks be capable of being closed during refueling operations. CTS 15.3.8.1 also requires a temporary third door on the outside of the personnel lock to be in place whenever both doors in a personnel lock are open (except for initial core loading.) Proposed ITS LCO 3.9.3.b requires one door in each airlock to be capable of being closed, during CORE ALTERATIONS, and during the movement of irradiated fuel assemblies within containment, consistent with NUREG 1431, LCO 3.9.4 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 relaxes the requirement of personnel airlock doors during refueling operations. This change is acceptable, because the allowance to have containment personnel airlocks open during fuel movements and CORE ALTERATIONS is based on the Point Beach confirmatory dose calculation of a fuel handling accident. This calculation assumes a ground level release with acceptable radiological consequences. The personnel airlocks are not assumed to be closed during the fuel handling accident, nor are the airlocks assumed to be closed within any amount of time following the fuel handling accident. Therefore, this change does not involve a significant increase in the probability or consequences of an accident previously evaluated. 2. Does the change create the possibility of a new or different kind of accident from any accident previously evaluated? The proposed change does not 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, because the allowance to have containment personnel airlocks open during fuel movements and CORE ALTERATIONS is based on the Point Beach confirmatory dose calculation of a fuel handling accident. The requirements will continue to assure that limiting conditions for refueling are properly maintained. Therefore, this change does not involve a significant reduction in a margin of safety.

NSHC Number	NSHC Text
L.04 Rev. B	In accordance with the criteria set forth in 10 CFR 50.92, PBNP has evaluated this proposed Technical Specifications change and determined it does not represent a significant hazards consideration. The following is provided in support of this conclusion. The CTS requires containment penetration closure during refueling operations. Proposed ITS requirements for containment penetration closure are applicable during Core Alterations and during movement of irradiated fuel assemblies within containment. The CTS definition of Refueling Operation is any operation that involves the movement of core components that could affect the reactivity of the core within the containment when the vessel head is removed. Core components which could affect the reactivity are considered to be control rods and fuel assemblies. The ITS definition of Core Alterations is "the movement of any fuel, sources, or reactivity control components, within the reactor vessel with the vessel head removed and fuel in the vessel." Since the proposed ITS applicability also includes the movement of irradiated fuel inside the containment, the combination of the defined term and specified applicability is equivalent to the CTS applicability, with the exception of the movement of components other than irradiated fuel within containment. 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 relaxes the requirement for closure of containment penetrations during refueling operations. This change is acceptable, because the requirement for closure of containment penetrations is based on the Point Beach confirmatory dose calculation of a fuel handling accident. This calculation assumes a ground level release with acceptable radiological consequences. The containment penetrations are not assumed to be closed during the fuel handling accident, nor are they assumed to be closed within any amount of time following the fuel handling accident. Therefore, this change does not involve a significant increase in the probability or consequences of an accident previously evaluated. 2. Does the change create the possibility of a new or different kind of accident from any accident previously evaluated? The proposed change does not 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, because the requirement for containment penetration closure is based on the Point Beach confirmatory dose calculation of a fuel handling accident. The requirements will continue to assure that limiting conditions for refueling are properly maintained. Therefore, this change does not involve a significant reduction in a 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 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.

B 3.9 REFUELING OPERATIONS

B 3.9.3 Containment Penetrations

BASES

BACKGROUND

During CORE ALTERATIONS or movement of irradiated fuel assemblies within containment, a release of fission product radioactivity within containment will be restricted from escaping to the environment when the LCO requirements are met. In MODES 1, 2, 3, and 4, this is accomplished by maintaining containment OPERABLE as described in LCO 3.6.1, "Containment." In MODE 6, the potential for containment pressurization as a result of an accident is not likely; therefore, requirements to isolate the containment from the outside atmosphere can be less stringent. Since there is no potential for containment pressurization, the Appendix J leakage criteria and tests are not required.

The containment serves to minimize the escape of fission product radioactivity to the environment that may be released from the reactor core following an accident, such that offsite radiation exposures are maintained well within the requirements of 10 CFR 100. Additionally, the containment provides radiation shielding from the fission products that may be present in the containment atmosphere following accident conditions.

The containment equipment hatch, which is part of the containment pressure boundary, provides a means for moving large equipment and components into and out of containment. During CORE ALTERATIONS or movement of irradiated fuel assemblies within containment, the equipment hatch must be held in place with all bolts.

The containment air locks, which are also part of the containment pressure boundary, provide a means for personnel access during MODES 1, 2, 3, and 4 unit operation in accordance with LCO 3.6.2, "Containment Air Locks." Each air lock has a door at both ends. The doors are normally interlocked to prevent simultaneous opening when containment OPERABILITY is required. During periods of unit shutdown when containment closure is not required, the door interlock mechanism may be disabled, allowing both doors of an air lock to remain open for extended periods when frequent containment entry is necessary. During CORE ALTERATIONS or movement of irradiated fuel assemblies within containment, one airlock door must always remain capable of being closed.

The requirements for containment purge and exhaust system penetration closure ensure that a release of fission product radioactivity within containment will be restricted to within regulatory limits.

BASES

BACKGROUND (continued)	The Containment Purge and Exhaust System includes a 36 inch purge penetration and a 36 inch exhaust penetration. During MODES 1, 2, 3, and 4, the two valves in each of the purge and exhaust penetrations are secured in the closed position. The Containment Purge and Exhaust System is not subject to a Specification in MODE 5. In MODE 6, large air exchanges are necessary to conduct refueling operations. The 36 inch purge system is used for this purpose, and all four valves are closed by the Containment Purge and Exhaust Isolation Instrumentation.
APPLICABLE SAFETY ANALYSES	During CORE ALTERATIONS or movement of irradiated fuel assemblies within containment, the most severe radiological consequences result from a fuel handling accident. The fuel handling accident is a postulated event that involves damage to irradiated fuel (Ref. 1). Fuel handling accidents, analyzed in Reference 2, include dropping a single irradiated fuel assembly and handling tool or a heavy object onto other irradiated fuel assemblies. The requirements of LCO 3.9.6, "Refueling Cavity Water Level," and the minimum decay time of 100 hours prior to CORE ALTERATIONS ensure that the release of fission product radioactivity subsequent to a fuel handling accident, results in doses that are well within the guideline values specified in 10 CFR 100. Standard Review Plan, Section 15.7.4, Rev. 1 (Ref. 2), defines "well within" 10 CFR 100 to be 25% or less of the 10 CFR 100 values. The acceptance limits for offsite radiation exposure will be 25% of 10 CFR 100 values or the NRC staff approved licensing basis (e.g., a specified fraction of 10 CFR 100 limits). Containment penetrations satisfy Criterion 3 of the NRC Policy Statement.
LCO	This LCO limits the consequences of a fuel handling accident in containment by limiting the potential escape paths for fission product radioactivity released within containment. The LCO requires any Containment Purge and Exhaust System penetration to be closed except for the OPERABLE containment purge and exhaust penetrations. For the OPERABLE containment purge and exhaust penetrations, this LCO ensures that these penetrations are isolable by the Containment Purge and Exhaust Isolation System. The OPERABILITY requirements for this LCO ensure that the automatic purge and exhaust valve closure specified in the FSAR can be achieved.

BASES

LCO (continued)

The containment personnel airlock doors may be open during movement of irradiated fuel in the containment and during CORE ALTERATIONS provided that one door is capable of being closed in the event of a fuel handling accident. Should a fuel handling accident occur inside containment, one personnel airlock door will be closed following an evacuation of containment.

The allowance to have containment personnel airlocks open during fuel movements and CORE ALTERATIONS is based on the Point Beach confirmatory dose calculation of a fuel handling accident. This calculation assumes a ground level release with acceptable radiological consequences. The personnel airlocks are not assumed to be closed during the fuel handling accident, nor are the airlocks assumed to be closed within any amount of time following the fuel handling accident. B RAI 3.9.3-1

APPLICABILITY

The containment penetration requirements are applicable during CORE ALTERATIONS or movement of irradiated fuel assemblies within containment because this is when there is a potential for a fuel handling accident. In MODES 1, 2, 3, and 4, containment penetration requirements are addressed by LCO 3.6.1. In MODES 5 and 6, when CORE ALTERATIONS or movement of irradiated fuel assemblies within containment are not being conducted, the potential for a fuel handling accident does not exist. Therefore, under these conditions no requirements are placed on containment penetration status.

ACTIONS

A.1 and A.2

If the containment equipment hatch, air locks, or any containment Purge and Exhaust System penetration is not in the required status, including the Containment Purge and Exhaust Isolation System not capable of automatic actuation when the purge and exhaust valves are open, the unit must be placed in a condition where the isolation function is not needed. This is accomplished by immediately suspending CORE ALTERATIONS and movement of irradiated fuel assemblies within containment. Performance of these actions shall not preclude completion of movement of a component to a safe position.

SURVEILLANCE REQUIREMENTS

SR 3.9.3.1

This Surveillance demonstrates that each of the containment penetrations required to be in its closed position is in that position. The Surveillance on the open purge and exhaust valves will demonstrate that the valves are not blocked from closing. Also the Surveillance will demonstrate that each valve operator has motive power, which will

BASES

SURVEILLANCE REQUIREMENTS (continued)

ensure that each valve is capable of being closed by an OPERABLE automatic containment purge and exhaust isolation signal.

The Surveillance is performed every 7 days during CORE ALTERATIONS or movement of irradiated fuel assemblies within containment. The Surveillance interval is selected to be commensurate with the normal duration of time to complete fuel handling operations. A surveillance before the start of refueling operations will provide two or three surveillance verifications during the applicable period for this LCO.

SR 3.9.3.2

This Surveillance demonstrates that each containment purge and exhaust valve actuates to its isolation position on manual initiation or on an actual or simulated high radiation signal. The 18 month Frequency maintains consistency with other similar ESFAS instrumentation and valve testing requirements. SR 3.6.3.5 demonstrates that the isolation time of each valve is in accordance with the Inservice Testing Program requirements. These Surveillances performed during MODE 6 will ensure that the valves are capable of closing after a postulated fuel handling accident to limit a release of fission product radioactivity from the containment.

The SR is modified by a Note stating that this demonstration is not applicable to valves in isolated penetrations. LCO 3.9.3.c.1 provides the option to close penetrations in lieu of requiring automatic isolation capability.

REFERENCES

1. FSAR. Section 14.2.1.
2. NUREG-0800, Section 15.7.4, Rev. 1, July 1981.