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NPL 2000-0465

October 19, 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 7 TO APPLICATION FOR AMENDMENT TO
FACILITY OPERATING LICENSE APPENDIX A:
TECHNICAL SPECIFICATIONS IMPROVEMENT PROJECT
SUPPLEMENTAL RESPONSE TO RAI ON ITS SECTION 3.6;
RESPONSE TO RAI ON ITS SECTIONS 3.7.4 and 3.7.5
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 April 19, 2000, the NRC issued a Request for Additional Information (RAI) to WE on ITS section 3.6. The WE response was provided June 19, 2000 (reference letter NPL 2000-0271). During a telephone conference with Point Beach personnel on September 21, 2000, NRC Staff requested clarifying information on portions of the WE response. In letters dated August 24 and September 8, 2000, the NRC issued RAIs to Nuclear Management Company on ITS sections 3.7.4 and 3.7.5 respectively.

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Attachment 1 of this letter includes our response to the Staff's questions in the above referenced telephone conference and in the August 24 and September 8 letters. 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. C."

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

C
RAI 3.6.4-2

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

We have 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, We conclude 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.

Nuclear Management Company (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 2000-0260)
- Supplement 3 dated June 19, 2000 (ITS section 3.6; reference letter NPL 2000-0271)
- Supplement 4 dated July 28, 2000 (ITS section 3.8; reference letter NPL 2000-0341)
- Supplement 5 dated August 17, 2000 (ITS sections 3.4 and 3.9; reference letter NPL 2000-0371)
- Supplement 6 dated September 14, 2000 (ITS section 5.5; reference letter NPL 2000-0411)

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

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

Sincerely,

Mark Reddemann
Site Vice President
Point Beach Nuclear Plant

Subscribed to and sworn before me
on this 20th day of October, 2000

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
SUPPLEMENTAL RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION
TECHNICAL SPECIFICATIONS IMPROVEMENT PROJECT SECTION 3.6, 3.7.4 and 3.7.5
POINT BEACH NUCLEAR PLANT UNITS 1 AND 2

SECTION 3.6

The following information is provided in response to the Nuclear Regulatory Commission staff's requests for clarifying information during a telephone conference on September 21, 2000 (regarding an April 19, 2000 RAI).

NRC Question 3.6.1-1:

CTS 1.D defines CONTAINMENT INTEGRITY. A markup of CTS 1.D shows that the requirements of CTS 1.D.1, 1.D.3 and a portion of 1.D.4 are relocated to ITS 3.6.2 and 3.6.3 by DOCs A.1. and A.4. The rest of CTS 1.D is incorporated into ITS LCO 3.6.1 and SR 3.6.1.1, and is covered by DOCs A.3 and A.4. While these changes are acceptable with regards to the Administrative changes made to CTS 1.D, the changes made to CTS 1.D are incomplete. The definition is relocated in its entirety to ITS B3.6.1 Bases BACKGROUND which makes this portion of the change a Less Restrictive (LA) change. See Comment Number 3.6.1.-2. Comment: Revise the CTS markup of CTS 1.D and provide a discussion and justification for this Less Restrictive (LA) change.

Response:

The CTS markup of CTS 1.D has been revised as requested. DOC LA.01 and associated NSHC have been created to justify this change

The entire CTS definition of CONTAINMENT INTEGRITY has been properly accounted for, as follows:

D. Containment Integrity*
Containment integrity is defined to exist when:

This title, footnote, and statement information are introductory and convey no requirements. Therefore, the title and statement information are covered by the A.1 DOCs in sections 3.6.2 and 3.6.3 and DOC A.3 of Section 3.6.1 of the submittal. The footnote is covered under the A.2 DOC in section 3.6.1.

1) Penetrations required to be isolated during accident conditions are either:
   a. Capable of being closed by an operable automatic containment isolation valve,
      OR
   b. Closed by an operable containment isolation valve,
      OR

c. Closed in accordance with Specifications 15.3.6.A.1.b and 15.3.6.A.1.c.

These requirements are properly covered by DOCs A.1 and A.2 in section 3.6.3.

2) The equipment hatch is properly closed.

This requirement is properly covered by DOC A.4 in section 3.6.1 and DOC A.2 in section 3.6.3.

3) At least one door in each personnel air lock is properly closed.

This requirement is properly covered by DOC A.2 in section 3.6.2.

4) The overall uncontrolled containment leakage is less than La.**

This requirement is properly covered by DOC A.4 in section 3.6.1.

Therefore, the entire definition has been appropriately covered by the applicable DOCs and the associated requirements are contained in the ITS. The definition in its entirety is also contained in the basis where it expounds on the associated requirements.

NRC Question 3.6.1-5:

CTS 1D.4, 3.6.E and 4.4.1 require leak rate testing in accordance with the Containment Leak Rate Testing Program which is based on the requirements of 10 CFR 50 Appendix J, Option B. STS SR 3.6.1.1 requires the visual examination and leakage rate testing be performed in accordance with 10 CFR 50 Appendix J as modified by approved exemptions. ITS SR 3.6.1.1 modifies STS SR 3.6.1.1 to conform to CTS 1.D.4, 3.6.E and 4.4.1 as modified in the CTS markup. The STS is based on Appendix J, Option A while the CTS and ITS are based on Appendix J, Option B. Changes to the STS with regards to Option A versus Option B are covered by a letter from Mr. Christopher I. Grimes to Mr. David J. Modeen, NEI, dated 11/2/95 and TSTF - 52, as modified by staff comments of 10/96, 12/98, and1/2000. The changes to ITS 3.6.1., 3.6.2, and their associated Bases are not in conformance with the letter and TSTF-52 as modified by staff comments. See Comment Numbers 3.6.1-6, 3.6.1-7, 3.6.2-1, and 3.6.2-2. Comment: Licensee should revise its submittal to conform to the 11/2/95 letter and TSTF-52 modified by the staff. See Comment Numbers 3.6.1-6, 3.6.1-7, 3.6.2-1 and 3.6.2-2.

Response:

The following changes were made to the submittal to conform to TSTF-52, Revision 3.

SR 3.6.1.1 (the exception to containment air lock testing was retained per the TSTF; the surveillance is covered in SR 3.6.2.1)

BASES - LCO 3.6.1 (wording discrepancies were corrected and the acceptance criteria of 1.0 L_a was stated)

BASES - SR 3.6.1.1 (references were deleted regarding exceptions to testing following an outage or shutdown during which Type A, B or C testing was performed)

BASES - APPLICABLE SAFETY ANALYSES (statement added specifying that L_a = 0.4% of containment air weight per day)

BASES - LCO (wording discrepancies were corrected)

BASES - SURVEILLANCE REQUIREMENTS (wording discrepancies were corrected)

In support of these changes, it should be noted that the current licensing basis for PBNP is based on a definition of P_a that differs from 10 CFR 50 Appendix J. Appendix J defines P_a as the calculated peak containment internal pressure. The PBNP current licensing basis defines P_a as the containment design pressure. Therefore, P_a is conservatively established at 60 psig for PBNP, which is about 7 psig greater than the approximately 53 psig peak pressure shown in the PBNP FSAR in section 14.3.4.

NRC Question 3.6.1-7:

SR 3.6.1.2 was deleted; however, this change relates to TSTF-343, which has not been approved by the NRC.

Response:

SR 3.6.1.2 and its associated reference were reinstated. As a result, JFD 2 has been revised in support of this change and DOC LB.01 has been marked as not used.

NRC Question 3.6.3-5:

One instance of a statement in the ITS B3.6.3 Bases Section refers to "non-essential penetrations" and the containment isolation valves associated with them.
Comment: Revise the ITS Bases to remove the terminology or implication that the specification only applies to "non-essential penetrations".

Response:

The reference to "non-essential" penetrations has been deleted from ITS B3.6.3 BASES – APPLICABLE SAFETY ANALYSES.

NRC Question 3.6.3-12:

STS 3.6.3 ACTION C specifies the required ACTIONS to be taken for an inoperable containment isolation valve in a penetration flow path with only one containment isolation valve and a closed system. STS 3.6.3 ACTION C has been modified by TSTF 30 Rev.2 to extend the Completion Time from 4 hours to 72 hours. This modification in the CTS and ITS is in accordance with TSTF 30 which is acceptable. However, the Bases changes are not in accordance with TSTF-30 Rev.2.
Comment: Licensee to update submittal to be in accordance with TSTF - 30 Rev.2 or provide additional justification for the deviations.

Response:

A statement providing the applicable Point Beach reference for a closed system has been added to the Bases for 3.6.3 in accordance with TSTF-30 Rev.2.

NRC Question 3.6.4-1:

The CTS markup of CTS 3.6.B.2 is modified to add the ITS 3.6.4 APPLICABILITY of MODES 1,2,3 and 4. This change is justified by DOC A.2 on the basis that the actions of CTS 3.6.B.2.b require the plant to be placed in COLD SHUTDOWN if the containment pressure cannot be maintained. If this were the only factor (Action statement) to take into consideration for this change, the justification probably would have been considered acceptable. However, the APPLICABILITY for containment pressure is controlled in the CTS by CTS 3.6 APPLICABILITY, 3.6 OBJECTIVE, 3.6.A.1 and Table 15.4.1-1 Item 27. The combination of CTS 3.6 APPLICABILITY, 3.6 OBJECTIVE and 3.6.A.1 would imply that the APPLICABILITY for internal pressure would be all plant conditions except the COLD SHUTDOWN and REFUELING SHUTDOWN conditions. However, CTS Table 15.4.1-1 Item 27 requires that the internal pressure requirement is applicable in "ALL" conditions. The change associated with the applicability change to CTS Table 15.4.1-1 (DOC A.5) uses the DOC A.2 justification as its basis. It should be noted that there are a number of specifications in the old and new STS which require a shutdown to COLD SHUTDOWN, but whose APPLICABILITY extends beyond COLD SHUTDOWN, e.g., Control Room Emergency Ventilation System. Based on the above discussion and the CTS, the Staff concludes that the CTS APPLICABILITY for containment pressure is all MODES/Conditions. Thus the changes (DOC A.2 and A.5) to the ITS APPLICABILITY are More Restrictive changes rather than Administrative changes.
Comment: Revise the CTS markup and provide a discussion and justification for these More Restrictive changes.

Response:

The CTS markups have been revised to recharacterize the changes (old DOC A.2 and A.5) as more restrictive (new DOC M.1 and M.2). A discussion and NSHC have been provided to justify this change.

NRC Question 3.6.6-3:

The markup of CTS 3.3.B.2.c specifies the remedial actions to be taken for inoperable containment spray and containment cooler valves. The corresponding action in the ITS for containment cooler valves is ITS 3.6.3 ACTION D. The addition of ITS 3.6.3 ACTION D is justified in the ITS by JFD 22. While the staff finds the addition of ITS 3.6.3 ACTION D acceptable, statements made in both the justification - JFD 22 and ITS B3.6.3 Bases - ACTION D are unacceptable. ITS B3.6.3 Bases - ACTION D states the following:

> "If the inoperable valve is capable of passing 100% of the assumed cooling water flow, but is inoperable due to loss of its ability to reposition within its assumed response time (e.g., loss of auto open capability, degraded stoke time, inoperable motor operator, etc;). SR 3.6.6.4 allows the inoperable valve to be secured in its required position (open). thereby eliminating the need for the valve to reposition upon receipt of an actuation signal. Securing the inoperable valve in its open position will result in exiting Condition D."

JFD 22 has similar wording. To start with the wrong ITS SR is referenced in the statements. ITS SR 3.6.6.4 deals with containment spray pumps, the correct SR would be ITS SR 3.6.6.5 which deals with containment spray valves and containment fan cooler service water outlet valves automatic operation. The intent of this SR is that it applies to those valves that during normal operating conditions are locked, sealed or otherwise secured in their normal operating position. Therefore the above Bases statement which states that SR 3.6.6.5 would allow the inoperable valve to be secured open is incorrect and not in accordance with the intent of the specification. Furthermore, locking the valve open does not restore the valve to OPERABLE status per the ACTION statement. The valve may be able to perform its safety function (pass water) but it is still considered inoperable; it cannot actuate when it receives an actuation signal. Thus, the statements are incorrect and do not meet the intent of the specifications.
Comment: Delete these sentences from the justification JFD 22 and ITS B3.6.6 Bases - ACTION D.

Response:

The sentences pertaining to restoration of operability of these valves were deleted from JFD 22 and the proposed Bases for ITS B3.6.6 - ACTION D in our original response to this RAI. However, we inadvertently stated that this was deleted from the proposed ITS Bases for SR 3.6.6.5 vice the ITS B3.6.6 Bases - ACTION D. This corrects our initial response.

NRC Question 3.6.6-7:

CTS 4.5.1.B.1 requires a system test of the Containment Spray System and specifies that "Operation of the system is initiated by tripping the normal actuation instrumentation." The ITS breaks this CTS surveillance up into two surveillances - ITS SR 3.6.6.5 and SR 3.6.6.6, however the ITS tests may be initiated by either an actual or simulated actuation signal. The CTS markup does not show this change "normal actuation" to "actual or simulated actuation" but it does show that the statement is relocated (DOC LA.1). This is incorrect. "Tripping the normal actuation" connotes only a simulated actuation. By adding the words "actual actuation" the change becomes a Less Restrictive (L) change.
Comment: Revise the CTS markup and provide a discussion and justification for this Less Restrictive (L) change.

Response:

L.05 has been revised to describe and justify that this change allows either a simulated or an actual actuation signal. The associated NSHC remains appropriate as initially written.

NRC Question 3.6.6-9:

CTS 4.5.1.C.2 specifies that the containment fan cooler accident fans shall be tested monthly to verify OPERABILITY. It also specifies that the performance shall be acceptable if the fan starts and the running current is verified. The CTS/ITS markups show this requirement as being relocated to the Bases as Insert B3.6.6-14. See Comment Number 3.6.6-8 for concerns with regards to justifying the relocation. STS B3.6.6 Bases - SR 3.6.6A.2 states that the purpose of the SR is to ensure that all associated controls are functioning properly and that blockage, fan or motor failure or excessive vibration can be detected for corrective action. ITS B3.6.6 Bases - SR 3.6.6.2 deletes all mention of associated controls and the items to be detected for corrective action. The justification (JFD 27) for this deletion states that the containment fan coolers do not have any associated controls nor does it have any installed vibration monitoring equipment. With regards to the deletion of the associated controls aspect, the Insert states explicitly what the associated controls are - fan run indication, motor running amps, and low flow alarms. Thus the deletion of the words associated with the "controls" should not be deleted. With regards to detection of excessive vibration, the STS does not specify or require that vibration monitors be installed. The vibration monitors could be portable, it could be done through visual observation, or through other means.
Comment: Revise the ITS markup to retain the STS wording, or provide additional discussion and justification for its deletion. See Comment Number 3.6.6-8.

Response:

The ITS Bases was revised to retain more of the STS wording as requested by the reviewer. The word "controls" was replaced with "indications" because the Point Beach fan coolers only have status indicators (e.g., fan run indication, motor running amps, low flow alarms), which provide indication of fan cooler status. These indicators do not have any control function over the fan coolers. The only control is the start/stop switch, which is tested by virtue of starting and stopping the fans in order to conduct the testing.

JFD 27 incorrectly stated, "the containment accident fan cooler unit accident mode fans do not have installed vibration monitoring equipment." This statement has been deleted from JFD 27. However, the PBNP CTS does not currently require vibration monitoring. Specification CTS 15.4.5-I.C.2 states, "Acceptable performance shall be that the accident fan starts and running current is verified." Therefore, the proposed ITS Bases are consistent with the current licensing basis for the PBNP system.

NRC Question 3.6.7-2:

CTS 3.3.B.1.d states that "All valves and piping associated with the above components and required to function during accident conditions, are operable." The CTS markup shows this requirement as being deleted by DOC A.5. DOC A.5 justifies the deletion based on definition of OPERABILITY. This is incorrect. This statement is not deleted, but has been relocated to ITS B3.6.7 Bases - LCO and is part of the discussion in this ITS Section describing what constitutes an OPERABLE Spray Additive System. Therefore, the change is a Less Restrictive (LA) change rather than an Administrative change.
Comment: Revise the CTS markup and provide additional discussion and justification for this Less Restrictive (LA) change.

Response:

The submittal has been revised to reclassify old DOC A.5 as new DOC LA.1 at the reviewer's request. Although the requirement of OPERABILITY has been relocated to ITS LCO 3.6.7, the specifics defining OPERABILITY have been relocated to ITS B3.6.7 Bases – LCO. Additional discussion and justification has been provided for this LA change, including a NSHC.

NRC Question 3.6.7-5:

CTS 4.5.1.B.1 requires a system test of the Spray Additive System and specifies that "Operation of the system is initiated by tripping the normal actuation instrumentation." The corresponding ITS SR is ITS SR 3.6.7.4, however the ITS tests may be initiated by either an actual or simulated actuation signal. The CTS markup does not show this change "normal actuation" to "actual or simulated actuation" but it does show that the statement is relocated (DOC LA.1). This is incorrect. "Tripping the normal actuation" connotes only a simulated actuation. By adding the

words "actual actuation" the change becomes a Less Restrictive (L) change. Comment: Revise the CTS markup and provide a discussion and justification for this Less Restrictive (L) change.

Response:

L.05 has been revised to describe and justify that this change allows either a simulated or an actual actuation signal. The associated NSHC remains appropriate as initially written.

SECTION 3.7.4

The following information is provided in response to the Nuclear Regulatory Commission (NRC) staff's requests for additional information dated August 24, 2000, and from the NRC staff's requests for additional information made during a conference call between the NRC staff and PBNP staff on August 14, 2000.

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

NRC Question 3.7.4-1:

The Atmospheric Dump Valves (ADVs) are remote air-operated and manual-operated valves and the ADV block valves are local manually-operated (only) valves upstream of the ADVs. Both are credited with being manually closed to isolate a stuck-open ADV and with being manually reopened to establish ADV flow. The proposed surveillance requirement in the Improved Technical Specification (ITS) 3.7.4 is to manually exercise the ADVs and the ADV block valves at an 18-month frequency with no steam pressure or steam flow required for the exercise tests. Provide design and/or qualification information which verifies that the ADVs and the ADV block valves are capable of being manually closed and opened within the required time period for the postulated ADV blowdown conditions. How will the proposed surveillance test with no pressure or flow assure that the valves continue to be capable of being operated under pressure and flow conditions?

Response:

The ADVs are air operated fail closed valves, with the capability to be remotely opened and closed. Local manual operation of the ADVs is credited during a Steam Generator Tube Rupture (SGTR) event coincident with a Loss of Offsite Power (LOOP).

The ADVs are ASME Class II valves, which are required by 10 CFR 50.55a to be tested in accordance with ASME Section XI. However, this testing does not encompass local manual operation. Proposed SR 3.7.4.1 will require local manual testing of the ADVs, with or without steam flow, at an 18 month frequency.

In June, 1996, a satisfactory demonstration of the ability to manually operate the ADVs locally with steam flow was performed. This one time test, in conjunction with the ASME Section XI

operation of the ADVs using the air operator and proposed SR 3.7.4.1, will verify the capability to manually operate the ADVs locally during a SGTR/LOOP event. The 18 month testing frequency proposed for ADV local manual operation is adequate based on the engineering judgement that the failure of the ability to manually operate these valves is highly improbable.

With regard to the ADV block valves, the ITS submittal has been revised such that the ADV block valves are only credited with isolation of a failed open ADV. The ADV block valves will not be credited for re-establishing ADV flow for the mitigation of a SGTR/LOOP event. If it is necessary to close an ADV block valve to isolate a failed open ADV, that ADV flowpath will be considered inoperable.

SR 3.7.4.2 which proposes to manually exercise the ADV block valves at an 18 month frequency, with or without steam flow, will be sufficient to ensure its capability to isolate a failed open ADV.

NRC Question 3.7.4-2:

Since the ADV block valves are gate valves and may be required to be reopened after being closed, provide verification that the valves will not be subject to thermal binding and/or pressure locking after being closed, or, if the valves are subject to thermal binding and/or pressure locking, provide the corrective actions taken to assure the valves can be successfully reopened.

Response:

The ITS submittal has been revised such that the ADV block valves are only credited with isolation of a failed open ADV. The ADV block valves will not be credited for re-establishing ADV flow for the mitigation of a SGTR/LOOP event. If it is necessary to close an ADV block valve to isolate a failed open ADV, that ADV flowpath will be considered inoperable.

NRC Question 3.7.4-3:

Allowing seven days to restore an inoperable ADV (LCO 3.7.4, Condition A) has not been adequately justified. Provide additional justification or propose a shorter allowed outage time (AOT).

Response:

Justification has been provided to support a 48 hour AOT based on risk insights from the Point Beach Probabilistic Risk Assessment (PRA) model. Although a 48 hour AOT is being proposed, the evaluation was performed based on the originally proposed 7-day AOT for one inoperable ADV flowpath. Two different methods were used to evaluate the risk impact of a 7-day AOT. The results from both methods show that a 7-day AOT will have minimal risk impact. However, a 48 hour AOT is being proposed so as to provide additional margin. This proposal is bounded by the following analysis.

The Point Beach PRA model used for this evaluation only includes internal events with the reactor at power. The Point Beach PRA model is a reasonable representation of the as-built, as-operated plant. It is awaiting an independent review by a peer review team. External events such as fires, floods, earthquakes, high winds, etc. and events with the reactor shutdown are not included in the numerical results. However, due to the limited use of ADVs to mitigate these events, having an ADV unavailable for maintenance should not impact external event or shutdown risk to any significant degree.

ADV AOT, Method 1

The first method of determining risk change evaluated the impact on average core damage risk. This was done by adding a test and maintenance unavailability to the random failure probability for both of the ADVs. The PRA model was then requantified including these two new values for ADV unavailability, and the resulting core damage frequency (CDF) was compared to the base CDF to determine a ΔCDF. A ΔCDF of less than 1E-06/yr indicates a minimal increase in risk. This criteria is consistent with that used in NRC Regulatory Guide 1.174, "An Approach for Using Probabilistic Risk Assessment In Risk-Informed Decisions On Plant-Specific Changes to the Licensing Basis."

The new failure probability for the ADVs was developed first by adding a new test and maintenance unavailability to the random failure probability. For a 7-day AOT, the result is:

$$(7 \text{ days} / 300 \text{ days}) + 4.0\text{E-}03 = 2.73\text{E-}02$$

This assumes one AOT period per valve per year, and 300 days of reactor operation (82% availability) per year. A higher reactor availability makes this estimated failure probability conservative.

Quantifying the PRA model with these new failure probabilities yields a CDF of 4.623 E-05/yr. Given a base CDF of 4.571E-05/yr, the ΔCDF is then:

$$4.623\text{E-}05/\text{yr} - 4.571\text{E-}05/\text{yr} = 5.2\text{E-}07/\text{yr}$$

This result is less than 1E-06/yr, which indicates that the risk increase for a 7-day AOT is small. Note that this method overestimates the actual risk increase because it includes the random probability of having both ADVs unavailable due to maintenance.

ADV AOT, Method 2

The second method of determining risk change evaluated the temporary risk increase due to having one of the ADVs (MS 2015, MS 2016) out of service. The first step determined the change in instantaneous risk due to having an ADV out of service. This change in core damage frequency per year was then multiplied by the fraction of a year that we anticipate to experience the temporary increase in risk, to arrive at a delta core damage probability (ΔCDP). The temporary risk increase is considered to be of low risk significance if the ΔCDP is less than 1E-06. This is consistent with the criteria in EPRI TR-105396, "PSA Applications Guide."

Quantifying the PRA model with each of the ADV's failure probability set to 1.0, one at a time, gave the following results:

For MS 2015 failed, ΔCDF = 5.994E-05/yr - 4.571E-05/yr = 1.42E-05/yr

For MS 2016 failed, ΔCDF = 4.696E-05/yr - 4.571E-05/yr = 1.25E-06/yr

The difference between the ΔCDFs for the two valves is due to asymmetries in the PRA model; the value for MS 2015 will be used because it yields a more conservative result.

The fraction of the year for the temporary risk increase is (7 days / 300 days/yr) = 2.33E-02 yr. Multiplying this by the ΔCDF will give a ΔCDP:

1.42E-05/yr * 2.33E-02 yr = 3.32E-07

This change in core damage probability is lower than the 1E-06 criteria; therefore, the temporary risk increase is of low risk significance.

Using either of these methods, the proposed 48 hour AOT for Condition A (one ADV flowpath inoperable) provides additional margin over the 7-day AOT used in this analysis and is therefore acceptable.

The decision to propose a 48 hour AOT was also based on dose associated with the currently analyzed steam generator tube rupture accident (which credits an ADV for accident mitigation). Based on the relatively low doses that that are predicted to occur at the site boundary and the low population zone as a result of the analyzed accident (compared to the 10 CFR 100 limits), extending the AOT from 24 hours to 48 hours was considered acceptable.

Both ADV Flowpaths Inoperable

For both ADV flowpaths inoperable concurrently (for which a 24-hour AOT is proposed), the core damage frequency risk increases to 3.431E-04/yr. Our base CDF value is 4.571E-05, yielding a CDF increase of 2.974E-04/yr. This CDF increase coupled with the delta core damage probability increase criteria for a temporary change of 1.0E-06 would justify a 29-hour AOT. Therefore, the proposed 24-hour AOT for Condition B (both ADV flowpaths operable) is consistent with acceptable changes for core damage probability in the PRA model.

SECTION 3.7.5

The following information is provided in response to the Nuclear Regulatory Commission (NRC) staff's requests for additional information dated September 8, 2000.

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

NRC Question 3.7.5-1:

Proposed Surveillance Requirements (SRs) 3.7.5.2 and 3.7.5.4 each have a note which states "Not required to be performed for the turbine driven AFW pump until 24 hours after Thermal Power reaches ≥ 5% RTP." However, in NUREG-1431 Revision 1 and Draft Revision 2, the SR notes state "Not required to be performed for the turbine driven AFW pump until [24 hours] after [≥ 1000] psig in the steam generator." The proposed SR notes are not consistent with NUREG-1431 Revision 1. Provide justification for the requirement of ≥ 5 percent RTP versus ≥ 1000 psig as stated in NUREG-1431.

Response:

The note modifying ITS SR 3.7.5.2 has been changed to reflect the allowances of the current licensing basis, i.e., not required to be performed for the turbine driven AFW pump until 24 hours after thermal power exceeds 2% RTP. The Frequency of SR 3.7.5.4 has been changed to reflect the requirement of the STS, i.e., not required to be performed for the turbine driven AFW pump until 24 hours after greater than or equal to 1000 psig in the steam generator. All associated CTS markups, STS markups, DOCs, JFDs and clean ITS copies have been revised to reflect these changes.

NRC Question 3.7.5-2:

The proposed SR 3.7.5.5 test frequency states "Prior to Thermal Power exceeding 5% RTP whenever unit has been in Mode 5, Mode 6, or defueled for a cumulative period of > 30 days." However, in NUREG-1431 Revision 1 and Draft Revision 2, the test frequency states, "Prior to entering Mode 2 whenever unit has been in Mode 5, Mode 6, or defueled for a cumulative period of > 30 days." The proposed test frequency is not consistent with NUREG-1431 Revision 1. Provided justification for the test frequency of prior to thermal power exceeding 5 percent RTP versus prior to entering Mode 2 as stated in NUREG-1431.

Response:

The note modifying ITS SR 3.7.5.5 has been changed to be consistent with the requirements of the current licensing basis, i.e., required to be performed prior to thermal power exceeding 2% RTP. All associated CTS markups, STS markups, DOCs, JFDs and clean ITS copies have been revised to reflect these changes.

ATTACHMENT 2
DISCARD AND INSERTION INSTRUCTIONS

VOLUME 7	
SECTION 3.6.1	
DISCARD	**INSERT**
DOC pages 1 of 5 through 5 of 5	DOC pages 1 of 6 through 6 of 6
CTS markup pages 1 of 10 and 5 of 10	CTS markup pages 1 of 10 and 5 of 10
JFD pages 1 of 4 through 4 of 4	JFD pages 1 of 3 through 3 of 3
ISTS markup page 3.6-2	ISTS markup page 3.6-2
ISTS Bases markup pages B 3.6.1-3 through B 3.6.1-5	ISTS Bases markup pages B 3.6.1-3 through B 3.6.1-5
ITS page 3.6.1-1	ITS page 3.6.1-1
ITS Bases pages B 3.6.1-2 through B 3.6.1-4	ITS Bases pages B 3.6.1-2 through B 3.6.1-4
NSHC pages 1 of 5 through 5 of 5	NSHC pages 1 of 4 through 4 of 4
SECTION 3.6.2	
DISCARD	**INSERT**
ISTS Bases markup pages B 3.6.2-2, B 3.6.2-3, B 3.6.2-7	ISTS Bases markup pages B 3.6.2-2, B 3.6.2-3, B 3.6.2-7
ITS Bases pages B 3.6.2-2, B 3.6.2-6	ITS Bases pages B 3.6.2-2, B 3.6.2-6
SECTION 3.6.3	
DISCARD	**INSERT**
ISTS Bases markup pages B 3.6.3-3 and B 3.6.3-8	ISTS Bases markup pages B 3.6.3-3 and B 3.6.3-8
ITS Bases pages B 3.6.3-2 and B 3.6.3-6	ITS Bases pages B 3.6.3-2 and B 3.6.3-6
SECTION 3.6.4	
DISCARD	**INSERT**
DOC pages 1 of 3 through 3 of 3	DOC pages 1 of 3 through 3 of 3
CTS markup pages 2 of 5, 5 of 5	CTS markup pages 2 of 5, 5 of 5
NSHC pages 1 of 4 through 4 of 4	NSHC pages 1 of 3 through 3 of 3

ATTACHMENT 2
DISCARD AND INSERTION INSTRUCTIONS

SECTION 3.6.6	
DISCARD	**INSERT**
DOC page 6 of 8	DOC page 6 of 8
JFD page 12 of 13	JFD page 12 of 13
ISTS Bases markup page B 3.6.6A-9	ISTS Bases markup page B 3.6.6A-9
ITS Bases page B 3.6.6-7 and B 3.6.6-8	ITS Bases page B 3.6.6-7 and B 3.6.6-8
SECTION 3.6.7	
DISCARD	**INSERT**
DOC pages 2 of 7, 3 of 7 and 6 of 7	DOC pages 2 of 7, 3 of 7 and 6 of 7
CTS markup page 2 of 8	CTS markup page 2 of 8
NSHC pages 1 of 8 through 8 of 8	NSHC pages 1 of 9 through 9 of 9
VOLUME 8	
SECTION 3.7.4	
DISCARD	**INSERT**
DOC pages 1 of 4 through 4 of 4	DOC pages 1 of 5 through 5 of 5
CTS markup page 2 of 5	CTS markup page 2 of 5
JFD pages 1 of 3 through 3 of 3	JFD pages 1 of 4 through 4 of 4
ISTS markup page 3.7-9	ISTS markup page 3.7-9
ISTS Bases markup pages B 3.7.4-2, B 3.7.4-3 and Inserts	ISTS Bases markup pages B 3.7.4-2, B 3.7.4-3 and Inserts
ITS page 3.7.4-1	ITS page 3.7.4-1
ITS Bases pages B 3.7.4-1 through B 3.7.4-4	ITS Bases pages B 3.7.4-1 through B 3.7.4-3
NSHC pages 2 of 5 and 3 of 5	NSHC pages 2 of 5 and 3 of 5
SECTION 3.7.5	
DISCARD	**INSERT**
DOC pages 1 of 11 through 11 of 11	DOC pages 1 of 10 through 10 of 10
CTS markup pages 7 of 12, 10 of 12 and 12 of 12	CTS markup pages 7 of 12, 10 of 12 and 12 of 12

ATTACHMENT 2
DISCARD AND INSERTION INSTRUCTIONS

SECTION 3.7.5 (continued)	
DISCARD	**INSERT**
JFD pages 1 of 9 through 9 of 9	JFD pages 1 of 7 through 7 of 7
ISTS markup pages 3.7-13 and 3.7-14	ISTS markup pages 3.7-13 and 3.7-14
ISTS Bases markup pages B 3.7.5-8, B 3.7.5-9 and Inserts (B 3.7.5-7, B 3.7.5-8 and B 3.7.5-9)	ISTS Bases markup pages B 3.7.5-8, B 3.7.5-9 and Inserts (B 3.7.5-7, B 3.7.5-8 and B 3.7.5-9)
ITS page 3.7.5-3 and 3.7.5-4	ITS page 3.7.5-3 and 3.7.5-4
ITS Bases pages B 3.7.5-1 through B 3.7.5-12	ITS Bases pages B 3.7.5-1 through B 3.7.5-9
NSHC pages 1 of 9 through 9 of 9	NSHC pages 1 of 8 through 8 of 8

ENCLOSURE

Description of Changes - NUREG-1431 Section 3.06.01

16-Oct-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:
15.03.06.A	LCO 3.06.01
15.03.06.A.01.A	DELETED
15.03.06.A.01.A.01	LCO 3.06.01 COND A LCO 3.06.01 COND A RA.1
15.03.06.A.01.A.02	LCO 3.06.01 COND B
15.03.06.A.01.A.02.A	LCO 3.06.01 COND B RA B.1
15.03.06.A.01.A.02.B	LCO 3.06.01 COND B RA B.2
15.03.06.E	LCO 3.06.01 SR 3.06.01.01 SR 3.06.01.02
15.04.02.B.02	SR 3.06.01.01
15.04.04.I	SR 3.06.01.01
15.04.04.II	SR 3.06.01.01

A.02 Rev. A

The CTS contains a footnote which provides reference to the section in the FSAR which discusses containment isolation valves. Reference to the FSAR in this fashion does not establish any regulatory requirements, as it is merely a reference. It is unnecessary to provide references in the Technical Specifications, references when necessary are provided in the Bases of the Improved Technical Specifications. Based on the reference not establishing any regulatory requirement, deletion of this reference from the Technical Specification is administrative in nature.

CTS:	ITS:
15.01.D *	DELETED

A.03 Rev. A

The definition of Containment Integrity has been moved from the Definitions Section of the Current Technical Specifications to proposed ITS LCO 3.6.1, Containment; LCO 3.6.2, Containment Air Locks; and LCO 3.6.3, Containment Isolation Valves. This change is administrative in that all of the CTS requirements continue to be addressed within the aforementioned LCOs. This change eliminates confusion associated with meeting the definition of CONTAINMENT INTEGRITY when required equipment/components are inoperable. This change is administrative in nature.

CTS:	ITS:
15.01.D	LCO 3.06.01

DOC Number	DOC Text
A.04 Rev. A	The CTS Definition of Containment integrity states that the overall uncontrolled containment leakage shall be maintained less than La. The CTS definition and the Containment Leakage Rate Testing Program establishes the as found and as left leakage limits at 1.0 La, and 0.6 La for combined Type B and C tests and 0.75 La for Type A tests. In the proposed ITS, the requirement to maintain Type A, B, and C leakage less than La is contained in LCO 3.6.1. The proposed ITS Containment Leakage Rate Testing Program contains the as found and as left containment leakage limits consistent with the CTS limits. CTS item 15.1.D.2, requires the equipment hatch to be properly closed. The equipment hatch is a Type B penetration. Proper installation is concluded through performance of an acceptable Type B leakage test as required by proposed ITS SR 3.6.1.1. Proposed SR 3.6.3.3 requires isolation valves and blind flanges located inside the containment to be verified closed prior to entry into Mode 4 from Mode 5 if not performed in the previous 92 days. The combination of these two SRs provides assurance that the equipment hatch is properly closed, thereby incorporating CTS item 15.1.D.2 into LCO 3.6.1 and 3.6.3. These changes are administrative. All of the CTS requirements continue to be addressed within the aforementioned LCOs and Surveillance Requirements. These changes eliminate confusion associated with meeting the definition of containment integrity when required equipment/components are inoperable.

CTS:	ITS:
15.01.D.02	SR 3.06.01.01
15.01.D.04	SR 3.06.01.01
15.01.D.04 **	SR 3.06.01.01

A.05 Rev. A	The CTS provides an introductory statement (Applicability) which simply states which systems/components are addressed within a given section. This same information while worded differently is contained within the title of each ITS LCO. Accordingly, this change is a change in format with no change in technical requirement.

CTS:	ITS:
15.03.06 APPL	LCO 3.06.01
15.04.04 APPL	LCO 3.06.01

A.06 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 3.06.01

DOC Number	DOC Text
A.07 Rev. A	CTS 15.3.6.A.1 requires containment integrity whenever a nuclear core is installed in the reactor, unless the reactor is in the cold shutdown condition. Proposed ITS LCO 3.6.1 require the containment to be operable in Modes 1, 2, 3, and 4. The ITS definition of Mode requires there to be fuel in the reactor to be in a defined Mode of Applicability (e.g. Mode 1, 2, 3, 4, 5, or 6) making the CTS and ITS equivalent regarding the presence of fuel. The CTS definition of Cold Shutdown requires the reactor to have a shutdown margin of at least 1% with RCS temperature less than or equal to 200 degrees. The ITS definition of Cold Shutdown (ITS Table 1.1-1 - Mode 5), is defined as Keff less than 0.99 with RCS temperature of less than or equal to 200 degrees making the CTS and ITS equivalent relative to temperature and reactivity. Based on the above, this change is administrative.

CTS:	ITS:
15.03.06.A.01	LCO 3.06.01

A.08 Rev. B	Not used.

CTS:	ITS:
N/A	N/A

A.09 Rev. A	The CTS provides an introductory statement (Objective) at the beginning of this Section of the Technical Specifications which provide a brief summary of the purpose for this Section. This information is contained in the Bases Section of the ITS. This information does not establish any regulatory requirements for the systems and components addressed within this Section. Accordingly, deletion of this information does not alter any requirement set forth in the Technical Specifications. This change is administrative and consistent with the format and presentation for the ITS as provided in NUREG 1431.

CTS:	ITS:
15.03.06 OBJ	DELETED
15.04.04 OBJ	DELETED

DOC Number	DOC Text
L.01 Rev. A	The CTS requires containment integrity under a number of conditions to include: 1) Whenever a nuclear core is installed in the reactor and the reactor is not in the cold shutdown condition; 2) When the reactor vessel head is removed unless the reactor is in the refueling shutdown condition; 3) Whenever positive reactivity changes are made by rod drive motion, except when testing one bank of rods at a time, rod disconnecting, and rod reconnecting provided the reactor is initially subcritical by at least 5% delta k/k; and 4) Whenever making positive reactivity changes by boron dilution unless the RCS boron concentration is maintained > 2100 ppm. The ITS will require containment integrity to be maintained in Modes 1, 2, 3, and 4 (whenever the reactor is not in cold shutdown). All other conditions and limitations have been deleted from the Technical Specifications. There are no shutdown accidents (RCS temperature less than or equal to 200 degrees) in the Point Beach current licensing basis which credits containment integrity for accident mitigation. Specifically; inadvertent RCS dilution in cold shutdown and refueling is terminated by operator action before the reactor reaches a Keff of 1.0, inadvertent rod withdrawal is terminated by the reactor protection system before fuel damage occurs, and accidental release of liquid and gaseous wastes are independent of containment status. This relaxation is consistent with analysis assumptions for Point Beach. Accordingly, these requirements may be deleted from the Technical Specifications as they are not required to provide adequate protection of public health and safety.

CTS:	ITS:
15.03.06.A.01	LCO 3.06.01
15.03.06.C	DELETED
15.03.06.D	DELETED
15.03.06.D *	DELETED
BASES	DELETED

DOC Number	DOC Text
LA.01 Rev. C	The title, footnote, and statement information of CTS 15.1.D are introductory and convey no requirements. Therefore, the title and statement information are covered by the A.1 DOCs in Sections 3.6.2 and 3.6.3 and DOC A.3 of Section 3.6.1 of the submittal. The footnote is covered under the A.2 DOC in Section 3.6.1. The requirements of CTS 15.1.D.1), including subsections a, b, and c, are properly covered by DOCs A.1 and A.2 in Section 3.6.3. The requirement of CTS 15.1.D.2) is properly covered by DOC A.4 in section 3.6.1 and DOC A.2 in Section 3.6.3. The requirement of CTS 15.1.D.3) is properly covered by DOC A.2 in Section 3.6.2. The requirement of CTS 15.1.D.4), including the footnote, is properly covered by DOC A.4 in Section 3.6.1. Therefore, the entire definition has been appropriately covered by the applicable DOCs and the associated requirements are contained in the ITS. The definition in its entirety is also contained in the basis where it expounds on the associated requirements.

CTS:	ITS:
15.01.D	B 3.06.01
15.01.D *	B 3.06.01
15.01.D.01	B 3.06.01
15.01.D.01.A	B 3.06.01
15.01.D.01.B	B 3.06.01
15.01.D.01.C	B 3.06.01
15.01.D.02	B 3.06.01
15.01.D.03	B 3.06.01
15.01.D.04	B 3.06.01
15.01.D.04 **	B 3.06.01

LB.01 Rev. C	Not used.

CTS:	ITS:
N/A	N/A

DOC Number	DOC Text
M.01 Rev. A	CTS 15.3.10.E.1 and 2 contain remedial actions for single and multiple containment tendon failures. Dependent upon the level of degradation incurred, either 15 days or 72 hours is allowed to restore the tendon(s) to operable status before requiring the unit to be placed into Hot Shutdown within 6 hours and Cold Shutdown within the following 30 hours. The Point Beach containment structure is constructed with sufficient margin to allow up to three adjacent tendons to be detensioned (inoperable) without a detrimental effect on containment integrity. The proposed ITS does not contain an explicit condition for tendon inoperabilities; however, upon discovery of a degraded condition, an assessment must be made relative to containment integrity. If the assessment concludes that containment integrity cannot be maintained, the proposed ITS will allow 1 hour to restore the containment to operable status before requiring the unit to be placed into Mode 3 within 6 hours and Mode 5 within 36 hours. Accordingly, deletion of the CTS provision which could allow containment integrity to be impaired for up to 72 hours before requiring the unit to be shutdown is a more restrictive change

CTS:	ITS:
15.03.06.E.01	DELETED
15.03.06.E.02	DELETED

LA.1 A.3 LCO 3.6.1 C RAI 3.6.1-1 A.1

D. Containment Integrity* A.2

<Also See LCOs 3.6.2 and 3.6.3>

Containment integrity is defined to exist when:

1) Penetrations required to be isolated during accident conditions are either:
   a. Capable of being closed by an operable automatic containment isolation valve,
      OR
   b. Closed by an operable containment isolation valve,
      OR
   c. Closed in accordance with Specifications 15.3.6.A.1.b and 15.3.6.A.1.c.

LA.1 C RAI 3.6.1-1

2) The equipment hatch is properly closed.
3) At least one door in each personnel air lock is properly closed.
4) The overall uncontrolled containment leakage is less than La.**

<See LCO 3.6.2 for Airlock contribution to the overall leakage rate limit.>

E. Protective Instrumentation Logic
1) Analog Channel
An analog channel is an arrangement of components and modules as required to generate a single protective action signal when required by a plant condition. An analog channel loses its identity where single action signals are combined.

<See Section 1.0>

SR 3.6.1.1 A.4 LA.1 C RAI 3.6.1-1

SR 3.6.1.1 and CLRTP Specification 5.5.16

A.4

A.2

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* ~~Containment isolation valves are discussed in FSAR Section 5.2.~~

** Prior to the first startup after performing a required Containment Leakage Rate Testing Program leakage test, the applicable leakage limits specified in TS 15.6.12.D.2 must be met.

A.1

LCO 3.6.1

SR 3.6.1.1

E. CONTAINMENT STRUCTURAL INTEGRITY

The structural integrity of the reactor containment shall be maintained in accordance with the surveillance criteria specified in the Containment Leakage Rate Testing Program and 15.4.4.II.

SR 3.6.1.2

C RAI 3.6.1-7

M.1

1. If more than one tendon is observed with a prestressing force between the predicted lower limit (PLL) and 90% of the PLL or if one tendon is observed with prestressing force less than 90% of the PLL, the tendon(s) shall be restored to the required level of integrity within 15 days or the reactor shall be in hot standby within the next six hours and in cold shutdown within the following 30 hours. An engineering evaluation of the situation shall be conducted and a special report submitted in accordance with Specification 15.4.4.II.D within 30 days.

2. With an abnormal degradation of the containment structural integrity in excess of that specified in 15.3.6.E.1, and at a level below the acceptance criteria of Specification 15.4.4.II, restore the containment structural integrity to the required level within 72 hours or be in hot shutdown within the next six hours and in cold shutdown within the following 30 hours. Perform an engineering evaluation of the containment structural integrity and provide a special report in accordance with Specification 15.4.4.II.D within 30 days.

<See Section 5.0.>

Justification For Deviations - NUREG-1431 Section 3.06.01

16-Oct-00

JFD Number	JFD Text	
01 Rev. A	The Bases for LCO 3.6.1 of NUREG 1431 was developed to address four groups of containment Designs; Ice Condensers, Sub-Atmospheric, Dual, and Atmospheric. Point Beach containment is an atmospheric design, as such the Bases for the Ice Condenser, Dual, and Sub-Atmospheric designs have not been incorporated. The Titles for LCO 3.6.1 and it associated Bases have been shortened to simply state "Containment". Inclusion of the type of design (e.g. Ice Condenser, Dual, Atmospheric, or Sub-Atmospheric) is a detail only relevant in distinguishing the NUREG variations.	
	ITS:	**NUREG:**
	B 3.06.01	B 3.06.01
	LCO 3.06.01	LCO 3.06.01
02 Rev. C	Not Used.	
	ITS:	**NUREG:**
	N/A	N/A
03 Rev. C	LCO 3.6.1 and its associated Bases have been modified to incorporate Option B to 10 CFR 50 Appendix J. These modifications include: 1) The Bases of LCO 3.6.1 states that the containment is designed to contain radioactive material following a design basis accident. This statement was revised to state that the containment is designed to contain radioactive material following a design basis "loss of coolant accident". As re-enforced by the positions established in Appendix J, Option B of 10 CFR 50 and its implementing documents, radioactive release from the containment as the result of a design basis accident is assumed to occur from primary system loss of coolant accidents. This change is consistent with the CTS Bases wording approved in amendment 169/173 on October 9, 1996 for the implementation of Option B. This change results in defining DBA as an acronym for Design Basis Accident in a later paragraph in this Bases section. 2) Various references to 10 CFR 50 Appendix J have been revised to 10 CFR 50 Appendix J Option B to provide for proper and complete reference to Appendix J.	
	ITS:	**NUREG:**
	B 3.06.01	B 3.06.01
04 Rev. A	The Bases for NUREG 1431 LCO 3.6.1 lists the pressurized sealing mechanism as an attribute associated with the containment penetration boundaries as a bracketed (design specific) discussion. Point Beach does not have a penetration pressurization system, therefore, reference to this bracketed attribute has been omitted.	
	ITS:	**NUREG:**
	B 3.06.01	B 3.06.01

Justification For Deviations - NUREG-1431 Section 3.06.01

16-Oct-00

JFD Number	JFD Text
05 Rev. A	The brackets have been removed and the proper plant specific information has been provided.

ITS:	NUREG:
B 3.06.01	B 3.06.01
SR 3.06.01.02	SR 3.06.01.02

06
Rev. A

NUREG 1431 contains the Surveillance Requirements and Actions for containment purge valves with resilient seals in LCO 3.6.3. This presentation establishes surveillance frequencies and Actions for containment purge valves which differ from those contained in LCO 3.6.1 for other containment isolation valves. Surveillance frequencies and Actions above and beyond those established in LCO 3.6.1 and through the Containment Leakage Rate Testing Program (SR 3.6.1.1) are not necessary for Point Beach. The CTS prior to October 9, 1996 (Technical Specification Amendment 169/173) required testing of the containment purge valves every 6 months based on the findings of generic issue B-20 "Containment Leakage Due to Seal Degradation". Amendment 169/173 eliminated the requirement for increased testing of the containment purge valves. As cited in the SER for amendments 169/173, the containment purge valve can be tested in accordance with the Regulatory Guide 1.163 "Performance-Based Containment Leak-Testing Program". The basis of this conclusion was that there has not been observable degradation supportive of increased testing frequencies which were established as part of Generic Issue B-20. Since 1992 there had been no leakage rate failures in excess of the previous Technical Specification or Appendix J acceptance criteria, nor were there failures in excess of the administrative leakage limit of 2000 standard cubic centimeters per minute.

Accordingly, the bracketed information contained in the Bases of SR 3.6.1.1, referring to LCO 3.6.3 for purge valve leakage limitations was not adopted.

ITS:	NUREG:
B 3.06.01	B 3.06.01

07
Rev. B

PBNP only has one equipment hatch for each containment, therefore the word "hatches" has been changed to singular form to reflect this.

ITS:	NUREG:
B 3.06.01	B 3.06.01

Justification For Deviations - NUREG-1431 Section 3.06.01

16-Oct-00

JFD Number	JFD Text	
08 Rev. A	The Bases of NUREG 1431 LCO 3.6.1 describes the containment penetrations that form the containment leakage barrier. Contained within the listing is a statement that "all equipment hatches are closed". The Point Beach containment has only a single containment equipment hatch which incorporates an airlock as well. As such, the ITS Bases has been changed requiring "the equipment hatch to be installed". The requirement for the airlock, which is incorporated into the equipment hatch to be closed and sealed is addressed as part of the previous Bases statement requiring each airlock to be operable. This deviation from the NUREG is necessary to reflect the Point Beach design.	
	ITS:	**NUREG:**
	B 3.06.01	B 3.06.01
09 Rev. A	NUREG 1431 LCO 3.9.2, "Unborated Water Source Isolation Valves", is not applicable to Point Beach as described in Justification for Deviation 01 of LCO 3.9.2. Corresponding reference changes have been made as necessary to maintain proper reference.	
	ITS:	**NUREG:**
	B 3.06.01	B 3.06.01

1

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.6.1.1	Perform required visual examinations and leakage rate testing except for containment air lock testing, in accordance with ~~10 CFR 50, Appendix J, as modified by approved exemptions.~~ ~~The leakage rate acceptance criterion is ≤ 1.0 L_a. However, during the first unit startup following testing performed in accordance with 10 CFR 50, Appendix J, as modified by approved exemptions, the leakage rate acceptance criteria are < 0.6 L_a for the Type B and Type C tests, and < 0.75 L_a for the Type A test.~~	~~NOTE SR 3.0.2 is not applicable~~ Approved TSTF-52 In accordance with ~~10 CFR 50, Appendix J, as modified by approved exemptions~~
SR 3.6.1.2	Verify containment structural integrity in accordance with the Containment Tendon Surveillance Program.	In accordance with the Containment Tendon Surveillance Program

Approved TSTF-52

the Containment Leakage Rate Testing Program.

C RAI 3.6.1-5

C RAI 3.6.1-7

5

5

APPLICABLE SAFEETY ANALYSES (Continued)

The containment satisfies Criterion 3 of the NRC Policy Statement.

Approved TSTF-52

LCO

Containment OPERABILITY is maintained by limiting leakage to ≤ 1.0 L_a, except prior to the first startup after performing a required ~~10 CFR 50, Appendix J,~~ leakage test. At this time, ~~the combined Type B and C leakage must be < 0.6 L_a, and the overall Type A leakage must be < 0.75 L_a.~~

Containment Leakage Rate Testing Program

the applicable leakage limits contained in the Containment Leakage Rate Testing Program must be met.

3

including the equipment hatch

7

Compliance with this LCO will ensure a containment configuration, including equipment hatches, that is structurally sound and that will limit leakage to those leakage rates assumed in the safety analysis.

, Option B

3

Individual leakage rates specified for the containment air lock (LCO 3.6.2) ~~[and purge valves with resilient seals (LCO 3.6.3)]~~ are not specifically part of the acceptance criteria of 10 CFR 50, Appendix J. Therefore, leakage rates exceeding these individual limits only result in the containment being inoperable when the leakage results in exceeding the acceptance criteria of ~~Appendix J~~.

6

overall

Approved TSTF-52

1.0 L_a

RAI 3.6.1-7

RAI 3.6.1-3

RAI 3.6.1-5

APPLICABILITY

In MODES 1, 2, 3, and 4, a ~~DBA~~ could cause a release of radioactive material ~~into~~ containment. In MODES 5 and 6, the probability and consequences of these events are reduced due to the pressure and temperature limitations of these MODES. Therefore, containment is not required to be OPERABLE in MODE 5 to prevent leakage of radioactive material from containment. The requirements for containment during MODE 6 are addressed in LCO 3.9.4 "Containment Penetrations."

design basis Loss of Coolant Accident

3

3

9

ACTIONS

A.1

In the event containment is inoperable, containment must be restored to OPERABLE status within 1 hour. The 1 hour Completion Time provides a period of time to correct the problem commensurate with the importance of maintaining containment during MODES 1, 2, 3, and 4. This time period

ACTIONS (Continued)

also ensures that the probability of an accident (requiring containment OPERABILITY) occurring during periods when containment is inoperable is minimal.

B.1 and B.2

If containment cannot be restored to OPERABLE status within the required 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 3 within 6 hours and to MODE 5 within 36 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 3.6.1.1

3 → containment

6

Maintaining the containment OPERABLE requires compliance with the visual examinations and leakage rate test requirements of ~~10 CFR 50, Appendix J (Ref. 1), as modified by approved exemptions~~. Failure to meet air lock ~~[and purge valve with resilient seal]~~ leakage limits specified in LCO 3.6.2 ~~[and LCO 3.6.3]~~ does not invalidate the acceptability of these overall leakage determinations unless their contribution to overall Type A, B, and C leakage causes that to exceed limits. As left leakage prior to the first startup after performing a required 10 CFR 50, ~~Appendix J~~, leakage test is required to be ≤ 0.6 L_a for combined Type B and C leakage, and ≤ 0.75 L_a for overall Type A leakage. At all other times between required leakage rate tests, the acceptance criteria is based on an overall Type A leakage limit of ≤ 1.0 L_a. At ≤ 1.0 L_a the offsite dose consequences are bounded by the assumptions of the safety analysis. SR Frequencies are as required by ~~Appendix J, as modified by approved exemptions. Thus, SR 3.0.2 (which allows Frequency extensions) does not apply~~. These periodic testing requirements verify that the containment leakage rate does not exceed the leakage rate assumed in the safety analysis.

3

≤

3

the Containment Leakage Rate Testing Program.

Containment Leakage Rate Testing Program

Approved TSTF-52

C

RAI 3.6.1-5

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.6.1.2

For ungrouted, post tensioned tendons, this SR ensures that the structural integrity of the containment will be maintained in accordance with the provisions of the Containment Tendon Surveillance Program. Testing and Frequency are consistent with the recommendations of Regulatory Guide 1.35 (Ref. 4). 1 5

C RAI 3.6.1-7

REFERENCES

1. 10 CFR 50, Appendix J. , Option B 3
2. FSAR, Chapter [15] 14 5
3. FSAR, Section [6.2] 5.1
4. Regulatory Guide 1.35, Revision [1] 3

C RAI 3.6.1-7

3.6 CONTAINMENT SYSTEMS

3.6.1 Containment

LCO 3.6.1 Containment shall be OPERABLE.

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

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. Containment inoperable.	A.1	Restore containment to OPERABLE status.	1 hour
B. Required Action and associated Completion Time not met.	B.1	Be in MODE 3.	6 hours
	AND		
	B.2	Be in MODE 5.	36 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.6.1.1	Perform required visual examinations and leakage rate testing except for containment air lock testing, in accordance with the Containment Leakage Rate Testing Program.	In accordance with the Containment Leakage Rate Testing Program C RAI 3.6.1-5
SR 3.6.1.2	Verify containment structural integrity in accordance with the Containment Tendon Surveillance Program.	In accordance with the Containment Tendon Surveillance Program C RAI 3.6.1-7

BASES

APPLICABLE SAFETY ANALYSES

The safety design basis for the containment is that the containment must withstand the pressures and temperatures of the limiting design basis Loss of Coolant Accident without exceeding the design leakage rate.

For the design basis Loss of Coolant Accident analyses, it is assumed that the containment is OPERABLE such that, the release of fission product radioactivity, release to the environment is controlled by the rate of containment leakage. The containment was designed with an allowable leakage rate of 0.4% of containment air weight per day (Ref. 3). This leakage rate, used to evaluate offsite doses resulting from accidents, is defined in 10 CFR 50, Appendix J, Option B (Ref. 1), as L_a: the maximum allowable containment leakage rate at the calculated peak containment internal pressure (P_a) resulting from the limiting design basis LOCA. The allowable leakage rate represented by L_a forms the basis for the acceptance criteria imposed on all containment leakage rate testing. L_a is assumed to be 0.4% per day in the safety analysis at P_a = 60 psig (Ref. 3). B RAI 3.6.1-7

Satisfactory leakage rate test results are a requirement for the establishment of containment OPERABILITY.

The containment satisfies Criterion 3 of the NRC Policy Statement.

LCO

Containment OPERABILITY is maintained by limiting leakage to $\leq$ 1.0 L_a, except prior to the first startup after performing a required Containment Leakage Rate Testing Program leakage test. At this time, the applicable leakage limits contained in the Containment Leakage Rate Testing Program must be met. B RAI 3.6.1-7 C RAI 3.6.1-5

Compliance with this LCO will ensure a containment configuration, including the equipment hatch, that is structurally sound and that will limit leakage to those leakage rates assumed in the safety analysis. B RAI 3.6.1-3

Individual leakage rates specified for the containment air lock (LCO 3.6.2) are not specifically part of the acceptance criteria of 10 CFR 50, Appendix J, Option B. Therefore, leakage rates exceeding these individual limits only result in the containment being inoperable when the leakage results in exceeding the overall acceptance criteria of 1.0 L_a. C RAI 3.6.1-5

BASES

APPLICABILITY

In MODES 1, 2, 3, and 4, a design basis Loss of Coolant Accident could cause a release of radioactive material into containment. In MODES 5 and 6, the probability and consequences of these events are reduced due to the pressure and temperature limitations of these MODES. Therefore, containment is not required to be OPERABLE in MODE 5 to prevent leakage of radioactive material from containment. The requirements for containment during MODE 6 are addressed in LCO 3.9.3, "Containment Penetrations."

ACTIONS

A.1

In the event containment is inoperable, containment must be restored to OPERABLE status within 1 hour. The 1 hour Completion Time provides a period of time to correct the problem commensurate with the importance of maintaining containment during MODES 1, 2, 3, and 4. This time period also ensures that the probability of an accident (requiring containment OPERABILITY) occurring during periods when containment is inoperable is minimal.

B.1 and B.2

If containment cannot be restored to OPERABLE status within the required 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 3 within 6 hours and to MODE 5 within 36 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 3.6.1.1

Maintaining the containment OPERABLE requires compliance with the visual examinations and containment leakage rate test requirements of the Containment Leakage Rate Testing Program. Failure to meet air lock leakage limits specified in LCO 3.6.2 does not invalidate the acceptability of these overall leakage determinations unless their contribution to overall Type A, B, and C leakage causes that to exceed limits. As left leakage prior to the first startup after performing a required Containment Leakage Rate Testing Program, leakage test is required to be $\leq 0.6\ L_a$ for combined Type B and C leakage, and $\leq 0.75\ L_a$ for overall Type A leakage. At all other times between required leakage rate tests, the acceptance criteria is based on an overall Type A leakage limit of $\leq 1.0\ L_a$. At $\leq 1.0\ L_a$ the offsite dose consequences C RAI 3.6.1-5

BASES

SURVEILLANCE REQUIREMENTS (continued)

are bounded by the assumptions of the safety analysis. SR Frequencies are as required by the Containment Leakage Rate Testing Program. These periodic testing requirements verify that the containment leakage rate does not exceed the leakage rate assumed in the safety analysis.

SR 3.6.1.2

For ungrouted, post tensioned tendons, this SR ensures that the structural integrity of the containment will be maintained in accordance with the provisions of the Containment Tendon Surveillance Program. Testing and Frequency are consistent with the recommendations of Regulatory Guide 1.35 (Ref. 4). C RAI 3.6.1-7

REFERENCES

1. 10 CFR 50, Appendix J, Option B.
2. FSAR, Chapter 14.
3. FSAR, Section 5.1.
4. Regulatory Guide 1.35, Revision 3.

C RAI 3.6.1-7

No Significant Hazards Considerations - NUREG-1431 Section 3.06.01

16-Oct-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? Containment integrity is not an initial condition of, or event precursor in any analyzed shutdown event (less than or equal to 200 degrees). Fuel handling events do not credit containment integrity nor filtration; dilution and rod withdrawal events are not impacted by containment status and are terminated prior to any release taking place; and liquid and gaseous release events are not impacted by containment status as the containment is not the assumed source of release for these events. Accordingly, the probability for analyzed event is not significantly increased as a result of this change. As previously stated, containment integrity is not assumed for any shutdown event, therefore, the consequences of an analyzed event is not significantly increased as a result of this change. 2. Does the change create the possibility of a new or different kind of accident from any accident previously evaluated? The proposed change does not 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. This proposed change makes the Mode of Applicability for the Containment consistent with the accident analyses which assume containment integrity. 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 change in applicability for containment integrity is consistent with the assumptions made in the various Point Beach accident analyses. Containment integrity will continue to be maintained in the various Operational Modes and Conditions for which containment integrity was assumed to be met. Therefore, the margin of safety is not significantly reduced as a result of this change

No Significant Hazards Considerations - NUREG-1431 Section 3.06.01

16-Oct-00

NSHC Number	NSHC Text
LA Rev. C	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.
LB Rev. C	Not Used.

No Significant Hazards Considerations - NUREG-1431 Section 3.06.01

16-Oct-00

NSHC Number	NSHC Text
M Rev. A	In accordance with the criteria set forth in 10 CFR 50.92, PBNP has evaluated this proposed Technical Specifications change and determined it does not represent a significant hazards consideration. The following is provided in support of this conclusion. 1. Does the change involve a significant increase in the probability or consequences of an accident previously evaluated? The proposed change provides more restrictive requirements for operation of the facility. These more stringent requirements do not result in operation that will increase the probability of initiating an analyzed event and do not alter the assumptions relative to the mitigation of an accident or transient event. These more restrictive requirements continue to ensure process variables, structures, systems and components are maintained consistent with the safety analyses. Therefore, this change does not increase the probability or consequences of an accident previously evaluated. 2. Does the change create the possibility of a new or different kind of accident from any accident previously evaluated? The proposed change does not require a physical alteration of the plant (no new or different type of equipment will be installed) or changes in parameters governing normal plant operation. The proposed change does impose different requirements. However, these changes are consistent with assumptions made in the safety analysis. Thus, this change does not create the possibility of a new or different kind of accident from any accident previously evaluated. 3. Does this change involve a significant reduction in a margin of safety? The imposition of more restrictive requirements either has no affect on or increases the margin of safety. Each change is providing additional restrictions to enhance plant safety. These changes are consistent with the safety analysis. Therefore, this change does not involve a reduction in a margin of safety.

C RAI 3.6.1-5

BASES

, Option B (Ref. 1), as L_a = 0.4% of containment air weight per day, the maximum allowable containment leakage rate at the calculated peak design containment internal pressure, Pa of 60 psig, following a design basis LOCA.

APPLICABLE SAFETY ANALYSES

The DBAs that result in a release of radioactive material within containment ~~are~~ is a loss of coolant accident ~~and a rod ejection accident~~ (Ref. 2). In the analysis of ~~each of~~ this ~~these~~ accidents, it is assumed that containment is OPERABLE ~~such that release~~ of fission products to the environment is controlled by the rate of containment leakage. The containment was designed with an allowable leakage rate of [0.1]% of containment air weight per day (Ref. 2). This leakage rate is defined in 10 CFR 50, Appendix J (Ref. 1), as L_a = [0.1]% of containment air weight per day, the maximum allowable containment leakage rate at the calculated peak containment internal pressure P_a = [14.4] psig following a DBA. This allowable leakage rate forms the basis for the acceptance criteria imposed on the SRs associated with the air locks.

The containment air locks satisfy Criterion 3 of the NRC Policy Statement.

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C RAI 3.6.1-5

LCO

Each containment air lock forms part of the containment pressure boundary. As part of containment, the air lock safety function is related to control of the containment leakage rate resulting from a DBA. Thus, each air lock's structural integrity and leak tightness are essential to the successful mitigation of such an event.

Each air lock is required to be OPERABLE. For the air lock to be considered OPERABLE, the air lock interlock mechanism must be OPERABLE, the air lock must be in compliance with the Type B air lock leakage test, and both air lock ~~doors~~ bulkheads must be OPERABLE. The interlock allows only one air lock door and its associated equalization valve of an air lock to be opened at one time. This provision ensures that a gross breach of containment does not exist when containment is required to be OPERABLE. Closure of a single door in each air lock is sufficient to provide a leak tight barrier following postulated events. Nevertheless, both doors and their associated equalization valves are kept closed when the air lock

The OPERABILITY of a single bulkhead (e.g., bulkhead door, door seals, equalization valve, interlock shaft seals, etc;)

LCO (continued)

Approved TSTF-52 or

is not being used for normal entry into ~~and~~ exit from containment.

4 a design basis LOCA

C RAI 3.6.1-5

APPLICABILITY

In MODES 1, 2, 3, and 4, a ~~DBA~~ design basis LOCA could cause a release of radioactive material to containment. In MODES 5 and 6, the probability and consequences of ~~these events~~ a design basis LOCA are reduced due to the pressure and temperature limitations of these MODES. Therefore, the containment air locks are not required in MODE 5 to prevent leakage of radioactive material from containment as a result of a design basis LOCA. The requirements for the containment air locks during MODE 6 are addressed in LCO 3.9.3, "Containment Penetrations."

4

ACTIONS

The ACTIONS are modified by a Note that allows entry and exit to perform repairs on the affected air lock component. If the outer door is inoperable, then it may be easily accessed for most repairs. It is preferred that the air lock be accessed from inside primary containment by entering through the other OPERABLE air lock. However, if this is not practicable, or if repairs on either door must be performed from the barrel side of the door then it is permissible to enter the air lock through the OPERABLE door, which means there is a short time during which the containment boundary is not intact (during access through the OPERABLE door). The ability to open the OPERABLE door, even if it means the containment boundary is temporarily not intact, is acceptable due to the low probability of an event that could pressurize the containment during the short time in which the OPERABLE door is expected to be open. After each entry and exit, the OPERABLE door must be immediately closed but is not required to be locked while repairs are actively being performed on the inoperable bulkhead. If ALARA conditions permit, entry and exit should be via an OPERABLE air lock.

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B RAI 3.6.2-3

A second Note has been added to provide clarification that, for this LCO, separate Condition entry is allowed for each air lock. This is acceptable, since the Required Actions for each Condition provide appropriate compensatory actions for each inoperable air lock. Complying with the Required

ACTIONS (continued)

an inoperable air lock to OPERABLE status, assuming that at least one door ~~is~~ and its associated equalization valve are 10 maintained closed in each affected air lock.

D.1 and D.2

If the inoperable containment air lock cannot be restored to OPERABLE status within the required 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 3 within 6 hours and to MODE 5 within 36 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 3.6.2.1 9 The acceptance criteria specified in the Containment Leakage Rate Testing Program for the air locks, limits airlock leakage to a small precentage of the combined Type B and C leakage limit.

Maintaining containment air locks OPERABLE requires compliance with the leakage rate test requirements of ~~10 CFR 50, Appendix J (Ref. 1), as modified by approved exemptions.~~ 4 the Containment Leakage Rate Testing Program. This SR reflects the leakage rate testing requirements with regard to air lock leakage (Type B leakage tests). ~~The acceptance criteria were established during initial air lock and containment OPERABILITY testing. The periodic testing requirements verify that the air lock leakage does not exceed the allowed fraction of the overall containment leakage rate.~~ The Frequency is required by ~~Appendix J (Ref. 1), as modified by approved exemptions. Thus, SR 3.0.2 (which allows Frequency extensions) does not apply.~~

The SR has been modified by two Notes. Note 1 states that an inoperable air lock door does not invalidate the previous successful performance of the overall air lock leakage test. This is considered reasonable since either air lock door is capable of providing a fission product barrier in the event of a DBA. Note 2 has been added to this SR requiring the results to be evaluated against the acceptance criteria ~~of~~ Approved TSTF-52 which is applicable to SR 3.6.1.1. This ensures that air lock leakage is properly

C RAI 3.6.1-5

BASES

APPLICABLE SAFETY ANALYSES

The DBA that results in a release of radioactive material within containment is a loss of coolant accident (Ref. 2). In the analysis of this accident, it is assumed that containment is OPERABLE such that release of fission products to the environment is controlled by the rate of containment leakage. The containment was designed with an allowable leakage rate of 0.4% of containment air weight per day (Ref. 2). This leakage rate is defined in 10 CFR 50, Appendix J, Option B (Ref. 1), as L_a = 0.4% of containment air weight per day, the maximum allowable containment leakage rate at the calculated peak design containment internal pressure, P_a of 60 psig, following a design basis LOCA. This allowable leakage rate forms the basis for the acceptance criteria imposed on the SRs associated with the air locks. C RAI 3.6.1-5

The containment air locks satisfy Criterion 3 of the NRC Policy Statement.

LCO

Each containment air lock forms part of the containment pressure boundary. As part of the containment pressure boundary, the air lock safety function is related to control of the containment leakage rate resulting from a DBA. Thus, each air lock's structural integrity and leak tightness are essential to the successful mitigation of such an event. C RAI 3.6.1-5

Each air lock is required to be OPERABLE. For the air lock to be considered OPERABLE, the air lock interlock mechanism must be OPERABLE, the air lock must be in compliance with the Type B air lock leakage test, and both air lock bulkheads must be OPERABLE. The interlock allows only one air lock door and its associated equalization valve of an air lock to be opened at one time. This provision ensures that a gross breach of containment does not exist when containment is required to be OPERABLE. The OPERABILITY of a single bulkhead (e.g., bulkhead door, door seals, equalization valve, interlock shaft seals, etc;) in each air lock is sufficient to provide a leak tight barrier following postulated events. Nevertheless, both doors and their associated equalization valves are kept closed when the air lock is not being used for normal entry into or exit from containment. C RAI 3.6.1-5

APPLICABILITY

In MODES 1, 2, 3, and 4, a design basis LOCA could cause a release of radioactive material to containment. In MODES 5 and 6, the probability and consequences of a design basis LOCA are reduced due to the pressure and temperature limitations of these MODES. Therefore, the containment air locks are not required in MODE 5 to prevent leakage of radioactive material from containment as a result of a design basis LOCA. The requirements for the containment air locks during MODE 6 are addressed in LCO 3.9.3, "Containment Penetrations."

BASES

ACTIONS (continued)

OPERABLE status, assuming that at least one door and its associated equalization valve are maintained closed in each affected air lock.

D.1 and D.2

If the inoperable containment air lock cannot be restored to OPERABLE status within the required 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 3 within 6 hours and to MODE 5 within 36 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 3.6.2.1

Maintaining containment air locks OPERABLE requires compliance with the leakage rate test requirements of the Containment Leakage Rate Testing Program. This SR reflects the leakage rate testing requirements with regard to air lock leakage (Type B leakage tests). The acceptance criteria specified in the Containment Leakage Rate Testing Program for the air locks, limits airlock leakage to a small percentage of the combined Type B and C leakage limit.

The Frequency is required by the Containment Leakage Rate Testing Program.

The SR has been modified by two Notes. Note 1 states that an inoperable air lock door does not invalidate the previous successful performance of the overall air lock leakage test. This is considered reasonable since either air lock door is capable of providing a fission product barrier in the event of a DBA. Note 2 has been added to this SR requiring the results to be evaluated against the acceptance criteria which is applicable to SR 3.6.1.1. This ensures that air lock leakage is properly accounted for in determining the combined Type B and C containment leakage rate. C RAI 3.6.1-5

SR 3.6.2.2

The bulkhead doors and equalization valves are interlocked with each other to prevent simultaneous opening of the doors and or equalizing valves in the redundant bulkheads. Since both the inner and outer bulkheads of an air lock are designed to withstand the maximum expected post accident containment pressure, OPERABILITY of either bulkhead will support containment OPERABILITY. Thus, the airlock

BASES

APPLICABLE SAFETY ANALYSES (Continued)

analyses of any event requiring isolation of containment is applicable to this LCO.

The DBAs that result in a release of radioactive material within containment are a loss of coolant accident (LOCA) and a rod ejection accident (Ref. 1). In the analyses for each of these accidents, it is assumed that containment isolation valves are either closed or function to close within the required isolation time following event initiation. This ensures that potential paths to the environment through containment isolation valves (including containment purge valves) are minimized. The safety analyses assume that the [42] inch purge valves are closed at event initiation.

The DBA analysis assumes that, within 60 seconds after the accident, isolation of the containment is complete and leakage terminated except for the design leakage rate, L a. The containment isolation total response time of 60 seconds includes signal delay, diesel generator startup (for loss of offsite power), and containment isolation valve stroke times.

[The single failure criterion required to be imposed in the conduct of plant safety analyses was considered in the original design of the containment purge valves. Two valves in series on each purge line provide assurance that both the supply and exhaust lines could be isolated even if a single failure occurred. The inboard and outboard isolation valves on each line are provided with diverse power sources, motor operated and pneumatically operated spring closed, respectively. This arrangement was designed to preclude common mode failures from disabling both valves on a purge line.]

[The purge valves may be unable to close in the environment following a LOCA. Therefore, each of the purge valves is required to remain sealed closed during MODES 1, 2, 3, and 4. In this case, the single failure criterion remains applicable to the containment purge valves due to failure in the control circuit associated with each valve. Again, the purge system valve design precludes a single failure from compromising the containment boundary as long as the system is operated in accordance with the subject LCO.]

1

BASES

ACTIONS (Continued)

is only applicable to penetration flow paths with two containment isolation valves. Condition A of this LCO addresses the condition of one containment isolation valve inoperable in this type of penetration flow path.

C.1 and C.2

Approved TSTF 30

With one or more penetration flow paths with one containment isolation valve inoperable, the inoperable valve flow path must be restored to OPERABLE status or the affected penetration flow path must be isolated. The method of isolation must include the use of at least one isolation barrier that cannot be adversely affected by a single active failure. Isolation barriers that meet this criterion are a closed and de-activated automatic valve, a closed manual valve, and a blind flange. A check valve may not be used to isolate the affected penetration flow path. Required Action C.1 must be completed within the [4] hour Completion Time. The specified time period is reasonable considering the relative stability of the closed system (hence, reliability) to act as a penetration isolation boundary and the relative importance of maintaining containment integrity during MODES 1, 2, 3, and 4. In the event the affected penetration flow path is isolated in accordance with Required Action C.1, the affected penetration flow path must be verified to be isolated on a periodic basis. This periodic verification is necessary to assure leak tightness of containment and that containment penetrations requiring isolation following an accident are isolated. The Completion Time of once per 31 days for verifying that each affected penetration flow path is isolated is appropriate because the valves are operated under administrative controls and the probability of their misalignment is low.

72

5

Insert B3.6.3-01

Approved TSTF 30

The closed system must meet the requirements of Ref 2.

Condition C is modified by a Note indicating that this Condition is only applicable to those penetration flow paths with only one containment isolation valve and a closed system. This Note is necessary since this Condition is written to specifically address those penetration flow paths in a closed system.

C

which utilize closed systems as one of the two containment barrier.

two

Approved TSTF 269

Required Action C.2 is modified by a Note that applies to valves and blind flanges located in high radiation areas and

s. Note 1

21

BASES

APPLICABLE SAFETY ANALYSES

The containment isolation valve LCO was derived from the assumptions related to minimizing the loss of reactor coolant inventory and establishing the containment boundary during major accidents. As part of the containment boundary, containment isolation valve OPERABILITY supports leak tightness of the containment. Therefore, the safety analyses of any event requiring isolation of containment is applicable to this LCO.

The DBA that results in a release of radioactive material within containment is a LOCA (Ref. 1). In the analyses for this accident, it is assumed that containment isolation valves are either closed or capable of closure to isolate penetrations. This ensures that potential paths to the environment through containment isolation valves (including containment purge valves) are minimized. The safety analyses assume that the purge supply and exhaust valves are closed at event initiation. C RAI 3.6.3-5 B RAI 3.6.3-4

No specific containment isolation time was assumed in the LOCA analysis. However, containment isolation is an implicit assumption in maintaining containment leakage within it's design leakage rate, L_a, and containment back pressure relative to RCS blowdown rate.

The purge valves may be unable to close in the environment following a LOCA. Therefore, each of the purge valves is required to remain closed with its control switch locked in the closed position during MODES 1, 2, 3, and 4. The single failure criterion is still applicable to the containment purge valves due to the potential for a failure in the control circuit associated with each valve. However, the purge system valve design precludes a single failure from compromising the containment boundary as long as the system is operated in accordance with the subject LCO.

The containment isolation valves satisfy Criterion 3 of the NRC Policy Statement.

LCO

Containment isolation valves form a part of the containment boundary. The containment isolation valves' safety function is related to minimizing the loss of reactor coolant inventory and establishing the containment boundary during a DBA.

The automatic power operated isolation valves are required to actuate to the closed position on an automatic isolation signal. The containment purge supply and exhaust valves must be maintained closed with their control switches in the locked closed position. The valves covered by this LCO are listed in the FSAR (Ref. 2).

BASES

ACTIONS (continued) C.1 and C.2

With one or more penetration flow paths with one containment isolation valve inoperable, the inoperable valve flow path must be restored to OPERABLE status or the affected penetration flow path must be isolated. The method of isolation must include the use of at least one isolation barrier that cannot be adversely affected by a single active failure. Isolation barriers that meet this criterion are a closed and de-activated automatic valve, a closed manual valve, and a blind flange. A check valve may not be used to isolate the affected penetration flow path. Required Action C.1 must be completed within the 72 hour Completion Time. The specified time period is reasonable considering the relative stability of the closed system (hence, reliability) to act as a penetration isolation boundary and the relative importance of maintaining containment integrity during MODES 1, 2, 3, and 4. In the event the affected penetration flow path is isolated in accordance with Required Action C.1, the affected penetration flow path must be verified to be isolated on a periodic basis. This periodic verification is necessary to assure leak tightness of containment and that containment penetrations requiring isolation following an accident are isolated. This SR does not apply to valves that are locked, sealed, or otherwise secured in the closed position, since these were verified to be in the correct position upon locking, sealing, or securing.

Condition C is modified by a Note indicating that this Condition is only applicable to those penetration flow paths with only one containment isolation valve and a closed system. The closed system must meet the requirements of Ref 2. This Note is necessary since this Condition is written to specifically address those penetration flow paths which utilize closed systems as one of the two containment barrier. C

Required Action C.2 is modified by two Notes. Note 1 applies to valves and blind flanges located in high radiation areas and allows these devices to be verified closed by use of administrative means. Allowing verification by administrative means is considered acceptable, since access to these areas is typically restricted. Note 2 applies to isolation devices that are locked, sealed, or otherwise secured in position and allows these devices to be verified closed by administrative means. Allowing verification by administrative means is considered acceptable, since the function of locking, sealing, or securing components is to ensure that these devices are not inadvertently repositioned. Therefore, the probability of misalignment of these valves, once they have been verified to be in the proper position, is small.

Description of Changes - NUREG-1431 Section 3.06.04

16-Oct-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:
15.03.06.B	LCO 3.06.04
15.03.06.B.01	LCO 3.06.04 LCO 3.06.04 COND A LCO 3.06.04 COND A RA A.1
15.03.06.B.02	LCO 3.06.04 COND B
15.03.06.B.02.A	LCO 3.06.04 COND B RA B.1
15.03.06.B.02.B	LCO 3.06.04 COND B RA B.2

A.02 Rev. C	Not used.

CTS:	ITS:
N/A	N/A

A.03 Rev. A	The Bases of the current Technical Specifications for this section have been completely replaced by revised Bases that reflect the format and applicable content of PBNP ITS Chapter 3.6, 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 3.06.04 B 3.06.04 B 3.06.04 B 3.06.04 B 3.06.04 B 3.06.04 B 3.06.04 B 3.06.04

Description of Changes - NUREG-1431 Section 3.06.04

16-Oct-00

DOC Number	DOC Text	
A.04 Rev. A	The CTS requires a channel check of the containment pressure instruments on a shiftly basis, which relative to this LCO has been concluded to be equivalent to ITS SR 3.6.4.1. SR 3.6.4.1 verifies containment pressure is within limits every 12 hours. The containment pressure instruments are also inputs to reactor protection and safeguard logic which is addressed in Section 3.3 of the ITS conversion package. Relative to this LCO these channels are only used to verify that containment pressure is within its required range. A channel check as discussed in CTS Section 15.4.1 is intended to be a simple observation of instrument function, which is fulfilled through verification of containment pressure by observation of other available channels. Performance of the proposed ITS surveillances while stated to verify operational limits still encompasses an observation of required channel function while clarifying the intended containment pressure check. This change is administrative.	
	CTS:	**ITS:**
	15.04.01 T 15.04.01-01 27	SR 3.06.04.01
A.05 Rev. C	Not used.	
	CTS:	**ITS:**
	N/A	N/A
L.01 Rev. A	The CTS required frequency for performance of the containment pressure verification is "once per shift", while the proposed frequency of performance for the ITS is every 12 hours. Verifying that containment pressure is within limits ensures that unit operation remains within the limits established to ensure that containment design pressures are not exceeded. The 12 hour Frequency of this SR was developed based on operating experience related to trending of containment pressure variations during the applicable MODES. Furthermore, the 12 hour Frequency is considered adequate in view of other indications available in the control room, including alarms, to alert the operator to an abnormal containment pressure condition.	
	CTS:	**ITS:**
	15.04.01 T 15.04.01-01 27	SR 3.06.04.01

Description of Changes - NUREG-1431 Section 3.06.04

16-Oct-00

DOC Number	DOC Text
M.01 Rev. C	The CTS does not specifically state a Mode of Applicability for containment internal pressure, but does however provide actions which ultimately require the unit to be placed into cold shutdown if the LCO is not met. By having actions that place the unit in cold shutdown, an implied applicability of Modes 1, 2, 3, and 4 (ITS Modes) can be derived. However, the APPLICABILITY for containment pressure is controlled in the CTS by CTS 15.3.6 APPLICABILITY, 15.3.6.A.1 and Table 15.4.1-1, Item 27. The combination of CTS 15.3.6 APPLICABILITY and 15.3.6.A.1 would imply that the APPLICABILITY for internal pressure would be all plant conditions except the COLD SHUTDOWN and REFUELING SHUTDOWN conditions. An Applicability of Modes 1, 2, 3, and 4 is consistent with the accident analysis assumptions, as a design basis loss of coolant accident in these modes could cause a release of radioactive materials to the containment. Since maintaining containment pressure within limits is essential to maintaining containment integrity and to ensure initial conditions assumed in the accident analyses are maintained, an applicability of MODES 1, 2, 3 and 4 has been established. This change is consistent with the accident analyses, corrects an interpretive discrepancy in the CTS, and is more restrictive. **CTS:** 15.03.06.B.02.B **ITS:** LCO 3.06.04
M.02 Rev. C	The plant condition required column of CTS Table 15.4.1-1 for line item 27 states that verification of containment pressure is required in all plant conditions. However, the appropriate Mode of Applicability for the containment pressure limit contained in the CTS has been determined to be ITS Modes 1, 2, 3, and 4 as provided in Description of Change M.1 to this Section. CTS 15.4.0.1 states that surveillance requirements shall be met when the system or component is required to be operable. By applying Specification 15.4.0.1, the CTS required mode of performance for this surveillance as related to verification of the containment pressure limit should also be ITS Modes 1, 2, 3, and 4. ITS SR 3.0.1 establishes the requirement that surveillances must be met when the LCO is applicable. As such, the ITS mode of performance for this surveillance is consistent with the safety analyses, corrects an interpretive discrepancy, and is more restrictive. **CTS:** 15.04.01 T 15.04.01-01 27 **ITS:** LCO 3.06.04

B. Internal Pressure A-1

B RAI 3.6.4-3

Cond A and RA

1. If the internal pressure exceeds 3 psig or the internal vacuum exceeds 2.0 psig, the condition shall be corrected within one hour.

Cond B and RA

2. If the above action cannot be completed within the time specified, place the affected unit in:
   a. hot shutdown within six hours, AND
   b. cold shutdown within 36 hours.

Add LCO Applicability - Modes 1, 2, 3, and 4 M.1 C RAI 3.6.4-1

C. Positive reactivity changes shall not be made by rod drive motion when the containment integrity is not intact except for the testing of one bank of rods at a time, rod disconnecting, and rod reconnecting provided the reactor is initially subcritical by at least 5% Δk/k.

D. Positive reactivity changes shall not be made by boron dilution when the containment integrity is not intact unless the boron concentration in the reactor is maintained > 2100 ppm*.

* 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 LCO 3.6.1>

B
RAI 3.6.4-3

A.1

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 Section 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	A	Q	ALL	
30.	DELETED					
31.	In-Core Thermocouples	M	R(14)	-	ALL	
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	< See LCO 3.4.11>
34.	PORV Operability	-	R	Q(11)	ALL	
35.	PORV Position Indicator	S(21)	R	R	ALL	

A.4 L.1 M.2 SR 3.6.4.1

C
RAI 3.6.4-1

No Significant Hazards Considerations - NUREG-1431 Section 3.06.04

16-Oct-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 3.06.04

16-Oct-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 extends the surveillance frequency for containment pressure verification from "each shift" (nominally 8 hours) to 12 hours. This relaxation is acceptable based on operating experience related to trending of containment pressure variations during the applicable MODES. Furthermore, the 12 hour Frequency is considered adequate in view of other indications available in the control room, including alarms, to alert the operator to an abnormal containment pressure condition. Therefore, this change does not significantly increase the probability or consequences of an accident previously evaluated. 2. Does the change create the possibility of a new or different kind of accident from any accident previously evaluated? The proposed change does not involve any physical alteration of plant systems, structures or components, nor does it alter parameters governing normal plant operation. The proposed change does not introduce a new mode of operation or alter the method of normal plant operation. Therefore, the possibility of a new or different kind of accident from any accident previously evaluated is not created. 3. Does this change involve a significant reduction in a margin of safety? There are no margins of safety related to safety analyses that are dependent upon the proposed change. The requirements will continue to assure that limiting conditions for the containment are properly maintained. Therefore, this change does not involve a significant reduction in a margin of safety.

No Significant Hazards Considerations - NUREG-1431 Section 3.06.04

16-Oct-00

NSHC Number	NSHC Text
M Rev. C	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.

DOC Number	DOC Text
L.03 Rev. A	CTS 15.4.5.I.B.3 requires the containment spray nozzles to be checked to ensure they are not obstructed at intervals not exceeding five years. The proposed ITS (SR 3.6.6.8) will require performances of this test once every 10 years, plus the 25% surveillance frequency extension allowed through application of SR 3.0.2 (a maximum of 12.5 years). This increase in frequency is considered acceptable based on the passive nature of the components. The containment spray nozzles are located near the top of the containment dome, in an area not subject to damage from personnel nor other components and debris. The containment spray nozzles are configured as "dry piping" and accordingly, are not subject to a harsh environment (contact with acids, caustics or other chemicals) during normal operation which could introduce significant age related degradation.

CTS:	ITS:
15.04.05.I.B.03	SR 3.06.06.09

DOC Number	DOC Text
L.04 Rev. B	CTS 15.4.5.I.B.1 and CTS 15.4.5.I.B.2 provides details on surveillance testing which are not necessary to describe the actual regulatory requirement. The requirement to run the pumps for at least 15 minutes in accordance with CTS 15.4.5.II.A.2 is an arbitrary requirement with no fundamental safety basis. Therefore, these details are being removed. The proposed ITS specifies the safety objective that must be fulfilled by the surveillance tests, while leaving the details associated with testing methods and acceptance verifications to licensee control. These type of details are better suited for procedural control and are not required to be in the ITS to provide adequate protection to the public health and safety. Changes to plant procedures and other plant controlled documents are subject to controls imposed by plant administrative procedures, which endorse applicable regulations and standards.

CTS:	ITS:
15.04.05.I.B.01	DELETED
15.04.05.I.B.02	DELETED
15.04.05.II.A.02	DELETED

DOC Number	DOC Text
L.05 Rev. C	CTS 15.4.5.I.B.1 requires the Containment Spray System test to be initiated by tripping the normal actuation instrumentation. The proposed ITS requirement in SR 3.6.6.5 and SR 3.6.6.6 allow initiation by an actual or simulated signal. The proposed ITS is less restrictive because it allows either a simulated or an actual signal. This change is insignificant because the actuation instrumentation for this system is appropriately surveilled in accordance with the requirements in Section 3.3 of the proposed ITS.

CTS:	ITS:
15.04.05.I.B.01	DELETED

JFD Number	JFD Text
26 Rev. A	The containment cooler service water outlet isolation valves have been added to SR 3.6.6.5, as an automatic valve, which is actuation tested every 18 months. The Point Beach design consists of four fan cooler units cooled by service water which is routed downstream of the fan coolers units into a single discharge header containing two cooling water motor operated valves which open upon receipt of a safety injection signal to increase cooling water flow to greater than or equal to analysis values. Only one outlet isolation valve is required to function to provide 100% flow to all four fan cooler units. Actuation testing of these valves is required every refueling outage by CTS 15.4.5.l.c.1. Operation of these valve is assumed for containment cooler operability, and has therefore been retained in the ITS. **ITS:** SR 3.06.06.05 **NUREG:** SR 3.06.06A.05
27 Rev. C	The Bases has been modified to reflect the means by which Point Beach verifies containment accident fan cooler unit accident fan operation. The Bases of NUREG 1431 SR 3.6.6.2 states that operation of the containment cooling train fan unit on a 31 day frequency ensures that all associated controls are functioning properly, and that blockage, fan or motor failure, and excessive vibration can be detected. The containment accident fan cooler unit accident mode fans do not have any associated controls which are verified through performance of this surveillance. Proper fan operation is verified through verification of the main control board run lights, clearing of low flow alarms, and verification of running current. This change is consistent with the acceptance criteria specified in the current Technical Specifications and is necessary to reflect the Point Beach design. **ITS:** B 3.06.06 **NUREG:** B 3.06.06

SURVEILLANCE REQUIREMENTS (continued)

Containment Spray System operation. This SR does not apply to valves that are locked, sealed, or otherwise secured in position, since these were verified to be in the correct position prior to locking, sealing, or securing. This SR does not require any testing or valve manipulation. Rather, it involves verification, through a system walkdown, that those valves outside containment (only check valves are inside containment) and capable of potent ially being mispositioned are in the correct position.

SR 3.6.6~~A~~2

Operating each ~~[required]~~ containment cooling ~~train~~ fan ~~unit~~ ~~for ≥ 15 minutes~~ ensures that all ~~trains~~ are OPERABLE and that all associated ~~controls~~ are functioning properly. It also ensures that blockage, fan or motor failure, ~~or~~ ~~excessive vibration~~ can be detected for corrective action. The 31 day Frequency was developed considering the known reliability of the ~~fan units~~ and ~~controls~~, the ~~two train~~ redundancy available, and the low probability of significant degradation of the ~~containment cooling train~~ occurring between surveillances. It has also been shown to be acceptable through operating experience.

Insert B 3.6.6-14

SR 3.6.6~~A~~3

Verifying that each ~~[required]~~ containment ~~cooling train~~ ESW ~~cooling flow rate to each cooling unit is ≥ [700] gpm~~ provides assurance that the design flow rate assumed in the safety analyses will be achieved (Ref. 3). The Frequency was developed considering the known reliability of the Cooling Water System, the ~~two train~~ redundancy available, and the low probability of a significant degradation of flow occurring between surveillances.

RAI 3.6.6-9

SR 3.6.6~~A~~4

Verifying each containment spray pump's developed head at the flow test point is greater than or equal to the required developed head ensures that spray pump performance has not degraded during the cycle. Flow and differential pressure are normal tests of centrifugal pump performance required by Section XI of the ASME Code (Ref. ~~8~~). Since the containment

RAI 3.6.6-13

BASES

ACTIONS (continued)

D.1

With one containment cooler service water outlet valve inoperable, the containment cooling water outlet valve must be restored to OPERABLE status within 72 hours. During this period, the remaining containment cooler service water outlet valve is capable of providing 100% of assumed cooling water flow to all four containment accident fan coolers. The 72 hour Completion Time was developed taking into account the auto open and flow capability afforded by the redundant cooling water outlet valve, and the low probability of DBA occurring during this period. B RAI 3.6.6-3

E.1 and E.2

If the Required Action and associated Completion Time of Condition C or D of this LCO are not met, the plant must be brought to a MODE in which the LCO does not apply. To achieve this status, the plant must be brought to at least MODE 3 within 6 hours and to MODE 5 within 36 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 3.6.6.1

Verifying the correct alignment for manual, power operated, and automatic valves in the containment spray flow path provides assurance that the proper flow paths will exist for Containment Spray System operation. This SR does not apply to valves that are locked, sealed, or otherwise secured in position, since these were verified to be in the correct position prior to locking, sealing, or securing. This SR does not require any testing or valve manipulation. Rather, it involves verification, through a system walkdown, that those valves outside containment (only check valves are inside containment) and capable of potentially being mispositioned are in the correct position.

SR 3.6.6.2

Operating each containment cooling unit's accident fan ensures that all accident fans are OPERABLE and that all associated indications are functioning properly. It also ensures that blockage, fan or motor failure, can be detected for corrective action. Acceptable performance is verified through verification of main control panel accident fan run indication, motor running amps, and clearing of low flow alarms. The 31 day Frequency was developed considering the known reliability of the accident fans and indications, the redundancy available, and the C RAI 3.6.6-9 C RAI 3.6.6-9

BASES

SURVEILLANCE REQUIREMENTS (continued)

low probability of significant degradation of the accident fans occurring between surveillances. It has also been shown to be acceptable through operating experience.

SR 3.6.6.3

Verifying that each containment accident fan cooler unit can achieve its assumed post accident flow rate with at least one containment accident fan cooler service water outlet valve open provides assurance that the design flow rate assumed in the safety analyses will be achieved (Ref. 3). The Frequency was developed considering the known reliability of the Cooling Water System, the redundancy available, and the low probability of a significant degradation of flow occurring between surveillances.

SR 3.6.6.4

Verifying each containment spray pump's developed head at the flow test point is greater than or equal to the required developed head ensures that spray pump performance has not degraded during the cycle. Flow and differential pressure are normal tests of centrifugal pump performance required by Section XI of the ASME Code (Ref. 4). Since the containment spray pumps cannot be tested with flow through the spray headers, they are tested on recirculation flow. This test confirms one point on the pump design curve and is indicative of overall performance. Such inservice tests confirm component OPERABILITY, trend performance, and detect incipient failures by abnormal performance. The Frequency of the SR is in accordance with the Inservice Testing Program. B RAI 3.6.6-13

SR 3.6.6.5 and SR 3.6.6.6

These SRs require verification that each automatic containment spray and containment accident fan cooler service water outlet valve actuates to its correct position and that each containment spray pump starts upon receipt of an actual or simulated actuation of a containment Hi-Hi pressure signal. This Surveillance is not required for valves that are locked, sealed, or otherwise secured in the required position under administrative controls. The 18 month Frequency is based on the need to perform these Surveillances under the conditions that apply during a plant outage and the potential for an unplanned transient if the Surveillances were performed with the reactor at power. Operating experience has shown that these components usually pass the Surveillances when performed at the 18 month Frequency. Therefore, the Frequency was concluded to be acceptable from a reliability standpoint.

DOC Number	DOC Text	
A.04 Rev. A	The CTS 15.3.3.B.1 requires the Iodine Removal System to be operable prior to the reactor being made critical. However, CTS 15.3.3.B.2 requires the unit to be placed into Hot Shutdown (ITS Mode 3) within 6 hours and Cold Shutdown (ITS Mode 5) within 36 hours, if this system is inoperable in excess of the allowable outage time, implying an Applicability of Modes 1, 2, 3, and 4 (ITS Modes). Proposed LCO 3.6.7 will require the Spray Additive System to be operable in Modes 1, 2, 3, and 4. This change is considered administrative as it is clarifying an ambiguous relationship between the LCO Applicability and Action Statement.	
	CTS:	**ITS:**
	15.03.03.B.01	LCO 3.06.07
A.05 Rev. C	Not used.	
	CTS:	**ITS:**
	N/A	N/A
A.06 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 3.06.07
A.07 Rev. A	CTS 15.4.5.I.b.2 Requires the performance of a system test during reactor shutdowns for major fuel reloadings. The CTS defines system test as being an actuation test, for which the only components in the spray additive system that receive an actuation signal are the spray additive tank outlet valves. Proposed ITS SR 3.6.7.4 requires verification that each automatic valve in the spray additive system that is not secured in its required position be actuated to its correct position on an actual or simulated actuation signal once every 18 months. This change is administrative, revising the CTS surveillance to a format and wording consistent with that used in NUREG 1431. The change in proposed frequency in addressed is Description of Change M.3 of this section.	
	CTS:	**ITS:**
	15.04.05.I.B.01	SR 3.06.07.04
A.08 Rev. A	CTS 15.3.3.B.1.a specifies that the spray additive tank shall contain sodium hydroxide with a minimum concentration of 30% by weight. This limitation has been moved to ITS surveillance requirement SR 3.6.7.3. Moving this limitation to SR 3.6.7.3 is administrative. An upper limit has been proposed for inclusion into this SR as discussed in Description of Change M.2 of this section.	
	CTS:	**ITS:**
	15.03.03.B.01.A	SR 3.06.07.03

DOC Number	DOC Text	
A.09 Rev. B	Not used.	
	CTS:	**ITS:**
	N/A	N/A
A.10 Rev. B	CTS 15.3.3.B.1.a specifies the spray additive tank level and NaOH concentration in the tank. If either of these limits are not met, CTS 15.3.0.B requires action to be initiated within 1 hour to place the affected unit in hot shutdown within the next 6 hours and cold shutdown within 36 hours. These required actions will be reflected in ITS 3.6.7, Conditions B and C, with the exception of allowing 84 hours to reach MODE 5 as discussed in DOC L.2.	
	CTS:	**ITS:**
	15.03.0.B	LCO 3.06.07 COND B LCO 3.06.07 COND B RA B.1 LCO 3.06.07 COND C LCO 3.06.07 COND C RA C.1 LCO 3.06.07 COND C RA C.2

DOC Number	DOC Text
L.05 Rev. C	CTS 15.4.5.I.B.1 requires the Spray Additive System test to be initiated by tripping the normal actuation instrumentation. The proposed ITS requirement in SR 3.6.7.4 allows initiation by an actual or simulated signal. The proposed ITS is less restrictive because it allows either a simulated or an actual signal. This change is insignificant because the actuation instrumentation for this system is appropriately surveilled in accordance with the requirements in Section 3.3 of the proposed ITS. **CTS:** 15.04.05.I.B.01 **ITS:** DELETED
LA.01 Rev. C	CTS 15.3.3.B.1.d establishes a requirement to maintain all valves and piping "associated" with the Iodine Removal System "and required to function during accident conditions" to be operable. This requirement is subsumed by the LCO statement, "The spray additive system shall be OPERABLE." Application of this concept is addressed through the definition of OPERABILITY, which requires all equipment required for the system to perform its specified safety function to be capable of performing their related support function. Additionally, the specifics defining OPERABILITY have been relocated to ITS 3.6.7 Bases - LCO. Valves are addressed through the valve testing requirements specified in the proposed ITS SR 3.6.7.8 and the Inservice Testing Program (Specification 5.5.8). **CTS:** 15.03.03.B.01.D **ITS:** LCO 3.06.07
M.01 Rev. A	CTS 15.3.3.B.1 contains a provision exempting the requirement to maintain the Iodine Removal System operable during low power physics testing. This provision has been deleted in the proposed Technical Specifications. Low power physics testing in the Improved Technical Specifications is a subset of Mode 2. While Mode 2 is typically a non limiting Mode, the operability requirements of this system is independent of physics testing, accordingly this provision has been deleted. This change represent a more restrictive changes as it involves the deletion of a flexibility that currently exists. **CTS:** 15.03.03.B.01 **ITS:** DELETED
M.02 Rev. A	CTS 15.3.3.B.1.a establishes the operational limits for the spray additive tank as being; not less than 2675 gallons in volume, and not less than 30% in concentration. The spray additive system is designed to establish a post Design Basis primary side Loss of Coolant Accident containment recirculation fluid pH of between approximately 7.0 and 9.0. This range is intended to minimize the evolution of iodines from the recirculation fluid as well as minimizing the potential for chloride and caustic stress corrosion. To maintain a pH range of approximately 7.0 to 9.0 an upper limit for concentration have been proposed. The addition of this limit will provide assurance that the upper pH limit is not exceeded. The addition of this limit is a more restrictive requirement. **CTS:** 15.03.03.B.01.A **ITS:** SR 3.06.07.03

A.1

B. Containment Cooling and Iodine Removal Systems

M.1 M.2

1. A reactor shall not be made critical, ~~except for low temperature physics tests,~~ unless the following conditions associated with that reactor are met:

M.4 A.4

SR 3.6.7.2 A.9

a. The spray additive tank contains not less than 2675 gal. of solution with a sodium hydroxide concentration of not less than 30% by weight. and ≤ 33 %

SR 3.6.7.3 A.8

b. Two containment spray pumps are operable.

c. Four accident fan-cooler units are operable.

LA.1

d. All valves and piping, associated with the above components and required to function during accident conditions, are operable.

C RAI 3.6.7-2

L.1

2. ~~During power operation, the requirements of 15.3.3.B-1 may be modified to allow any ONE of the following limitations (a., b., or c.) to be in effect at any one time.~~

If the Required Action and completion Time of Condition A or B are not met

~~If the system is not restored to meet the requirements of 15.3.3.B-1 within the time period specified,~~ the reactor shall be placed in the hot shutdown condition within six hours and in cold shutdown within ~~36~~ 84 hours. L.2

B RAI 3.6.7-6

A.10

a. One or two accident fan cooler may be out of service provided the fan coolers are restored to operable status within 72 hours. The remaining accident fan coolers shall be operable.

b. One containment spray pump may be out of service provided the pump is restored to operable status within 72 hours. The remaining containment spray pump shall be operable.

One Spray Additive System flowpath

c. Any valve required for the functioning of the system during accident conditions may be inoperable provided repairs are completed within 72 hours. ~~Prior to initiating repairs, all valves in the system that provide the duplicate function shall be operable.~~ (Exception: If a spray pump is removed from service per b. above, valves associated with that train may be removed from service for the period specified for the pump.)

L.1

A.10

<See LCO 3.6.6>

B RAI 3.6.7-6

B. Spray Additive System inoperable for any reason other than Condition A.	B.1	Restore at least one Spray Additive System flowpath to OPERABLE status.	1 hour

No Significant Hazards Considerations - NUREG-1431 Section 3.06.07

16-Oct-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. 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 proposed change will allow the spray additive system to be inoperable concurrent with the containment fan coolers or containment spray train, in addition to addressing a loss of redundancy for the spray additive system. Inoperability of the spray additive system concurrent with the containment fan cooler units is acceptable based on the fact that these two systems perform functions which are not interrelated. The spray additive system is required to promote retention of iodines in the recirculation fluids after a Loss of Coolant Accident (LOCA), in addition to long term containment corrosion considerations. Sodium hydroxide is added to the containment spray flow stream for reduction of containment iodine. The containment fan coolers are designed to maintain containment pressure and temperature within limits, the containment fan coolers and the spray additive system have no functional relationships nor dependencies. The containment spray system provides containment pressure and temperature control in addition to delivery of sodium hydroxide to the containment to maximize the absorption of iodines from the containment atmosphere and minimize the evolution of iodines from the containment recirculation fluids. Based on the system design, the loss of a containment spray train and spray additive flowpath within the same train, independent or concurrently results in the same level of degradation relative to the spray additive function. Additionally, an inoperable spray additive system flowpath results in the same level of degradation as an inoperable redundant valve. 1. Does the change involve a significant increase in the probability or consequences of an accident previously evaluated? This change does not result in the introduction of any new or different equipment. Through not introducing any new failure modes and mechanisms, this change would not result in a significant change in the probability of previously evaluated accidents. The consequences of previously evaluated accidents are not significantly altered by allowing multiple inoperabilities to exist. As discussed above, the allowable inoperabilities either result in the same level of degradation as a single inoperability, or are in unrelated functions. The allowable plant configurations will continue to be bounded by the existing containment pressure analysis. Accordingly, the consequences of previously evaluated accidents are not significantly changed. 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 allow operation for a limited period of time with multiple inoperabilities, while still bounded by the existing 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?

No Significant Hazards Considerations - NUREG-1431 Section 3.06.07

16-Oct-00

NSHC Number	NSHC Text
	The allowable combination of inoperabilities involve equipment which does not result in any increase in risk state or are associated with unrelated functions which do not have any interdependencies. Based on this, the potential for common mode failure within redundant components during the increased time allowed for overlapping inoperabilities is insignificant. In this fashion, the margin inherent to redundant systems and components is not significantly impacted by the small increase in allowable restoration time. Considering the low probability of coincident entry into multiple Conditions with the low probability of an accident occurring during this time, the margin of safety is not significantly reduced.

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? This change does not result in any equipment or hardware changes. The spray additive systems allowable restoration time is not assumed to be an initiators of any analyzed event. The proposed change extends the allowable time to reach Mode 5 after the unit is placed into Mode 3 by 48 hours. During this added 48 hours relative to multiple inoperabilities, the consequences of an event will continue be bounded by the existing containment pressure analysis. Loss of functional capability is acceptable based on the absence of an iodine re-evolution mechanism over the pH range of concern. Secondarily, any re-evolution should be offset by the conservatisms used in the offsite and onsite dose calculations relative to containment leakage rates. Accordingly, the consequences of previously evaluated accidents are not significantly changed. Therefore, the proposed change does not increase the probability or consequences of an accident previously evaluated. 2. Does the change create the possibility of a new or different kind of accident from any accident previously evaluated? The proposed change does not 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 allow continuous operation with an inoperable containment spray train. 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 increased time allowed to reach Mode 5 is acceptable based on the allowable combinations of inoperabilities involving equipment which does not result in any increase in risk state or are associated with unrelated functions which do not have an interdependencies. In addition, this additional time is acceptable based on the conservatisms inherent to the unit being placed in Mode 3. Dose considerations (both offsite and control room) are projected based on a core operating at 102% of rated power and the containment pressure analysis is based upon a higher energy state (temperature) for the reactor coolant system. The reduced consequences from these specifics alone offset the increased time allowed to operate in a condition capable of event mitigation, but incapable of a single failure. Loss of functional capability for the spray additive function does not result in any significant changes in onsite or offsite doses. This is based on conservative assumption made relative to containment leakage rate, and the lack of a significant driver which would result in re-evolution of iodines back into the containment atmosphere over the containment sump pH range of concern. Considering the low probability of coincident entry into multiple Conditions or loss of functional capability with the low probability of an accident occurring during this time, an increase in the

NSHC Number	NSHC Text
	allowable time to reach Mode 5 does not significantly affect any margin of safety.
L.03 Rev. A	In accordance with the criteria set forth in 10 CFR 50.92, PBNP has evaluated this proposed Technical Specifications change and determined it does not represent a significant hazards consideration. The following is provided in support of this conclusion. 1. Does the change involve a significant increase in the probability or consequences of an accident previously evaluated? This change does not result in any equipment or hardware changes. The proposed change extends the spray additive tank sodium hydroxide sampling frequency from once every month to once every 184 days. There are no permanently connected fill or drain lines; therefore, this tank is not subject to rapid or uncontrolled changes in level and concentration. The frequency of surveillance testing is not an initiator of any analyzed event. This increase in frequency is acceptable based on the static nature of the tank. Further, the proposed frequency is acceptable based on industry data, which supports that the proposed frequency is adequate in providing assurance that tank concentration will be maintained thereby, maintaining the equipment in an operable state. Based on the equipment being maintained in an operable state, the consequence for previously evaluated accidents remains unchanged. Accordingly, the probability and consequences of previously evaluated accident is not significantly changed. 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 spray additive tank is normally static, it is not used as a process tank, and there are no permanently connected fill or drain lines, therefore this tank is not subject to rapid or uncontrolled changes in level and concentration. Intentional changes to tank level and concentration are performed in a controlled manner and will include post evolution sampling when necessary. Based on the above, it has been concluded that increasing the testing interval will not result in any significant increase in undetectable surveillance failures. 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 increased surveillance interval is acceptable based on the industry data that has concluded that the likelihood of a concentration change is low based on the static nature of the tank. The likelihood for an uncontrolled chemistry change is insignificant, and it has been concluded that sodium hydroxide concentration does not significantly change due to aging. Based on the above, this change does not represent 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. 1. Does the change involve a significant increase in the probability or consequences of an accident previously evaluated? The proposed change does not involve any physical alteration of plant systems, structures or components, changes in parameters governing normal plant operation, or methods of operation. The proposed change results in the deletion of details which are not necessary to describe the actual regulatory requirement, or provide adequate protection of the public health and safety. Accordingly, there will be no significant change in the probability or consequences of accidents previously evaluated. 2. Does the change create the possibility of a new or different kind of accident from any accident previously evaluated? The proposed change does not involve any physical alteration of plant systems, structures or components, nor does it alter parameters governing normal plant operation. The proposed change does not introduce a new mode of operation. Therefore, the possibility of a new or different kind of accident from any accident previously evaluated is not created. 3. Does this change involve a significant reduction in a margin of safety? The deletion of details which are not necessary to describe the actual regulatory requirement, or provide adequate protection of the public health and safety, does not result in a significant reduction in the margin of safety.

NSHC Number	NSHC Text
L.05 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. CTS 15.4.5.I.B.1 specifies the Spray Additive System test to be initiated by tripping the normal actuation instrumentation. ITS SR 3.6.7.4 permits initiation by an actual or simulated signal to satisfy the requirements. 1. Does the change involve a significant increase in the probability or consequences of an accident previously evaluated? The Spray Additive System is used to mitigate the consequences of an accident; however, it is not an initiator of any previously analyzed accident. As such the relaxing the requirements under which the Spray Additive System testing is performed does not affect the results of the surveillance and will not increase the probability of any accident previously evaluated. The proposed actions continue to provide adequate assurance of Operability for required equipment and therefore, do not involve an increase in the consequences of any accident previously evaluated. 2. Does the change create the possibility of a new or different kind of accident from any accident previously evaluated? The proposed change does not 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. 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? This change does not involve a significant reduction in a margin of safety because the Operability of the equipment continues to be evaluated in the same manner. The results of the Spray Additive System testing are not affected by the nature of the initiating signal, because the system cannot discriminate whether the signals are actual or simulated. The intent of the surveillance requirement has not been altered and does not result in a reduction in the margin of safety.

NSHC Number	NSHC Text
LA Rev. C	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.

Description of Changes - NUREG-1431 Section 3.07.04

16-Oct-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:
15.03.04	LCO 3.07.04
15.03.04.A.05	LCO 3.07.04 LCO 3.07.04 COND C LCO 3.07.04 COND C RA C.1

DOC Number	DOC Text
A.02 Rev. A	The CTS provides an introductory statement (Applicability) which simply states which systems/components are addressed within a given section. This same information, while worded differently, is contained within the title of each ITS LCO. Accordingly, this change is a change in format with no change in technical requirement.

CTS:	ITS:
15.03.04 APPL	LCO 3.07.04

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

CTS:	ITS:
15.03.04 OBJ	B 3.07.04

DOC Number	DOC Text
A.04 Rev. A	The Bases of the current Technical Specifications for this section have been completely replaced by revised Bases that reflect the format and applicable content of PBNP ITS, consistent with the Standard Technical Specifications for Westinghouse Plants, NUREG-1431. The revised Bases are as shown in the PBNP ITS Bases.

CTS:	ITS:
BASES	B 3.07.04

DOC Number	DOC Text
L.01 Rev. A	CTS 15.3.4.A requires the Steam Generator Atmospheric Dump Valves (ADVs) to be operable prior to the reactor being made critical. CTS requirement 15.3.4.A.5 requires the unit to be placed into Hot shutdown within 6 hours and Cold Shutdown within 36 hours if an inoperable ADV is not restored to operable status within the time allotted in the Technical Specifications, implying a Mode of Applicability of ITS Modes 1, 2, 3, and 4. The proposed ITS establishes a Mode of Applicability for the ADVs and their associated block valve of Mode 1, 2, 3, and Mode 4 when the Steam Generators are relied upon for heat removal. In Modes 1, 2, and 3, and Mode 4 when the Steam Generators are relied upon for heat removal, the ADVs are required to be operable to provide the capability to cool the unit down to RHR entry conditions whenever the condenser steam dump valves are not available. In addition, in Modes 1, 2, and 3, the ADVs are utilized to cool the unit down to maintain RCS subcooling in response to a Steam Generator Tube Rupture coincident with a loss of offsite power. In MODE 4 when the steam generators are not relied upon for heat removal, the residual heat removal system is operable and in operation providing decay heat removal. In addition, the RCS and steam generator temperatures have been reduced to a temperature sufficiently below the saturation pressure corresponding to the steam generator safety valves lift setpoints, precluding radiological releases to the environs as a result of a SGTR. In MODE 5 or 6, an SGTR is not a credible event. Based on a Mode of Applicability of 1, 2, 3, and Mode 4 when the Steam Generators are relied upon for heat removal, the default Actions for LCO non-compliance have been revised to require the unit to be placed into Mode 3 within 6 hours and Mode 4 without reliance upon the steam generators for heat removal within 18 hours. These time frames are reasonable, based on operating experience, to reach the required unit conditions from full power conditions in an orderly manner and without challenging unit systems. These proposed changes clarify an ambiguous LCO, and Required Action, however this change constitutes a relaxation in the current Mode of Applicability. The proposed Mode of Applicability and Required Actions are consistent with analysis assumptions for Point Beach.

CTS:	ITS:
15.03.04.A	LCO 3.07.04
15.03.04.A.05	LCO 3.07.04 COND C RA C.2

DOC Number	DOC Text
L.02 Rev. C	CTS 15.3.4.A.5 allows 24 hours to restore one inoperable ADV. The proposed change allows 48 hours for one ADV and 24 hours for two ADVs inoperable. Prior to 1994, the ADVs were not subject to LCO requirements in the Point Beach Technical Specifications. The ADV LCO requirements were added in 1994 based on a review of the FSAR accident analysis requirements that discuss the use of the ADVs for mitigation of the SG Tube Rupture and Main Steam Line Break accidents. When the decision was made to add LCO requirements for ADVs into the PBNP Technical Specifications, NUREG 1431 was used as guidance for creation of the ADV LCO requirements. The NUREG 1431 requirements for ADVs is based on a 4-loop RCS. This does not correspond to PBNP which has a 2-loop RCS. The judgement was made to adopt the basic requirement of NUREG 1431 which requires an ADV to be restored to operable status within 24 hours if only one is operable. This condition would occur for PBNP if either ADV was inoperable, because PBNP has only two ADVs. This requirement is overly restrictive for PBNP. Therefore, a 48 hour completion time is proposed for the first ADV and 24 hours for the second. Based on risk insights from the Point Beach Probabilistic Risk Assessment (PRA) model, the information below is provided in support of the extensions to the allowed outage times (AOT). Two different methods were used to evaluate the risk impact of a 7-day AOT for an inoperable ADV flowpath. The results from both methods show that a 7-day AOT will have minimal risk impact. (A 48 hour AOT has been chosen to provide additional margin and is bounded by the above analysis.) The Point Beach PRA model used for this evaluation only includes internal events with the reactor at power. External events such as fires, floods, earthquakes, high winds, etc. and events with the reactor shutdown are not included in the numerical results. However, due to the limited use of ADVs to mitigate these events, having an ADV unavailable for maintenance should not impact external event or shutdown risk to any significant degree. The first method of determining risk change evaluated the impact on average core damage risk. This was done by adding a test and maintenance unavailability to the random failure probability for both of the ADVs. The PRA model was then requantified including these two new values for ADV unavailability, and the resulting core damage frequency (CDF) was compared to the base CDF to determine a delta CDF. A delta CDF of less than 1E-06/yr indicates a minimal increase in risk. This criteria is consistent with that used in NRC Regulatory Guide 1.174, "An Approach for Using Probabilistic Risk Assessment In Risk-Informed Decisions On Plant-Specific Changes to the Licensing Basis." The new failure probability for the ADVs was developed first by adding a new test and maintenance unavailability to the random failure probability: (7 days / 300 days) + 4.0E-03 = 2.73E-02 This assumes one AOT period per valve per year, and 300 days of reactor operation (82% availability) per year. A higher reactor availability makes this estimated failure probability conservative. Quantifying the PRA model with these new failure probabilities yields a CDF of 4.623 E-05/yr. Given a base CDF of 4.571E-05/yr, the delta CDF is then: 4.623E-05/yr - 4.571E-05/yr = 5.2E-07/yr

DOC Number	DOC Text

This result is less than 1E-06/yr, which indicates that the risk increase for a 7-day AOT is small. Note that this method overestimates the actual risk increase because it includes the random probability of having both ADVs unavailable due to maintenance.

The second method of determining risk change evaluated the temporary risk increase due to having one of the ADVs (MS 2015, MS 2016) out of service. The first step determined the change in instantaneous risk due to having an ADV out of service. This change in core damage frequency per year was then multiplied by the fraction of a year that we anticipate to experience the temporary increase in risk, to arrive at a delta core damage probability (delta CDP). The temporary risk increase is considered to be of low risk significance if the delta CDP is less than 1E-06. This is consistent with the criteria in EPRI TR-105396, "PSA Applications Guide."

Quantifying the PRA model with each of the ADV's failure probability set to 1.0, one at a time, gave the following results:

For MS 2015 failed, delta CDF = 5.994E-05/yr - 4.571E-05/yr = 1.42E-05/yr
For MS 2016 failed, delta CDF = 4.696E-05/yr - 4.571E-05/yr = 1.25E-06/yr

The difference between the delta CDFs for the two valves is due to asymmetries in the PRA model; the value for MS 2015 will be used because it yields a more conservative result.

The fraction of the year for the temporary risk increase is (7 days / 300 days/yr) = 2.33E-02 yr. Multiplying this by the delta CDF will give a delta CDP: 1.42E-05/yr * 2.33E-02 yr = 3.32E-07

This change in core damage probability is lower than the 1E-06 criteria; therefore, the temporary risk increase is of low risk significance.

For both ADV flowpaths inoperable concurrently (for which a 24-hour AOT is proposed), the core damage frequency risk increases to 3.431E-04/yr. Our base CDF value is 4.571E-05, yielding a CDF increase of 2.974E-04/yr. This CDF increase coupled with the delta core damage probability increase criteria for a temporary change of 1.0E-06 would justify a 29-hour AOT. Therefore, the proposed 24-hour AOT for Condition B (both ADV flowpaths inoperable) is acceptable.

CTS:	ITS:
15.03.04.A.05	LCO 3.07.04 COND A
	LCO 3.07.04 COND A RA A.1
	LCO 3.07.04 COND A RA A.1 NOTE
	LCO 3.07.04 COND B
	LCO 3.07.04 COND B RA B.1

Description of Changes - NUREG-1431 Section 3.07.04

16-Oct-00

DOC Number	DOC Text	
LB.01 Rev. A	CTS 15.4.1, Table 15.4.1-2, Item 28 requires a complete cycle of the Steam Generator Atmospheric Dump Valves (ADVs) once per quarter. The ADVs at Point Beach are Class II components, and as such are required to be tested per ASME Section XI in accordance with 10 CFR 50.55a. Since this testing is duplicative of the ASME required tests, it can be removed from the Technical Specifications while remaining to be applicable to Point Beach. As such, this test is not required to be in the ITS to provide adequate protection of public health and safety.	
	CTS:	**ITS:**
	15.04.01 T 15.04.01-02 28	IST
M.01 Rev. A	The CTS does not contain any testing requirements which verify that the ADVs and their associated block valves are capable of being locally operated. Local operation of these valves should be verified on a periodic basis, as local operation is the assumed mode of operation relative to a Steam Generator Tube Rupture with a loss of offsite power. The proposed ITS will require local operation of the ADVs and their associate block valves to be verified on an 18 month frequency. This frequency is acceptable based on engineering judgment and the inherent reliability of manual actuators.	
	CTS:	**ITS:**
	NEW	SR 3.07.04.01 SR 3.07.04.02

< See LCOs 3.7.5/3.7.6 >

3. A minimum of 13,000 gallons of water per operating unit in the condensate storage tanks and an unlimited water supply from the lake via either leg of the plant Service Water System.

4. System piping and valves required to function during accident conditions directly associated with the above components operable.

Cond B/ RA B.1

L.2 → both

LCO 3.7.4

5. Both atmospheric steam dump lines shall be operable. If either of the atmospheric steam dump lines is determined to be inoperable, restore the inoperable line to an operable status within 24 hours. If operability cannot be restored, be in hot shutdown within six hours and cold shutdown within 24 hours.

Cond C/ RA C.1/ RA C.2

B. The dose equivalent iodine-131 activity on the secondary side of the steam generator shall not exceed 1.0 µCi/g.

< See LCO 3.7.18 >

L.1 → Be in Mode 4 without reliance upon the Steam Generators for heat removal - 18 hours

C. During power operation the requirements of 15.3.4.A.2.a and b may be modified to allow the following components to be inoperable for a specified time. If the system is not restored to meet the requirements of 15.3.4.A.2.a and b within the time period specified, the specified action must be taken. If the requirements of 15.3.4.A.2.a and b are not satisfied within an additional 48 hours, the appropriate reactor(s) shall be cooled down to less than 350°F.

1. Two Unit Operation - One of the four operable auxiliary feedwater pumps may be out-of-service for the below specified times. A turbine driven auxiliary feedwater pump may be out of service for up to 72 hours. If the turbine driven auxiliary feedwater pump cannot be restored to service within the 72 hour time period the associated reactor shall be in hot shutdown within the next 12 hours. A motor driven auxiliary feedwater pump may be out of service for up to 7 days. If the inoperable motor driven auxiliary feedwater pump cannot be restored to service within the 7 day time period both of the reactors shall be in hot shutdown within the next 12 hours.

< See LCO 3.7.5 >

L.2 → ADD CONDITION A: If one ADV flowpath is inoperable, restore the inoperable flowpath to operable status within 48 hours.

C RAI 3.7.4-1

Justification For Deviations - NUREG-1431 Section 3.07.04

16-Oct-00

JFD Number	JFD Text	
01 Rev. A	Brackets have been removed and the appropriate plant specific information has be input.	
	ITS:	**NUREG:**
	B 3.07.04	B 3.07.04
	LCO 3.07.04	LCO 3.07.04
	LCO 3.07.04 COND C RA C.2	LCO 3.07.04 COND C RA C.2
02 Rev. A	Point Beach has two ADV Lines, one per steam generator, therefore, NUREG 1431 section 3.7.4 has been modified accordingly.	
	ITS:	**NUREG:**
	B 3.07.04	B 3.07.04
	LCO 3.07.04 COND B	LCO 3.07.04 COND B
	LCO 3.07.04 COND B RA B.1	LCO 3.07.04 COND B RA B.1

JFD Number	JFD Text
03 Rev. C	The ADVs are air operated fail closed valves, with the capability to be remotely opened and closed. Local manual operation of the ADVs is credited during a Steam Generator Tube Rupture (SGTR) event coincident with a Loss of Offsite Power (LOOP). The ADVs are ASME Class II valves, which are required by 10 CFR 50.55a to be tested in accordance with ASME Section XI. However, this testing does not encompass local manual operation. Proposed SR 3.7.4.1 will require local manual testing of the ADVs, with or without steam flow, at an 18 month frequency. In June, 1996, a satisfactory demonstration of the ability to manually operate the ADVs from the local station with steam flow was performed. This one time test, in conjunction with the ASME Section XI operation of the ADVs using the air operator and proposed SR 3.7.4.1, will verify the capability to manually operate the ADVs locally during a SGTR/LOOP event. The 18 month testing frequency proposed for ADV local manual operation is adequate based on the engineering judgement that the failure of the ability to manually operate these valves is highly improbable. The ADV block valves are only credited with isolation of a failed open ADV. The ADV block valves are not credited for re-establishing ADV flow for the mitigation of a SGTR/LOOP event. If it is necessary to close an ADV block valve to isolate a failed open ADV, that ADV flowpath will be considered inoperable. SR 3.7.4.2 which proposes to manually exercise the ADV block valves at an 18 month frequency, with or without steam flow, is sufficient to ensure its capability to isolate a failed open ADV. **ITS:** / **NUREG:** B 3.07.04 / B 3.07.04 SR 3.07.04.01 / SR 3.07.04.01 SR 3.07.04.02 / SR 3.07.04.02
04 Rev. A	The normal source of water for the Auxiliary Feedwater System (AFW) is the condensate storage tank; however, the safety related water supply is from the service water system. The Bases have been rewritten to address this as Point Beach's design basis. **ITS:** / **NUREG:** B 3.07.04 / B 3.07.04

Justification For Deviations - NUREG-1431 Section 3.07.04

16-Oct-00

JFD Number	JFD Text	
05 Rev. A	Point Beach's ADVs are air operated fail closed valves, with the capability to be remotely opened and closed. Motive air to the ADV is from the non-safety related instrument air system, with no backup nitrogen or accumulators. The Bases has therefore omitted all discussion related to backup nitrogen and air accumulators. The ADVs are considered operable when they are capable of being locally stroked. Failure of the instrument air system is accounted for via local manual operation.	
	ITS:	**NUREG:**
	B 3.07.04	B 3.07.04
06 Rev. A	Point Beach's ADVs are considered operable when they are capable of local manual operation. The Bases have been modified to reflect this as Point Beach's design basis.	
	ITS:	**NUREG:**
	B 3.07.04	B 3.07.04
07 Rev. A	As discussed in Justification For Deviation 2 of this LCO, Point Beach has one ADV per steam generator, rendering the plant incapable of sustaining a single failure of an ADV in the unaffected steam generator during a steam generator tube rupture coincident with a loss of condenser steam dump capabilities. Bases statement relating to single failure criteria has therefore been omitted.	
	ITS:	**NUREG:**
	B 3.07.04	B 3.07.04
08 Rev. A	The Bases for the Applicability associated with LCO 3.7.4 (the ADVs), does not provide any discussion of why the ADVs are not required to be operable when the steam generators are not relied upon for heat removal. This discussion has been added for completeness.	
	ITS:	**NUREG:**
	B 3.07.04	B 3.07.04
09 Rev. A	Point Beach's ADV block valves are not power operated valves. These valve are manually operated, and as such do not fall under ASME Section XI relative to surveillance testing. Accordingly, reference to ASME testing in the Bases of SR 3.7.4.2 has been changed to reflect Point Beach's design.	
	ITS:	**NUREG:**
	B 3.07.04	B 3.07.04
10 Rev. A	Automatic actuation of the ADVs is not relied upon for the mitigation of any analyzed events for Point Beach. Therefore reference to automatic operation of the ADVs has been deleted.	
	ITS:	**NUREG:**
	B 3.07.04	B 3.07.04

Justification For Deviations - NUREG-1431 Section 3.07.04

16-Oct-00

JFD Number	JFD Text	
11 Rev. A	FSAR Chapter 14 has been added as reference 2 for the bases of section 3.7.4. FSAR Chapter 14 is the appropriate Point Beach accident analysis reference.	
	ITS:	**NUREG:**
	B 3.07.04	B 3.07.04
12 Rev. A	"Atmospheric Dump Valves (ADV) Lines" has been changed to "Atmospheric Dump Valve (ADV) Flowpaths", to reflect the nomenclature currently used at Point Beach.	
	ITS:	**NUREG:**
	B 3.07.04	B 3.07.04
	LCO 3.07.04	LCO 3.07.04
	LCO 3.07.04 COND A	LCO 3.07.04 COND A
	LCO 3.07.04 COND A RA A.1	LCO 3.07.04 COND A RA A.1
	LCO 3.07.04 COND B	LCO 3.07.04 COND B
13 Rev. A	An ADV block valve can be used to mitigate a failed open ADV. The ADV block valves are not used to mitigate a failed closed ADV. Accordingly, the LCO 3.7.4 Bases discussion of the ADV block valves has been modified to reflect this distinction.	
	ITS:	**NUREG:**
	B 3.07.04	B 3.07.04
14 Rev. C	NUREG-1431, Required Action A.1 and the associated Bases have been modified to allow 48 hours to restore an inoperable ADV to operable status. This change reflects risk insights from the Point Beach Probabilistic Risk Assessment (PRA) model. Two different methods were used to evaluate the risk impact of a 7-day AOT for an inoperable ADV flowpath. The results from both methods show that a 7-day AOT will have minimal risk impact. A 48 hour AOT has been chosen to provide additional margin and is bounded by the above analysis.	
	ITS:	**NUREG:**
	B 3.07.04	B 3.07.04
	LCO 3.07.04 COND A RA A.1	LCO 3.07.04 COND A RA A.1

ADVs Lines 3.7.4

12 → ADV Flowpaths

12 → Flowpaths

3.7 PLANT SYSTEMS

3.7.4 Atmospheric Dump Valve~~s~~ (ADV) Lines

1 → Two

LCO 3.7.4 [Three] ADV lines shall be OPERABLE.

APPLICABILITY: MODES 1, 2, and 3.
MODE 4 when steam generator is relied upon for heat removal.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One required ADV line inoperable. (12 → flowpath)	A.1 --------NOTE--------- LCO 3.0.4 is not applicable. ---------- Restore required ADV line to OPERABLE status. (flowpath ← 12)	7 days ← 48 hours (14) C RAI 3.7.4-3
B. Two ~~or more~~ required ADV lines inoperable. (2 → flowpaths; 12 → flowpaths)	B.1 Restore one ADV line to OPERABLE status.	24 hours
C. Required Action and associated Completion Time not met.	C.1 Be in MODE 3. AND C.2 Be in MODE 4 without reliance upon steam generator for heat removal.	6 hours; 18 ← 1; [18] hours

ADVs Lines
B 3.7.4
12 ADV Flowpaths

APPLICABLE SAFETY ANALYSES (continued)

4 or the Service Water System

cool the unit to RHR entry conditions with only one steam generator and one ADV, utilizing the cooling water supply available in the CST.

11 2

In the accident analysis presented in Reference ~~1~~ 1, the ADVs are assumed to be used by the operator to cool down the unit to RHR entry conditions for accidents accompanied by a loss of offsite power. Prior to operator actions to cool down the unit, the ~~ADVs and~~ 10 main steam safety valves (MSSVs) are assumed to operate automatically to relieve steam and maintain the steam generator pressure below the design value. For the recovery from a steam generator tube rupture (SGTR) event, the operator is ~~also~~ 10 required to perform a limited cooldown to establish adequate subcooling as a necessary step to terminate the primary to secondary break flow into the ruptured steam generator. The time required to terminate the primary to secondary break flow for an SGTR is more critical than the time required to cool down to RHR conditions for this event ~~and also for other accidents~~ 10. Thus, the SGTR is the limiting event for the ADVs. ~~The number of ADVs required to be OPERABLE to satisfy the SGTR accident analysis requirements depends upon the number of unit loops and consideration of any single failure assumptions regarding the failure of one ADV to open on demand.~~ 7

The ADVs are equipped with block valves in the event an ADV spuriously ~~fails to open or~~ 13 fails to close during use.

The ADVs satisfy Criterion 3 of the NRC Policy Statement.

LCO 1 two [Three] ADV lines (flowpaths) are required to be OPERABLE. One ADV line (12 flowpath) is required from each of [three] steam generators to ensure that at least one ADV line (12 flowpath) is available to conduct a unit cooldown following an SGTR, in which one steam generator becomes unavailable ~~, accompanied by a single, active failure of a second ADV line on an unaffected steam generator~~ 7. The block valves must be OPERABLE to isolate a failed open ADV line (12 flowpath). A closed block valve does not render it or its ADV line inoperable ~~if operator action time to open the block valve is supported in the accident analysis.~~ 3 renders

C

(continued)

RAI 3.7.4-1
RAI 3.7.4-2

ADVs Lines B 3.7.4

12 ADV Flowpaths

LCO (continued)

Failure to meet the LCO can result in the inability to cool the unit to RHR entry conditions following an event in which the condenser is unavailable for use with the Steam Bypass System.

An ADV is considered OPERABLE when it is capable of providing controlled relief of the main steam flow and capable of fully opening and closing on demand.

APPLICABILITY

In MODES 1, 2, and 3, and in MODE 4, when a steam generator is being relied upon for heat removal, the ADVs are required to be OPERABLE.

Insert 3.7.4-3

In MODE 5 or 6, an SGTR is not a credible event.

8

ACTIONS

is reasonable to repair an inoperable ADV flowpath, based on the availablility of

14

and the low probability of an event occurring during this period that would require the ADV flowpath.

A.1

12 flowpath

48 hours 14 48 hour

With one required ADV line inoperable, action must be taken to restore OPERABLE status within 7 days. The 7 day Completion Time allows for the redundant capability afforded by the remaining OPERABLE ADV lines, a nonsafety grade backup in the Steam Bypass System, and MSSVs. Required Action A.1 is modified by a Note indicating that LCO 3.0.4 does not apply.

C

RAI 3.7.4-1
RAI 3.7.4-2
RAI 3.7.4-3

B.1

2 flowpaths 12 flowpath

With two ~~or more~~ ADV lines inoperable, action must be taken to restore ~~all but~~ one ADV line to OPERABLE status. Since the block valve can be closed to isolate an ADV, some repairs may be possible with the unit at power. The 24 hour Completion Time is reasonable to repair inoperable ADV lines, based on the availability of the Steam Bypass System and MSSVs, and the low probability of an event occurring during this period that would require the ADV lines.

2 an

12 flowpath

C.1 and C.2

flowpaths 12 flowpath

If the ADV lines cannot be restored to OPERABLE status within the associated Completion Time, the unit must be

(continued)

LCO 3.7.4 Bases Inserts

Insert B 3.7.4-1:

when the ADVs are capable of being locally opened and closed. C RAI 3.7.4-1

Insert B 3.7.4-2:

To perform a controlled cooldown of the RCS, the ADVs must be able to be opened locally and throttled through their full range. This SR ensures that the ADVs are capable of being locally operated by cycling the valve, with or without steam flow, at least once per fuel cycle. This test is in addition to the ASME quarterly inservice test required by 10 CFR 50.55a. The Frequency is considered acceptable based on engineering judgement and reliability. C RAI 3.7.4-1

Insert B 3.7.4-3:

In MODE 4 when the steam generators are not relied upon for heat removal (residual heat removal system in operation), the RCS and steam generator temperatures have been reduced to a temperature sufficiently below the saturation pressure which corresponds to the steam generator safety valves lift setpoints to preclude radiological releases to the environs as a result of a SGTR.

Insert B 3.7.4-4:

The Frequency is considered acceptable based on engineering judgement and reliability.

3.7 PLANT SYSTEMS

3.7.4 Atmospheric Dump Valve (ADV) Flowpaths

LCO 3.7.4 Two ADV flowpaths shall be OPERABLE.

APPLICABILITY: MODES 1, 2, and 3,
MODE 4 when steam generator is relied upon for heat removal.

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. One required ADV flowpath inoperable.	A.1	----------NOTE---------- LCO 3.0.4 is not applicable. ---------- Restore required ADV flowpath to OPERABLE status.	48 hours C RAI 3.7.4-1
B. Two required ADV flowpaths inoperable.	B.1	Restore one ADV flowpath to OPERABLE status.	24 hours
C. Required Action and associated Completion Time not met.	C.1	Be in MODE 3.	6 hours
	AND		
	C.2	Be in MODE 4 without reliance upon steam generator for heat removal.	18 hours

B 3.7 PLANT SYSTEMS

B 3.7.4 Atmospheric Dump Valves (ADVs)

BASES

BACKGROUND

The ADVs provide a method for cooling the unit to residual heat removal (RHR) entry conditions should the preferred heat sink via the Steam Bypass System to the condenser not be available, as discussed in the FSAR, Section 10.1 (Ref. 1). This is done in conjunction with the Auxiliary Feedwater System providing cooling water from the condensate storage tank (CST) or the service water system. The ADVs may also be required to meet the design cooldown rate during a normal cooldown when steam pressure drops too low for maintenance of a vacuum in the condenser to permit use of the Steam Dump System.

One ADV flowpath for each of the two steam generators is provided. Each ADV flowpath consists of one ADV and an associated block valve.

The ADVs are provided with upstream block valves to permit their being tested at power, and to provide an alternate means of isolation. The ADVs are equipped with pneumatic controllers to permit control of the cooldown rate.

A description of the ADVs is found in Reference 1. The ADVs are OPERABLE when the ADVs are capable of being locally opened and closed. C RAI 3.7.4-1

APPLICABLE SAFETY ANALYSES

The design basis of the ADVs is established by the capability to cool the unit to RHR entry conditions. The design rate of approximately 50°F per hour is applicable for one steam generator. This rate is adequate to cool the unit to RHR entry conditions with only one steam generator and one ADV, utilizing the cooling water supply available in the CST or the service water system.

In the accident analysis presented in Reference 2, the ADVs are assumed to be used by the operator to cool down the unit to RHR entry conditions for accidents accompanied by a loss of offsite power. Prior to operator actions to cool down the unit, the main steam safety valves (MSSVs) are assumed to operate automatically to relieve steam and maintain the steam generator pressure below the design value. For the recovery from a steam generator tube rupture (SGTR) event, the operator is required to perform a limited cooldown to establish adequate subcooling as a necessary step to terminate the primary to secondary break flow into the ruptured steam generator. The time required to terminate the primary to secondary break flow for an SGTR is more critical than the time required to cool down to RHR conditions for this event. Thus, the SGTR is the limiting event for the ADVs.

BASES

APPLICABLE SAFETY ANALYSES (continued)

The ADVs are equipped with block valves in the event an ADV spuriously fails to close during use.

The ADVs satisfy Criterion 3 of the NRC Policy Statement.

LCO

Two ADV flowpaths are required to be OPERABLE. One ADV flowpath is required from each of two steam generators to ensure that at least one ADV flowpath is available to conduct a unit cooldown following an SGTR, in which one steam generator becomes unavailable. The block valves must be OPERABLE to isolate a failed open ADV flowpath. A closed block valve renders its ADV flowpath inoperable. C

RAI 3.7.4-1
RAI 3.7.4-2

Failure to meet the LCO can result in the inability to cool the unit to RHR entry conditions following an event in which the condenser is unavailable for use with the Steam Bypass System.

An ADV is considered OPERABLE when it is capable of providing controlled relief of the main steam flow and capable of fully opening and closing on demand.

APPLICABILITY

In MODES 1, 2, and 3, and in MODE 4, when a steam generator is being relied upon for heat removal, the ADVs are required to be OPERABLE.

In MODE 4 when the steam generators are not relied upon for heat removal (residual heat removal system in operation), the RCS and steam generator temperatures have been reduced to a temperature sufficiently below the saturation pressure which corresponds to the steam generator safety valves lift setpoints to preclude radiological releases to the environs as a result of a SGTR.

In MODE 5 or 6, an SGTR is not a credible event.

ACTIONS

A.1

With one required ADV flowpath inoperable, action must be taken to restore OPERABLE status within 48 hours. The 48 hour Completion Time Completion Time is reasonable to repair an inoperable ADV flowpath, based on the availability of a nonsafety grade backup in the Steam Bypass System, and MSSVs, and the low probability of an event occurring during this period that would require the ADV flowpath. Required Action A.1 is modified by a Note indicating that LCO 3.0.4 does not apply. C

RAI 3.7.4-1
RAI 3.7.4-2
RAI 3.7.4-3

BASES

ACTIONS (continued)

B.1

With two ADV flowpaths inoperable, action must be taken to restore one ADV flowpath to OPERABLE status. Since the block valve can be closed to isolate an ADV, some repairs may be possible with the unit at power. The 24 hour Completion Time is reasonable to repair an inoperable ADV flowpath, based on the availability of the Steam Bypass System and MSSVs, and the low probability of an event occurring during this period that would require the ADV flowpath.

C.1 and C.2

If the ADV flowpaths cannot be restored to OPERABLE status within the associated Completion Time, the unit must be placed in a MODE in which the LCO does not apply. To achieve this status, the unit must be placed in at least MODE 3 within 6 hours, and in MODE 4, without reliance upon steam generator for heat removal, within 18 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required unit conditions from full power conditions in an orderly manner and without challenging unit systems.

SURVEILLANCE REQUIREMENTS

SR 3.7.4.1

To perform a controlled cooldown of the RCS, the ADVs must be able to be opened locally and throttled through their full range. This SR ensures that the ADVs are capable of being locally operated by cycling the valve, with or without steam flow, at least once per fuel cycle. This test is in addition to the ASME quarterly inservice test required by 10 CFR 50.55a. The Frequency is considered acceptable based on engineering judgement and reliability. C RAI 3.7.4-1

SR 3.7.4.2

The function of the block valve is to isolate a failed open ADV. Cycling the block valve both closed and open, with or without steam flow, demonstrates its capability to perform this function. The Frequency is considered acceptable based on engineering judgement and reliability.

REFERENCES

1. FSAR. Section 10.1.
2. FSAR. Chapter 14.

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. Mode of Applicability for the ADVs has been changed to be consistent with the accident analysis assumptions. The ADVs are required to be operable to provide the capability to cool the unit down to RHR entry conditions whenever the condenser steam dump valves are not available. In addition, in Modes 1, 2, and 3, the ADVs are utilized to cool the unit down to maintain RCS subcooling in response to a Steam Generator Tube Rupture coincident with a loss of condenser steam dumps. Based on revising the Mode of Applicability to ITS Modes 1, 2, 3, and Mode 4 when the Steam Generators are relied upon for heat removal, the default Actions for LCO non-compliance have been revised to ultimately require the unit to be placed into Mode 4 without reliance upon the steam generators for heat removal. 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 Atmospheric Dump Valves (ADVs) are used to cool the unit down to Residual Heat Removal (RHR) entry condition during routine shutdowns and for recovery from Steam Generator Tube Rupture (SGTR), when the main condenser is not available. The probability for analyzed event (SGTR) and unit cooldowns are independent of the required mode of applicability for the ADVs. The proposed Mode of Applicability will provide assurance that the ADVs are operable when the ADVs are required to function in support of unit cooldown operations. The proposed Conditions and Required Actions will similarly, require the unit to be placed into a condition where the ADVs are not required to function in support of unit cooldowns. As such, the probability and consequences of previously analyzed event are not increased significantly. 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. This change makes the Mode of Applicability for the ADVs consistent with the current accident analyses assumptions. 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 change in applicability for ADVs is consistent with the assumptions made in the various Point Beach accident analyses. The ADVs will be maintained operable in accordance with the proposed ITS in the operational Modes and Conditions for which ADVs are required to function. In this fashion the margin of safety is not significantly changed.

NSHC Number	NSHC Text
L.02 Rev. C	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 Atmospheric Dump Valves (ADVs) are used to cool the unit down to Residual Heat Removal (RHR) entry condition during routine shutdowns and for recovery from Steam Generator Tube Rupture (SGTR) and Main Steam Line Break events, when the main condenser is not available. The probability for analyzed event (SGTR and MSLB) and unit cooldowns are independent of the number of operable ADVs. Therefore, the probability and consequences of previously analyzed events are not increased significantly. 2. Does the change create the possibility of a new or different kind of accident from any accident previously evaluated? New or different kinds of accidents can only be created by new or different accident initiators or sequences. 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. This proposed change does not create any new or different accident initiators or sequences because this change to the LCO condtions, action statements and allowable outage times for the ADVs does not create any different accident initiators or sequences. The PBNP emergency operating procedures contain guidance for mitigation of a SGTR and a MSLB for situations where the ADVs are not available. Therefore, this proposed Technical Specifications change does not create the possibility of an accident of a different type than any previously evaluated in the Point Beach FSAR. 3. Does this change involve a significant reduction in a margin of safety? The margins of safety for Point Beach are based on the design and operation of the reactor and containment and the safety systems that provide their protection. This change does not affect the design and operation of the reactor and containment. This change proposes to increase the allowed outage time for one ADV from 24 hours to 48 hours and to allow two ADVs to be inoperable for 24 hours. This proposed change does not significantly reduce any margin of safety, because other equipment, such as the condenser steam dump, can be used to mitigate SGTR and MSLB accidents if the ADVs are not able to be operated. Therefore, this proposed Technical Specifications change does not involve a significant reduction in a margin of safety because accident mitigation is still able to be achieved.

Description of Changes - NUREG-1431 Section 3.07.05

16-Oct-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:
15.03.04	LCO 3.07.05
15.03.04.C	LCO 3.07.05 COND D
15.04.01 T 15.04.01-01 20 (13)	SR 3.07.05.05
15.04.08	LCO 3.07.05

A.02 Rev. A	The CTS provides an introductory statement (Applicability) which simply states which systems/components are addressed within a given section. This same information, while worded differently, is contained within the title of each ITS LCO. Accordingly, this change is a change in format with no change in technical requirement.

CTS:	ITS:
15.03.04 APPL	LCO 3.07.05
15.04.08 APPL	LCO 3.07.05

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

CTS:	ITS:
15.03.04 OBJ	B 3.07.05
15.04.08 OBJ	B 3.07.05

A.04 Rev. A	The Bases of the current Technical Specifications for this section have been completely replaced by revised Bases that reflect the format and applicable content of PBNP ITS, consistent with the Standard Technical Specifications for Westinghouse Plants, NUREG-1431. The revised Bases are as shown in the PBNP ITS Bases.

CTS:	ITS:
BASES	B 3.07.05

Description of Changes - NUREG-1431 Section 3.07.05

16-Oct-00

DOC Number	DOC Text
A.05 Rev. A	CTS states that both motor driven auxiliary feedwater pumps, the turbine driven auxiliary feedwater pump, the flow paths, and essential instrumentation associated with these pumps are required to be operable. The ITS states that one turbine driven and two motor driven auxiliary feedwater pump systems are required to be operable. In specifying that the pump system must be operable, all of the elements contained within the CTS are addressed. The proposed ITS Surveillance Requirements contained in LCO 3.7.5 require periodic verification of the auxiliary feedwater pumps, flowpaths, and automatic start and alignment capabilities, while proposed LCO 3.3.2 addresses the required ESF instrumentation and actuation logic. In addition, through application of the ITS definition of Operability, the pump system and all of its associated support equipment must be capable of performing their specified safety functions. Accordingly, the ITS captures all of the elements specified within the CTS in a presentation consistent with NUREG 1431. This change is administrative. **CTS:** 15.03.04.A.02.B **ITS:** LCO 3.07.05
A.06 Rev. A	The CTS states that during power operation, the requirements of Specifications 15.3.4.A.2.a and b (i.e. pumps, piping, and essential instrumentation for single and two unit operation) may be modified to allow the auxiliary feedwater pumps to be inoperable for a limited period of time before requiring a unit shutdown. This Specification establishes the structure for the remedial actions in the CTS. The ITS contains specific usage rules for consistent application of the Conditions and Required Actions associated with varying system inoperabilities consistent with the format and presentation of NUREG 1431. Accordingly, deletion of a specific Specification directing usage of Actions is unnecessary, as it duplicates the ITS usage rules. This change is administrative. **CTS:** 15.03.04.C **ITS:** DELETED

DOC Number	DOC Text
A.07 Rev. C	CTS 15.4.8.1 requires each AFW pump to be started quarterly, however, if the test comes due for the turbine driven pump when the unit is not at power, the test is required to be performed within 24 hours of entering power operation. CTS 15.1.h defines "power operation" as the condition when the reactor is critical and the average neutron flux of the power range instrumentation indicates greater than 2 percent of rated power. Proposed SR 3.7.5.2 is modified by a note which states that performance of the pump test is not required for the turbine driven AFW pump until 24 hours after THERMAL POWER is greater than 2% RTP. Table 15.4.1-1, Note 13 requires completion of flow path verification prior to entering power operation (greater than 2% power) whenever the unit has been in cold shutdown for greater than 30 days. Proposed ITS SR 3.7.5.5 states that the required AFW flowpaths are to be verified prior to THERMAL POWER exceeding 2% RTP, whenever the unit has been in MODE 5, MODE 6, or defueled for a cumulative period of > 30 days. Therefore, changing the above frequencies from "within 24 hours of entering power operation" and "prior to entering power operation" to "24 hours after THERMAL POWER exceeds 2% RTP" and "prior to THERMAL POWER exceeding 2% RTP" is an administrative change.

CTS:	ITS:
15.04.01 T 15.04.01-01 20 (13)	SR 3.07.05.05
15.04.08.01.A	SR 3.07.05.02
15.04.08.01.B	SR 3.07.05.02 NOTE

L.01 Rev. A	CTS 15.3.4.C only provides actions that address the inoperability of the auxiliary feedwater (AFW) pumps. As such, piping, valve, and instrumentation inoperabilities which render a pump inoperable could be interpreted as requiring entry into CTS 15.3.0.B (similar to ITS LCO 3.0.3). The ITS addresses inoperability of the AFW pump systems (turbine and motor driven), thereby encompassing any component within a given pump system which could render a pump (pump system) incapable of performing its intended function. This change is acceptable because any component which renders a pump system inoperable is equivalent to the inoperability of the pump itself.

CTS:	ITS:
15.03.04.C.02	LCO 3.07.05 COND B LCO 3.07.05 COND B RA B.1 LCO 3.07.05 COND C LCO 3.07.05 COND C RA C.1

DOC Number	DOC Text
L.02 Rev. A	CTS 15.3.4.C.1 only provides Actions for a single inoperable auxiliary feedwater (AFW) pump during two unit operation. This Description of Change addresses the proposed ITS Action, which will allow an inoperable turbine driven AFW pump on each unit simultaneously during two unit operation. The inoperability of two or more AFW pump systems on the same unit is addressed by Description of Change M.2 of this Section. Each turbine driven AFW pump is dedicated to a unit and is capable of supplying 200% of the design AFW flow to both steam generators on its respective unit. Based on the turbine driven auxiliary feedwater pump being dedicated to a specific unit, an inoperability on one unit should impact that unit alone; however, the CTS only provides Actions for a single inoperable AFW pump during two unit operation, thereby requiring each unit to initiate the Actions of CTS 15.3.0.B. CTS 15.3.0.B requires both units to be placed into hot shutdown (equivalent to ITS Mode 3) within 7 hours, ultimately requiring at least one unit to be then cooled down to less than 350 degrees F before the Actions for a single unit operating can then be applied. Application of the single unit operating LCO then allows the operating unit to continue to operate for up to 72 hours from the time the AFW pump became inoperable prior to requiring the unit to be placed into hot shutdown (ITS Mode 3) in 12 hours and less than 350 degrees (ITS Mode 4) within 60 hours. The proposed ITS will allow a turbine driven AFW pump on each unit to be inoperable for up to 72 hours before requiring the affected units to be placed into Mode 3 within 6 hours and Mode 4 within 18 hours. Operation with a turbine driven AFW pump inoperable on each unit for up to 72 hours is reasonable to restore the pump to operable status before requiring a unit shutdown based on redundant capabilities afforded by the motor driven pump systems, a reasonable time to effect repairs, the low probability of a DBA occurring during this time period and the fact that the turbine driven pumps are dedicated to their respective unit, thereby, only affecting the unit that the pump system supplies. Requiring a unit to be shutdown based on the inoperability of opposite unit equipment is an unnecessary action. The opposite unit's turbine driven AFW pump is not credited to operate nor does it affect the risk or consequences to its complementary unit. Based on the availability of the motor driven AFW pumps, the accident analysis remains bounded for both units during the proposed Completion Time.

CTS:	ITS:
15.03.04.C.01	DELETED

DOC Number	DOC Text
L.03 Rev. A	Both turbine driven AFW pump steam supply lines are required to be operable to consider the turbine driven AFW pump system to be operable. Therefore, the inoperability of a steam supply line results in entry into the Actions for an inoperability of a turbine driven AFW pump, which allows up to 72 hours to restore the pump to operable status before requiring a unit shutdown. The proposed ITS will allow a single steam supply to be inoperable for up to 7 days before requiring the unit to be placed into Mode 3 within 6 hours and Mode 4 within 18 hours. The proposed Condition and Required Action represents a 96 hour extension of the allowable outage time for an inoperable turbine driven AFW pump steam supply. This extension is bounded by the current accident analysis and is acceptable based on the redundant capabilities provided by the remaining operable motor driven pump systems, and the low probability of an accident occurring during this time period which would affect the availability of the remaining steam supply. The Completion Time for this Action is limited to 7 days from entry into the Condition or 10 days from failure to meet the LCO, whichever is more restrictive. The proposed 10 day completion time limits the maximum time the LCO may be not met as a result of multiple overlapping Conditions.
	CTS: / **ITS:** NEW / LCO 3.07.05 COND A LCO 3.07.05 COND A LCO 3.07.05 COND A RA A.1
L.04 Rev. A	The current Technical Specifications require entry into LCO general requirement 15.3.0.B if the entire AFW system is inoperable. This is inappropriate because the actions for 15.3.0.B require that the affected unit be placed in hot shutdown within 7 hours. AFW is needed for decay heat removal when the unit is in hot shutdown. If the entire AFW system is inoperable the appropriate action would be to initiate action to restore AFW immediately. If this situation were to occur and the current Technical Specifications were applied, it is highly likely that Notice of Enforcement Discretion would be requested to avoid placing the plant in a condition in which AFW is needed for decay heat removal. Therefore, the proposed Technical Specification requirements for three AFW pump systems inoperable provides the appropriate required action for this condition and the proposed requirements are considered a substantial improvement over the current Technical Specifications requirements. The proposed condition and required action provide adequate protection of the public health and safety because the appropriate action has been established for the condition of inoperability of all three AFW pump systems.
	CTS: / **ITS:** NEW / LCO 3.07.05 COND E LCO 3.07.05 COND E RA E.1 LCO 3.07.05 COND E RA E.1 NOTE
L.05 Rev. C	Not used.
	CTS: / **ITS:** N/A / N/A

DOC Number	DOC Text
LA.01 Rev. A	The CTS contains separate Specifications and Required Actions for single and two unit operation. This structure clarifies the shared interrelationship of the motor driven AFW pumps, requiring both motor driven AFW pump systems to be operable whenever either unit is above 350 degrees F. When a motor driven AFW pump is inoperable, the CTS requires both units to be placed on a restoration time clock. The auxiliary feedwater (AFW) system consists of a total of four pumps; two motor driven auxiliary feedwater pump systems which are shared by both units, and one dedicated turbine driven pump per unit. Both motor driven AFW pumps are required to be operable to support one or two unit operation, while the turbine driven pumps are only required to support operation of their respective unit. The proposed ITS will require the turbine driven and two motor driven pump systems to be operable to support a unit in Modes 1, 2, 3, in addition to the motor driven pump systems supplying any steam generators relied upon for heat removal in Mode 4. The ITS is written to be applied on a unit specific basis. The LCO requirements are to be applied to each unit independently. Conditions and Required Actions are applicable to each affected unit as well. Based on application of the LCO to each unit independently, the number of pump systems required to be operable will remain the same, with the sharing of the motor driven pump systems addressed in the Bases. The number of shared components is a detail which is not necessary in the Technical Specification itself, as each unit is required to met its minimum operability requirement independent of the other. The shared interrelationship of the motor driven pump systems is a detail associated with system design and configuration, which are adequately addressed in the Bases and through the 10 CFR 50.59 process. These details are not required to be in the ITS to provide adequate protection of 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 Technical Specifications and the 10 CFR 50.59 process as applicable. Similarly, the Actions for inoperable AFW pumps are applicable to each affected unit, with the restoration time for a single inoperable motor or turbine driven AFW pump remaining the same. The Actions for multiple inoperable pumps are addressed in Description of Change L.2 (multiple inoperable turbine driven pumps on opposite units) and Description of Change M.2 (multiple inoperable pumps affecting the same unit).

CTS:	ITS:
15.03.04.A.02.A	DELETED
15.03.04.A.02.B	DELETED
15.03.04.C.01	LCO 3.07.05
15.03.04.C.02	DELETED

DOC Number	DOC Text
LA.02 Rev. A	The CTS states that the auxiliary feedwater system is required to have an unlimited water supply from the lake via either leg of the plant service water system, and that the piping and valves which are necessary for the auxiliary feedwater system to function during accident conditions are required. The ability to supply service water to the auxiliary feedwater pumps is verified via testing of the service water supply valves. The service water supply valves are ASME Class 3 components which are required to be tested in accordance with ASME Section XI by 10 CFR 50.55a. As such, while not specifically stated, service water suction supply valve testing will continue to be required in accordance with this regulatory requirement. The piping required to function during accident conditions is an attribute of system design and configuration, which is adequately captured through application of the definition of operability. As such, these details are not required to be in the ITS to provide adequate protection of public health and safety. These attributes are discussed within the Bases for the proposed Point Beach ITS, 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:
15.03.04.A.03	B 3.07.05
15.03.04.A.04	B 3.07.05

LA.03 Rev. A	The CTS provides acceptance criteria for AFW pump and valve operability tests, which simply requires satisfactory control board indication changes and visual observation of equipment to verify that it has operated satisfactorily. These acceptance limits are vague and non-prescriptive. ASME Section XI testing of AFW pumps and valves is required in accordance with 10 CFR 50.55a and specified in Section 5.0 of the ITS. The Inservice Testing Program contains specific acceptance criteria reflective of component performance capability. As such, the CTS details (observation of control board indication and visual observation of equipment) are not required in the ITS to provide adequate protection of public health and safety. The details and methods used to obtain equipment performance information is adequately controlled in Station procedures with the Technical Specifications and Regulations simply establishing a requirement to perform the testing. Changes to plant procedures and other plant controlled documents are subject to controls imposed by plant administrative procedures, which endorse applicable regulations and standards.

CTS:	ITS:
15.04.08.02	DELETED

LB.01 Rev. A	The CTS requires the auxiliary feedwater pump discharge valves and service water suction supply valves to be tested by operator action on a quarterly basis. These valves are ASME Class 3 valves and as such are required to be tested in accordance with ASME Section XI as required by 10 CFR 50.55a. The CTS frequency for valve testing (quarterly) is consistent with the ASME required frequency (once every 92 days). Accordingly, the testing of these valves is established and required by regulation without the need to duplicate these requirements in the Technical Specifications.

CTS:	ITS:
15.04.08.01.C	DELETED

DOC Number	DOC Text
M.01 Rev. A	CTS 15.3.4.C.2 requires the unit to be placed into hot shutdown (equivalent to ITS Mode 3) within 12 hours if a motor driven or turbine driven auxiliary feedwater (AFW) pump exceeds the allowable outage time (7 days and 72 hours respectively). Once the unit is placed into hot shutdown, the CTS allows an additional 48 hours before the unit must be cooled down to less than 350 degrees (equivalent to ITS Mode 4). As such, once the allowable outage time for an inoperable pump system has expired, the CTS will require the unit to be placed in ITS Mode 3 within 12 hours and ITS Mode 4 within 60 hours. For this same set of conditions, the ITS will require the unit to be placed into Mode 3 within 6 hours and Mode 4 within 18 hours. The proposed reduction in time frames allowed to reach Mode 3 and Mode 4 are more restrictive than the CTS, and are being made for consistency with NUREG 1431.

CTS:	ITS:
15.03.04.C	LCO 3.07.05 COND D RA D.2
15.03.04.C.02	LCO 3.07.05 COND D RA D.1

M.02 Rev. A	The CTS only provides Actions for a single inoperable auxiliary feedwater (AFW) pump during single and two unit operation. This Description of Change addresses the proposed ITS Action for simultaneous inoperability of two or more AFW pump systems. The simultaneous inoperability of both turbine driven AFW pumps during two unit operation is addressed by Description of Change L.2 of this LCO. Based on the CTS only containing Actions for a single inoperable AFW pump, the CTS would require entry into LCO 15.3.0.B if two motor driven AFW pump systems or a turbine and a motor driven pump system were inoperable simultaneously. CTS 15.3.0.B requires the unit to be placed into hot shutdown (equivalent to ITS Mode 3) within seven hours and cold shutdown (equivalent to ITS Mode 5) within 37 hours, but does not contain a time limit for achieving less than or equal to 350 degrees (ITS Mode 4). Accordingly, the CTS does not specify a time limit for when the reactor must be cooled to less than or equal to 350 degrees. The proposed ITS will require the unit to be placed into Mode 3 within 6 hours and Mode 4 within 18 hours when two AFW pump systems are inoperable simultaneously. The reduced time frame to achieve Mode 3 (7 hours to 6 hours) and the specific time frame to reach Mode 4 (18 hours) are more restrictive requirements. These time frames are consistent with the time frames specified in NUREG 1431.

CTS:	ITS:
15.03.04.C.02	DELETED
NEW	LCO 3.07.05 COND D LCO 3.07.05 COND D RA D.1 LCO 3.07.05 COND D RA D.2

DOC Number	DOC Text
M.03 Rev. A	The CTS does not contain a specific Condition to address multiple inoperable auxiliary feedwater (AFW) pumps. If multiple overlapping inoperability were to occur (e.g. alternating between an inoperable turbine driven and motor driven AFW pump), the CTS does not establish any limitation requiring LCO compliance to be re-established. The proposed ITS contains a Completion Time limit which requires restoration of LCO compliance within 10 days of first component becoming inoperable. The limit of 10 days is the summation of the longest and shortest Completion Times within this LCO and is consistent with NUREG 1431. The addition of this Completion time is an additional restriction not contained in the CTS.

CTS:	ITS:
15.03.04.C.02	LCO 3.07.05 COND B RA B.1 LCO 3.07.05 COND C RA C.1

M.04 Rev. A	The proposed ITS has added three new surveillances to verify alignment, automatic pump start, and automatic valve realignment capabilities in support of system operability. The addition of these tests will provide added assurance of AFW system operability, by testing assumed functions. Proposed SR 3.7.5.1 requires performance of a 31 day surveillance to verify valves that are not locked sealed or otherwise secured in position are in their required positions. Proposed SR 3.7.5.3 and SR 3.7.5.4 verify AFW pump automatic start and automatic valve realignment capabilities. These tests are not required in Mode 4 due to reduced heat removal requirements and there being adequate time for operator action to manually start and align the AFW pump systems if necessary.

CTS:	ITS:
NEW	SR 3.07.05.01 SR 3.07.05.03 SR 3.07.05.03 NOTE SR 3.07.05.04 SR 3.07.05.04 NOTE 1 SR 3.07.05.04 NOTE 2

DOC Number	DOC Text
M.05 Rev. A	The CTS requires the auxiliary feedwater (AFW) system to be operable whenever reactor coolant temperature is greater than 350 degrees (equivalent to ITS Modes 1, 2, and 3). The proposed ITS will continue to require the AFW system to be operable in Modes 1, 2, and 3, while adding a requirement to maintain the motor driven AFW pumps associated with steam generators required for decay heat removal in accordance with proposed ITS LCO 3.4.6. Inclusion of this Applicability, ensures the capability to provide make up water to steam generator(s) relied upon for decay heat removal. In keeping with the proposed Applicability, the ITS also contain a Required Action to address the loss of one or both motor driven AFW pumps systems in Mode 4. The Action proposed is consistent with those required in proposed ITS LCO 3.4.6 for loss of the steam generators as a heat sink, requiring initiation of action to restore the AFW pump system to operable status.

CTS:	ITS:
15.03.04.A	LCO 3.07.05
NEW	LCO 3.07.05 NOTE LCO 3.07.05 COND F LCO 3.07.05 COND F RA F.1

M.06 Rev. A	CTS 15.4.8.1 requires the motor and turbine driven auxiliary feedwater (AFW) pumps to be tested periodically, only requiring that the pumps be started and verified to be running satisfactorily. The AFW pumps are ASME Class 3 components which are required to be tested per 10 CFR 50.55a in accordance with the ASME Section XI testing program (the Inservice Testing Program). The ITS requires verification that the AFW pumps will develop their required head at the flow test point when tested at a frequency in accordance with the Inservice Testing Program. As such, the ITS frequency of testing will continue to be the same as stated in Description of Change A.7 of this Section. Inclusion of a requirement to verify that the developed pump head is above the required pump head is a new Technical Specifications acceptance criteria, not contained in the CTS. As such, verification of this limit is an additional restriction placed on pump testing in accordance with NUREG 1431. This change is more restrictive.

CTS:	ITS:
15.04.08.01.A	SR 3.07.05.02

CTS INSERTS

Insert 3.7.5-1:

SURVEILLANCE		FREQUENCY
SR 3.7.5.2	----------NOTE---------- Not required to be performed for the turbine driven AFW pump until 24 hours after THERMAL POWER reaches > 2% RTP. ---------- A.7	
	M.6 Verify the developed head of each required AFW pump at the flow test point is greater than or equal to the required developed head.	In accordance with the Inservice Testing Program

C RAI 3.7.5-1

Insert 3.7.5-2: M.4

SURVEILLANCE		FREQUENCY
SR 3.7.5.1	Verify each manual, power operated, and automatic valve in each required water and steam flowpath, that is not locked, sealed, or otherwise secured in position, is in the correct position.	31 days
SR 3.7.5.3	----------NOTE---------- Not applicable in MODE 4 when steam generator is relied upon for heat removal. ----------	
	Verify each AFW automatic valve that is not locked, sealed, or otherwise secured in position, actuates to the correct position on an actual or simulated actuation signal.	18 months
SR 3.7.5.4	----------NOTES---------- 1. Not required to be performed for the turbine driven AFW pump until 24 hours after ≥ 1000 psig in the steam generator. 2. Not applicable in MODE 4 when steam generator is relied upon for heat removal. ----------	
	Verify each AFW pump starts automatically on an actual or simulated actuation signal.	18 months

C RAI 3.7.5-1

CTS INSERTS

Insert 3.7.5-7:

A.1

SURVEILLANCE		FREQUENCY
SR 3.7.5.5	Verify proper alignment of the required AFW flow paths by verifying flow from the condensate storage tank to each steam generator supplied by the respective AFW pump system.	Prior to exceeding 2% RTP whenever unit has been in MODE 5, MODE 6, or defueled for a cummulative period of > 30 days

A.7

C RAI 3.7.5-2

C RAI 3.7.5-2

Insert 3.7.5-8:

L.4

E. Three AFW pump systems inoperable in MODE 1, 2, or 3.	E.1 ---------NOTE-------- LCO 3.0.3 and all other LCO Required Actions requiring MODE changes are suspended until one AFW pump system is restored to OPERABLE status. ---------- Initiate action to restore one AFW pump system to OPERABLE status.	Immediately

< See LCO 3.4.12 >

NOTES USED IN TABLE 15.4.1-1 (continued)

(10) When used for the Low Temperature Overpressure 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.

< See LCO 3.4.11 >

< See LCO 3.4.12 >

(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.

(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.

(14) Calibration is to be a verification of response to a source.

< See LCO 3.3.3 >

SR 3.7.5.5
See Insert 3.7.5-7

A.7

C
RAI 3.7.5-2

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

< See LCO 3.4.3 >

< See LCOs 3.3.1 and 3.3.2>

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

(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.

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

< See LCO 3.1.5 >

(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 LCO 3.3.1 >

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

< See LCO 3.4.11 >

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

< See LCO 3.1.5, 3.1.6, 3.1.7 and 3.3.1 >

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

< See LCOs 3.3.1 and 3.3.2 >

(24) Neutron detectors excluded from calibration.

< See LCO 3.3.1 >

Justification For Deviations - NUREG-1431 Section 3.07.05

16-Oct-00

JFD Number	JFD Text
01 Rev. A	The LCO, Surveillances, Required Actions, and associated Bases has been modified to reflect the Point Beach Auxiliary Feedwater (AFW) system design. The AFW system is divided into three redundant and diverse pump systems per unit. The AFW system consists of a total of four pumps; two motor driven auxiliary feedwater pump systems which are shared by both units, and one dedicated turbine driven pump per unit. Both motor driven AFW pump systems are required to be operable to support one or two unit operation, while the turbine driven pump systems are only credited in the safety analysis to support operation of their respective unit. Each of the two shared motor driven AFW pumps are capable of supplying 100% of the AFW systems design flow rate. AFW pump "P-38A" supplies the "A" steam generator in both units while AFW pump "P-38B" supplies the "B" steam generators. Each AFW pump discharges through an air operated back-pressure control valve and normally closed automatic discharge isolation valves. The air operated back-pressure control valve functions to prevent the motor driven AFW pump from tripping on high current at low steam generator pressures. The back-pressure control valves are provided with a backup nitrogen supply to provide pneumatic pressure in the event of a loss of instrument air. The normally closed discharge motor operated valves automatically reposition to provide 100% of the respective AFW pumps flow to the affected unit. This is accomplished by providing an open signal to the affected units discharge isolation valves, and a close signal to the unaffected units discharge isolation valves whenever the system receives an automatic start signal. Each turbine driven AFW pump is dedicated to its respective unit and is capable of supplying 200% of the design AFW flow rate. The turbine driven AFW pump system supplies both steam generators of its respective unit. The turbine is started by opening at least one of the two DC motor operated steam supply valves. Steam to the turbine can be supplied from each steam generator, via connections to the main steam lines upstream of the main steam isolation valves. The turbine bearing oil is normally cooled by service water with an alternate source of cooling water from the firewater system. The AFW pumps are fed from a common suction header from the condensate storage tanks. The service water system provides the back up safety related source of water for the AFW system via manually operated motor operated valves to each AFW pump suction. Each pump has a recirculation line back to the condensate storage tanks to ensure minimum flow to dissipate pump heat. Each steam generator has a single AFW supply line which is common to the turbine and respective motor driven AFW pumps which supply the steam generator.

ITS:	NUREG:
B 3.07.05	B 3.07.05
	N/A
	N/A
LCO 3.07.05	LCO 3.07.05
LCO 3.07.05 COND A	LCO 3.07.05 COND A

Justification For Deviations - NUREG-1431 Section 3.07.05

16-Oct-00

JFD Number	JFD Text	
	LCO 3.07.05 COND F	LCO 3.07.05 COND E
	LCO 3.07.05 COND F RA F.1	LCO 3.07.05 COND E RA E.1
	SR 3.07.05.05	SR 3.07.05.05
02 Rev. A	Brackets have been removed and site specific information provided.	
	ITS:	**NUREG:**
	B 3.07.05	B 3.07.05
	LCO 3.07.05 COND A	LCO 3.07.05 COND A
	LCO 3.07.05 COND B	LCO 3.07.05 COND B
	LCO 3.07.05 COND D	LCO 3.07.05 COND C LCO 3.07.05 COND C
	SR 3.07.05.02 NOTE	SR 3.07.05.02 NOTE
	SR 3.07.05.03	SR 3.07.05.03
	SR 3.07.05.04 NOTE 1	SR 3.07.05.04 NOTE 1
	SR 3.07.05.05	SR 3.07.05.05
03 Rev. A	The Bases has been modified to reflect Point Beach's AFW System design. The ITS states that the AFW System is designed to supply water to the steam generator by delivering at least the minimum required flow rate at pressures corresponding to the lowest steam generator safety valve set pressure plus 3%. The Point Beach AFW pumps are sized to provide the design AFW flow rate with Steam Generator pressure at 1192 psig (approximately 7% over the highest Steam Generator Safety Valve setpoint and 9% over the lowest).	
	ITS:	**NUREG:**
	B 3.07.05	B 3.07.05
04 Rev. A	The Bases for NUREG 1431 lists a number of Design Basis Accidents and transient which are generically considered to be the most limiting. This statement has been modified to reflect the most limiting event for Point Beach. Main Feedwater Line Break inside the containment is not within Point Beach's Licensing Basis, while a break outside containment is not a limiting event relative to AFW capacity. The limiting event for Point Beach is a loss of normal feedwater, which has been retained in the Bases. The appropriate FSAR reference for the loss of normal feedwater has been provided and subsequent references have been renumbered as necessary to reflect this change. Reference has been provided to the appropriate FSAR Section which contains the design basis. Subsequent reference number has been changed to reflect the addition of this reference.	
	ITS:	**NUREG:**
	B 3.07.05	B 3.07.05

Justification For Deviations - NUREG-1431 Section 3.07.05

16-Oct-00

JFD Number	JFD Text
05 Rev. A	The Bases have been modified to reflect the accident analysis assumptions for Point Beach. The AFW system is assumed to function in the mitigation of; steam generator tube rupture, main steam line break, loss of normal feedwater, and loss of all AC power to the station auxiliaries. The AFW system must also be capable of isolating AFW to a ruptured steam generator in addition to isolating the steam supply to the turbine driven AFW pump associated with the ruptured steam generator following a SGTR. The Point Beach AFW System will be initiated during a LOCA; however, the AFW system is not assumed in the mitigation of primary side Loss of Coolant Accidents (LOCA). Point Beach has analyzed LOCA events assuming no credit for the AFW system. The large break LOCA analysis does not assume secondary heat removal and the small break LOCA was analyzed without AFW to be conservative and to limit the modeling required to address all possible combinations and time delays for various AFW system configurations. **ITS:** B 3.07.05 — **NUREG:** B 3.07.05
06 Rev. A	The automatic start signals for the turbine and motor driven AFW pump systems are not identical, and have therefore been moved in the Bases to earlier discussions specific to the motor and turbine driven pump systems for clarity. **ITS:** B 3.07.05 — **NUREG:** B 3.07.05
07 Rev. A	The Bases for NUREG 1431 states that sufficient AFW flow must be available to account for losses such as pump recirculation flow and line breaks. There are no calculations which establish a leak limit while demonstrating excess pumping capacity to compensate for system leakage. Additionally, at Point Beach, the pump recirculation line is isolated during the event. The AFW system is designed to account for the ability to withstand a single failure. Sufficiency of AFW flow capacity resulting from leakage is accounted for via single failure which renders an entire pump system unavailable. Point Beach design bases provide for the closure of the pump recirculation line and the current licensing basis for Point Beach does not include feedwater line break scenarios. As such, reference to flow losses due to line breaks and pump recirculation have been deleted from the Bases of the ITS. **ITS:** B 3.07.05 — **NUREG:** B 3.07.05
08 Rev. A	The Bases for Required Action A.1 contains an incomplete sentence. The NUREG Bases states "If one of the two steam supplies to the turbine driven AFW train is inoperable, action must be taken to restore OPERABLE status within 7 days". The proposed ITS has been changed to complete the sentence, stating that the "inoperable steam supply" must be restored to OPERABLE status. **ITS:** B 3.07.05 — **NUREG:** B 3.07.05

JFD Number	JFD Text
09 Rev. A	The LCO Bases implies that the AFW system is only required to mitigate the consequences of events which challenge the RCS pressure boundary, while the AFW system is actually assumed to function for several other events to include Steam Generator Tube Rupture, and Main Steam Line Break which do not directly challenge the RCS pressure boundary. As such, the Bases has been changed to state that the AFW system will perform its design safety function, to mitigate the consequences of design basis accidents and transients.

ITS:	NUREG:
B 3.07.05	B 3.07.05

JFD Number	JFD Text
10 Rev. A	NUREG 1431 LCO 3.7.5 Condition B addresses the inoperability of an AFW train. Condition B has been rewritten to address the inoperability of a turbine driven AFW pump system, with new Condition C added to address the inoperability of a motor driven pump system. These changes are necessary to reflect the Point Beach AFW system design and retain the current licensing basis allowable outage times for the motor driven and turbine driven AFW pumps. As described in Justification for Deviation 1 of this Section, the Point Beach AFW system consists of three pump systems. The CTS allows 72 hours to restore a turbine driven pump to operable status and 7 days to restore a motor driven pump before requiring a unit shutdown. The ITS Completion Time limit of 10 days contained in Condition B has been retained and applied both Conditions to limit LCO non-compliance consistent with NUREG 1431.

ITS:	NUREG:
B 3.07.05	B 3.07.05
LCO 3.07.05 COND B	LCO 3.07.05 COND B
LCO 3.07.05 COND B RA B.1	LCO 3.07.05 COND B RA B.1
LCO 3.07.05 COND C	LCO 3.07.05 COND B
LCO 3.07.05 COND C RA C.1	LCO 3.07.05 COND B RA B.1
LCO 3.07.05 COND D	LCO 3.07.05 COND C
LCO 3.07.05 COND D RA D.1	LCO 3.07.05 COND C RA C.1 LCO 3.07.05 COND C RA C.1
LCO 3.07.05 COND D RA D.2	LCO 3.07.05 COND C RA C.2 LCO 3.07.05 COND C RA C.2
LCO 3.07.05 COND E	LCO 3.07.05 COND D
LCO 3.07.05 COND E RA E.1	LCO 3.07.05 COND D RA D.1
LCO 3.07.05 COND E RA E.1 NOTE	LCO 3.07.05 COND D RA D.1 NOTE
LCO 3.07.05 COND F	LCO 3.07.05 COND E
LCO 3.07.05 COND F RA F.1	LCO 3.07.05 COND E RA E.1

JFD Number	JFD Text
11 Rev. A	The terminology used in NUREG 1431 LCO 3.7.5 Condition C has been changed to reflect the Point Beach design. As discussed in Justification for Deviation 1 of this Section, the Point Beach AFW design consists of three pump systems instead of three trains of AFW as addressed in the NUREG. **ITS:** / **NUREG:** B 3.07.05 / B 3.07.05 LCO 3.07.05 COND D / LCO 3.07.05 COND C
12 Rev. A	Condition C of NUREG 1431 LCO 3.7.5 (proposed Condition D)addresses the inoperability of two AFW trains in Mode 1, 2, and 3. The acceptability of a single motor driven AFW train in Mode 4 has been previously addressed in the LCO Section of the Bases. Therefore, this Bases information is being deleted. **ITS:** / **NUREG:** B 3.07.05 / B 3.07.05 SR 3.07.05.04 / SR 3.07.05.04 SR 3.07.05.05 / SR 3.07.05.05
13 Rev. A	The Default Condition (Condition C) for LCO 3.7.5 has been modified to reflect the addition of new Conditions C. Condition C has been added to address Point Beach specific features and licensing basis as described in Justification for Deviation 10 of this Section. New Condition C is applicable in Modes 1, 2, and 3. As such, if the Required Actions are not completed within their specified Completion Times, the unit must be placed in a MODE in which the LCO does not apply. **ITS:** / **NUREG:** B 3.07.05 / B 3.07.05 LCO 3.07.05 COND D / LCO 3.07.05 COND C
14 Rev. A	The steam supply valves to the turbine driven AFW pump and the AFW pump back up suction supply valves from the service water system are not designated as AFW system valves at Point Beach. NUREG 1431 SR 3.7.5.1 requires verification that all AFW manual, power operated, and automatic valves that are not locked, sealed, or otherwise secured in position are in their required positions. This SR is intended to address all valve within the system flow path, inclusive of the turbine driven steam supplies and service water suction lines. As such, the wording of the surveillance has been altered, removing reference to "AFW" valves, eliminating any potential misapplication of the SR. **ITS:** / **NUREG:** SR 3.07.05.01 / SR 3.07.05.01

Justification For Deviations - NUREG-1431 Section 3.07.05

16-Oct-00

JFD Number	JFD Text	
15 Rev. A	The Bases for NUREG 1431 LCO 3.7.5 Required Action A.1 discusses application of a modified Completion Time ("10 days from discovery of failure to met the LCO") which limits the maximum time allowed for LCO non-compliance. NUREG 1431 contained two conditions which could result in indefinite non-compliance with LCO 3.7.5, which therefore required this modified Completion Time, however, the proposed ITS has added a Condition, resulting in the need to modify the Bases associated with Required Action A.1. The proposed change merely recognizes the existence of multiple conditions that could lead to indefinite non-compliance.	
	ITS:	**NUREG:**
	B 3.07.05	B 3.07.05
16 Rev. A	The proposed changes to the Bases clarify the scope of testing for proposed SR 3.7.5.3. As addressed in Justification for Deviation 14 of this Section, the AFW system interfaces with other systems containing manual and automatic valves (i.e. service water and main steam) which are not designated as AFW system valves. SR 3.7.5.3 requires testing of all automatic AFW valves, which would consist of the motor driven AFW pump discharge motor operated valves (i.e. AF-4020, 4021, 4022, and 4023). Testing of other automatic valves not designated as AFW valves, but required to support the AFW pump systems, are addressed in SR 3.7.5.4. SR 3.7.5.4 verifies that the main steam supply valves to the turbine driven AFW pump will automatically open by testing the pump automatic start capability.	
	ITS:	**NUREG:**
	B 3.07.05	B 3.07.05
17 Rev. A	The AFW system is not required to maintain automatic start capabilities in Mode 4 because sufficient time exists for manual initiation of the system if necessary. The Bases has been modified to reflect manual start capability in addition to operation of the residual heat removal system in Mode 4.	
	ITS:	**NUREG:**
	B 3.07.05	B 3.07.05
18 Rev. C	Not used.	
	ITS:	**NUREG:**
	N/A	N/A

JFD Number	JFD Text
19 Rev. C	NUREG 1431 requires the performance of AFW pump testing in accordance with the frequency specified in the Inservice Testing Program (SR 3.7.5.2), AFW pump automatic start testing (SR 3.7.5.4), and verification of proper AFW valve alignment by verifying flow to each steam generator (SR 3.7.5.5). SR 3.7.5.2 and SR 3.7.5.4 are modified by Notes which allow performance of the SRs to be delayed for the turbine driven AFW pump until suitable test conditions are established, and the frequency associated with SR 3.7.5.5 does not require SR 3.7.5.5 to be completed until conditions are appropriate for performing the test. Similar to NUREG 1431, CTS 15.4.8.1.b establishes a bounding limit for completion of turbine driven AFW pump testing, and Note 13 of Table 15.4.1-1 establishes the bounding limit for completion of AFW flow path verification. CTS 15.4.8.1.b requires completion of turbine driven pump testing within 24 hours of entering power operation, and Note 13 of Table 15.4.1-1 requires completion of flow path verification prior to entering power operation whenever the unit has been in cold shutdown for greater than 30 days. Furthermore, CTS 15.1.h defines "power operation" as a condition when the reactor is critical and the average neutron flux of the power range instrumentation indicates greater than 2 percent of rated power. Proposed ITS SR 3.7.5.2 (AFW pump testing) is modified by a Note which allows performance of the test to be deferred for the turbine driven AFW pump until within 24 hours of after exceeding 2% RTP. This exception is consistent with the current licensing basis and prevents excessive RCS cooldowns as a result of steam drawn from the steam generators during pump testing. This Note allows suitable test conditions to be established while allowing a reasonable time period to complete the SR. Proposed ITS SR 3.7.5.5 (AFW flow path verification) is not required to be completed until prior to exceeding 2% power whenever the unit has been in Mode 5, MODE 6, or defueled for a cumulative period of > 30 days. This exception is consistent with the current licensing basis and prevents excessive RCS cooldowns during testing of the turbine driven AFW pump as a result of steam drawn from the steam generators during pump testing. Testing can be accomplished at lower power levels than proposed in SR 3.7.5.2 as the duration of the test proposed in ITS SR 3.7.5.5 is shorter. This frequency allows suitable test conditions to be established while still specifying an acceptable limit for completion of the SR.

ITS:	NUREG:
B 3.07.05	B 3.07.05
SR 3.07.05.02	SR 3.07.05.02
SR 3.07.05.05	SR 3.07.05.05

20 Rev. A	Reviewer note for AFW flow path testing has been deleted. AFW flow path testing has been retained for all AFW flowpaths. Each flowpath is independent.

ITS:	NUREG:
B 3.07.05	B 3.07.05

Approved TSTF 245

Insert 3.7.5-1.

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.7.5.1	Verify each ~~AFW~~ 14 manual, power operated, and automatic valve in each water flow path, [and in both steam supply flow paths to the steam turbine driven pump,] 2 that is not locked, sealed, or otherwise secured in position, is in the correct position.	31 days Approved TSTF 101
SR 3.7.5.2	----------NOTE---------- 2 Not required to be performed for the turbine driven AFW pump until [24 hours] after ~~≥ [1000] psig in the steam generator.~~ 19 THERMAL POWER exceeds 2% RTP. ---------- Verify the developed head of each AFW pump at the flow test point is greater than or equal to the required developed head.	In accordance with the Inservice Testing Program C RAI 3.7.5-1 ~~[31] days on a STAGGERED TEST BASIS~~
SR 3.7.5.3	----------NOTE---------- Not applicable in MODE 4 when steam generator is relied upon for heat removal. ---------- Verify each AFW automatic valve that is not locked, sealed, or otherwise secured in position, actuates to the correct position on an actual or simulated actuation signal.	2 [18] months

Replace with Insert 3.7.5-1.

Approved TSTF 245

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE	FREQUENCY
SR 3.7.5.4 --------- 2 ----NOTES---------- 1. Not required to be performed for the turbine driven AFW pump until 24 hours after ≥ [1000] psig in the steam generator. (1000 — 2)	
2. Not applicable in MODE 4 when steam generator is relied upon for heat removal. (Replace with Insert 3.7.5-1. — Approved TSTF-245)	
Verify each AFW pump starts automatically on an actual or simulated actuation signal. (2)	2 18 months (2)
SR 3.7.5.5 Verify proper alignment of the requir ed AFW flow paths by verifying flow from the condensate storage tank to each steam generator. (1 — supplied by the respective AFW pump system)	Prior to entering MODE 2, whenever unit has been in MODE 5 or 6 for > 30 days (THERMAL POWER exceeding 2% RTP — 19) (2)

Approved TSTF-268 → . MODE 6, or defueled for a cumulative period of > 30 days

RAI 3.7.5-1

RAI 3.7.5-2

BASES

Approved TSTF 101

SURVEILLANCE REQUIREMENTS (continued)

The [31] day Frequency on a STAGGERED TEST BASIS results in testing each pump once every 3 months, as required by Reference 2.

19

Insert B 3.7.5-7

[This SR is modified by a Note indicating that the SR should be deferred until suitable test conditions are established. This deferral is required because there is insufficient steam pressure to perform the test.]

SR 3.7.5.3

16

motor driven AFW pump discharge motor operated valve (AF-4020, 4021, 4022, and 4023) actuate to their correct positions

This SR verifies that AFW can be delivered to the appropriate steam generator in the event of any accident or transient that generates an ESFAS, by demonstrating that each automatic valve in the flow path actuates to its correct position on an actual or simulated actuation signal. This Surveillance is not required for valves that are locked, sealed, or otherwise secured in the required position under administrative controls. The [18] month Frequency is based on the need to perform this Surveillance under the conditions that apply during a unit outage and the potential for an unplanned transient if the Surveillance were performed with the reactor at power. The [18] month Frequency is acceptable based on operating experience and the design reliability of the equipment.

2

Approved TSTF 245

Insert B 3.7.5-10

Insert B 3.7.5-8

17

This SR is modified by a Note that states the SR is not required in MODE 4. In MODE 4, the required AFW train is already aligned and operating.

SR 3.7.5.4

This SR verifies that the AFW pumps will start in the event of any accident or transient that generates an ESFAS by demonstrating that each AFW pump starts automatically on an actual or simulated actuation signal in MODES 1, 2, and 3. In MODE 4, the required pump is already operating and the autostart function is not required. The [18] month Frequency is based on the need to perform this Surveillance under the conditions that apply during a unit outage and the potential for an unplanned transient if the Surveillance were performed with the reactor at power.

2

This SR is modified by [a] [two] Note[s]. Note 1 indicates

two Notes

2

(continued)

RAI 3.7.5-1

BASES

SURVEILLANCE REQUIREMENTS (continued)

that the SR be deferred until suitable test conditions are established. This deferral is required because there is insufficient steam pressure to perform the test.] [The]

Note [2] states that the SR is not required in MODE 4. [In MODE 4, the required pump is already operating and the autostart function is not required.] [In MODE 4, the heat removal requirements would be less providing more time for operator action to manually start the required AFW pump.]

[Reviewer's Note: Some plants may not routinely use the AFW for heat removal in MODE 4. The second justification is provided for plants that use a startup feedwater pump rather than AFW for startup and shutdown.

[SR 3.7.5.5

This SR verifies that the AFW is properly aligned by verifying the flow paths from the CST to each steam generator prior to entering MODE 2 after more than 30 days in MODE 5 or 6. OPERABILITY of AFW flow paths must be verified before sufficient core heat is generated that would require the operation of the AFW System during a subsequent shutdown. The Frequency is reasonable, based on engineering judgement and other administrative controls that ensure that flow paths remain OPERABLE. To further ensure AFW System alignment, flow path OPERABILITY is verified following extended outages to determine no misalignment of valves has occurred. This SR ensures that the flow path from the CST to the steam generators is properly aligned. (This SR is not required by those units that use AFW for normal startup and shutdown.)]

REFERENCES

1. FSAR, Section [10.4.9]
2. ASME, Boiler and Pressure Vessel Code, Section XI.

2. FSAR, Section 14.1.10.

3. ASME, Boiler and Pressure Vessel Code, Section XI.

LCO 3.7.5 BASES INSERTS

Insert B 3.7.5-7:

This SR is modified by a Note indicating that performance of this SR for the turbine driven AFW pump is required to be completed within 24 hours after the unit exceeds 2% of RTP. This exception is required to prevent excessive RCS cooldowns as a result of steam drawn from the steam generators and the cooling effect of AFW water pumped into the steam generators during pump testing. This Note allows suitable test conditions to be established while allowing a reasonable time period to complete the SR during unit startups and low power operation.

C RAI 3.7.5-1

Insert B 3.7.5-8:

In MODE 4, AFW actuation does not need to be OPERABLE because either AFW or residual heat removal (RHR) will already be in operation to remove decay heat or sufficient time is available to manually place either system in operation.

Insert B 3.7.5-9:

Not used.

C RAI 3.7.5-1

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
F. One or more required AFW pump systems inoperable in MODE 4.	F.1 Initiate action to restore AFW pump system(s) to OPERABLE status.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.7.5.1	----------NOTE---------- AFW pump system(s) may be considered OPERABLE during alignment and operation for steam generator level control, if it is capable of being manually realigned to the AFW mode of operation. ---------- Verify each manual, power operated, and automatic valve in each water flow path, and in both steam supply flow paths to the steam turbine driven pump, that is not locked, sealed, or otherwise secured in position, is in the correct position.	31 days
SR 3.7.5.2	----------NOTE---------- Not required to be performed for the turbine driven AFW pump until 24 hours after THERMAL POWER exceeds 2% RTP. ---------- Verify the developed head of each required AFW pump at the flow test point is greater than or equal to the required developed head.	In accordance with the Inservice Testing Program

C RAI 3.7.5-1

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.7.5.3	----------NOTE---------- AFW pump system(s) may be considered OPERABLE during alignment and operation for steam generator level control, if it is capable of being manually realigned to the AFW mode of operation. ---------- Verify each AFW automatic valve that is not locked, sealed, or otherwise secured in position, actuates to the correct position on an actual or simulated actuation signal.	18 months
SR 3.7.5.4	----------NOTES---------- 1. Not required to be performed for the turbine driven AFW pump until 24 hours after ≥ 1000 psig in the steam generator. 2. AFW pump system(s) may be considered OPERABLE during alignment and operation for steam generator level control, if it is capable of being manually realigned to the AFW mode of operation. ---------- Verify each AFW pump starts automatically on an actual or simulated actuation signal.	18 months
SR 3.7.5.5	Verify proper alignment of the required AFW flow paths by verifying flow from the condensate storage tank to each steam generator supplied by the respective AFW pump system.	Prior to THERMAL POWER exceeding 2% RTP whenever unit has been in MODE 5, MODE 6, or defueled for a cumulative period of > 30 days

C RAI 3.7.5-1

C RAI 3.7.5-2

B 3.7 PLANT SYSTEMS

B 3.7.5 Auxiliary Feedwater (AFW) System

BASES

BACKGROUND

The AFW System automatically supplies feedwater to the steam generators to remove decay heat from the Reactor Coolant System upon the loss of normal feedwater supply. The AFW pumps provide cooling water to the steam generator secondary side via connections to the main feedwater (MFW) piping inside containment. The steam generators function as a heat sink for core decay heat. The heat load is dissipated by releasing steam to the atmosphere from the steam generators via the main steam safety valves (MSSVs) (LCO 3.7.1) or atmospheric dump valves (LCO 3.7.4). If the main condenser is available, steam may be released via the steam bypass valves and recirculated to the CST.

The AFW System consists of three independent pump systems; two motor driven AFW pumps which are shared between the two units, and one dedicated steam turbine driven pump per unit. Each motor driven pump is capable of providing 100% of the design AFW flow rate, while the turbine driven pump is capable of providing 200% of the design flowrate. Each pump is provided with a recirculation line to maintain pump discharge flow above the minimum required flow rate for pump cooling. Each AFW pump system can be manually aligned to take suction from the service water system. The normal source of water for the AFW pumps is the Condensate Storage Tank (CST) and the safety related supply is the Service Water (SW) System. Motor operated valves are provided to allow the suction supply for the AFW pumps to be manually transferred to the SW system. For an AFW pump system to be considered OPERABLE, its associated service water suction supply valve must be operable. CST low level alarms and AFW pump low suction pressure alarms and trips are provided to alert personnel that the AFW pump suction supply must be manually swapped.

Each motor driven AFW pump is powered from an independent safeguards power supply and feeds one steam generator in each unit. AFW pump P-38A supplies AFW flow to the Unit 1 and Unit 2 A steam generators, while AFW pump P-38B supplies the Unit 1 and Unit 2 B steam generators. Each motor driven AFW pump's discharge header contains two normally closed automatic motor operated valves. Upon receipt of an AFW actuation signal, the discharge valve associated with the affected unit receives an automatic open signal and the discharge valve associated with the unaffected unit receives an automatic close signal. This feature will ensure that 100% of the motor driven AFW pump flow will be delivered to the affected unit, thereby, assuring that the accident analysis flowrates are met. Each motor driven AFW pump

BASES

BACKGROUND (continued)

is also equipped with a backpressure control valve, which is designed to preclude the motor driven AFW pump from tripping on an overcurrent condition at low steam generator pressures.

The motor driven AFW pump systems actuate automatically on steam generator water level (low-low) and upon receipt of an safety injection (SI) signal. If offsite power is available, the motor driven AFW pump systems actuate immediately. If offsite power is not available, the safeguards buses shed their normal operating loads and are connected to the emergency diesel generators (EDGs). The motor driven AFW pump systems are then actuated per their programmed time sequence. While not credited in any DBA analysis, the motor driven AFW pump systems also actuate on; a trip of all MFW pumps, and by the Anticipated Transient Without Scram Mitigating System Actuation Circuit.

Each unit's turbine driven AFW pump receives steam from both steam generator main steam lines upstream of the main steam isolation valves. Each of the two steam feed lines can supply 100% of the required steam flow to the turbine driven AFW pump. Both steam supply lines must be OPERABLE to consider the turbine driven AFW pump OPERABLE. All power-operated valves associated with the turbine driven AFW pump system are DC-powered, with the exception of the service water suction supply valve (Unit 1 and Unit 2 AF-4006) which is powered from a 480 Volt AC safeguards bus.

The turbine driven AFW pump system actuates automatically on a steam generator water level - low-low in both steam generators. While not credited in any DBA analysis, the turbine driven AFW pump system also actuates on; a trip of all MFW pumps, undervoltage on both main feedwater pump buses, and by the Anticipated Transient Without Scram Mitigating System Actuation Circuit.

The AFW System is capable of supplying feedwater to the steam generators during normal unit startup, shutdown, and hot standby conditions.

One pump at full flow is sufficient to remove decay heat and cool the unit to residual heat removal (RHR) entry conditions. Thus, the requirement for diversity in motive power sources for the AFW System is met.

The AFW System is designed to supply sufficient water to the steam generator(s) to remove decay heat with steam generator pressure at the setpoint of the MSSVs. Subsequently, the AFW System supplies sufficient water to cool the unit to RHR entry conditions, with steam released through the ADVs.

The AFW System is discussed in the FSAR, Section 10.2 (Ref. 1).

BASES

APPLICABLE SAFETY ANALYSES

The AFW System mitigates the consequences of any event with loss of normal feedwater.

The design basis of the AFW System is to supply water to the steam generator to remove decay heat and other residual heat by delivering at least the minimum required flow rate to the steam generators at pressures in excess of the steam generator safety valve set pressure.

In addition, the AFW System must supply enough makeup water to replace steam generator secondary inventory lost as the unit cools to MODE 4 conditions.

The AFW system is assumed to function in the mitigation of Design Basis Accidents (DBAs) and transients to include; Steam Generator Tube Rupture (SGTR), main steam line break, loss of normal feedwater, and loss of all AC power to the station auxiliaries. The AFW system must be capable of isolating AFW to the ruptured steam generator following a SGTR in addition to isolating the steam supply to turbine driven AFW pump associated with the ruptured steam generator. Although the AFW System will be initiated during the Small Break LOCA, the event has been analyzed with no credit for AFW. The Small Break LOCA was analyzed without AFW to be conservative and to limit the modeling required to address all possible combinations and time delays for various AFW system configurations.

The limiting Design Basis Accident (DBA) for the AFW System is the loss of normal feedwater event (Ref. 2).

The ESFAS automatically actuates the AFW turbine driven pump and associated power operated valves and controls when required to ensure an adequate feedwater supply to the steam generators during loss of power. DC power operated valves are provided for each AFW line to control the AFW flow to each steam generator.

The AFW System satisfies the requirements of Criterion 3 of the NRC Policy Statement.

LCO

This LCO provides assurance that the AFW System will perform its design safety function to mitigate the consequences of Design Basis Accidents and transients. Three AFW pump systems, consisting of two shared motor driven pump systems and one dedicated turbine driven pump system are required to be OPERABLE to ensure the availability of RHR capability for all events accompanied by a loss of offsite power and a single failure. This is accomplished by powering two of the pumps from independent emergency buses. The third AFW pump is powered by a different means, a steam driven turbine supplied with steam from a source that is not isolated by closure of the MSIVs.

BASES

LCO (continued)

The AFW System is configured into three pump systems. The AFW System is considered OPERABLE when the components and flow paths required to provide redundant AFW flow to the steam generators are OPERABLE, and the components required to manually transfer AFW pump suction supply to the service water system are OPERABLE. This requires that the two motor driven AFW pumps be OPERABLE, each capable of supplying AFW to a separate steam generator. The turbine driven AFW pump is required to be OPERABLE with redundant steam supplies from each main steam line upstream of the MSIVs, and shall be capable of supplying AFW to both of the steam generators. The piping, valves, instrumentation, and controls in the required flow paths also are required to be OPERABLE.

The LCO is modified by a Note indicating that only the motor driven AFW pumps which are associated with steam generators required to be operable for heat removal (per LCO 3.4.6) are required to be OPERABLE in MODE 4. This is because of the reduced heat removal requirements and short period of time in MODE 4 during which the AFW is required and the insufficient steam available in MODE 4 to power the turbine driven AFW pump.

APPLICABILITY

In MODES 1, 2, and 3, the AFW System is required to be OPERABLE in the event that it is called upon to function when the MFW is lost. In addition, the AFW System is required to supply enough makeup water to replace the steam generator secondary inventory, lost as the unit cools to MODE 4 conditions.

In MODE 4 the AFW System may be used for heat removal via the steam generators.

In MODE 5 or 6, the steam generators are not normally used for heat removal, and the AFW System is not required.

ACTIONS

A.1

If one of the two steam supplies to the turbine driven AFW pump system is inoperable, action must be taken to restore the inoperable steam supply to OPERABLE status within 7 days. The 7 day Completion Time is reasonable, based on the following reasons:

a. The redundant OPERABLE steam supply to the turbine driven AFW pump;

b. The availability of redundant OPERABLE motor driven AFW pumps; and

BASES

ACTIONS (continued) c. The low probability of an event occurring that requires the inoperable steam supply to the turbine driven AFW pump.

The second Completion Time for Required Action A.1 establishes a limit on the maximum time allowed for any combination of Conditions to be inoperable during any continuous failure to meet this LCO.

The 10 day Completion Time provides a limitation time allowed in this specified Condition after discovery of failure to meet the LCO. This limit is considered reasonable for situations in which multiple Conditions are entered concurrently. The AND connector between 7 days and 10 days dictates that both Completion Times apply simultaneously, and the more restrictive must be met.

B.1

With the turbine driven AFW pump system (e.g., pump, flow path, or turbine) inoperable in MODE 1, 2, or 3, action must be taken to restore the pump system to OPERABLE status within 72 hours. The 72 hour Completion Time is reasonable, based on redundant capabilities afforded by the remaining OPERABLE motor driven AFW pump systems, time needed for repairs, and the low probability of a DBA occurring during this time period.

The second Completion Time for Required Action B.1 establishes a limit on the maximum time allowed for any combination of Conditions to be inoperable during any continuous failure to meet this LCO.

The 10 day Completion Time provides a limitation on the time allowed in this specified Condition after discovery of failure to meet the LCO. This limit is considered reasonable for situations in which multiple Conditions are entered simultaneously. The AND connector between the 72 hour and 10 day Completion Times dictates that both Completion Times apply simultaneously, and the more restrictive must be met.

C.1

With one of the motor driven AFW pump systems (e.g., pump or flow path) inoperable in MODE 1, 2, or 3, action must be taken to restore the pump system to OPERABLE status within 7 day. The 7 day Completion Time is reasonable, based on redundant capabilities afforded by the remaining OPERABLE motor driven and turbine driven AFW pump systems, time needed for repairs, and the low probability of a DBA occurring during this time period.

BASES

ACTIONS (continued) The second Completion Time for Required Action C.1 establishes a limit on the maximum time allowed for any combination of Conditions to be inoperable during any continuous failure to meet this LCO.

The 10 day Completion Time provides a limitation on the time allowed in this specified Condition after discovery of failure to meet the LCO. This limit is considered reasonable for situations in which multiple Conditions are entered simultaneously. The AND connector between the 7 day and 10 day Completion Times dictates that both Completion Times apply simultaneously, and the more restrictive must be met.

D.1 and D.2

When Required Action A.1 , B.1, or C.1 cannot be completed within the required Completion Time, or if two AFW pump systems are inoperable in MODE 1, 2, or 3, the unit must be placed in a MODE in which the LCO does not apply. To achieve this status, the unit must be placed in at least MODE 3 within 6 hours, and in MODE 4 within 18 hours.

The allowed Completion Times are reasonable, based on operating experience, to reach the required unit conditions from full power conditions in an orderly manner and without challenging unit systems.

E.1

If all three AFW pump systems are inoperable in MODE 1, 2, or 3, the unit is in a seriously degraded condition with no safety related means for conducting a cooldown, and only limited means for conducting a cooldown with non-safety related equipment. In such a condition, the unit should not be perturbed by any action, including a power change, that might result in a trip. The seriousness of this condition requires that action be started immediately to restore one AFW train to OPERABLE status.

Required Action E.1 is modified by a Note indicating that all required MODE changes or power reductions are suspended until one AFW pump system is restored to OPERABLE status. In this case, LCO 3.0.3 is not applicable because it could force the unit into a less safe condition.

F.1

In MODE 4, either the reactor coolant pumps or the RHR loops can be used to provide forced circulation. This is addressed in LCO 3.4.6, "RCS Loops-MODE 4." With one or more required motor driven pump systems inoperable, action must be taken to immediately restore the inoperable pump system(s) to OPERABLE status. The immediate Completion Time is consistent with LCO 3.4.6.

BASES

SURVEILLANCE REQUIREMENTS

SR 3.7.5.1

Verifying the correct alignment for manual, power operated, and automatic valves in the AFW System water and steam supply flow paths provides assurance that the proper flow paths will exist for AFW operation. This SR does not apply to valves that are locked, sealed, or otherwise secured in position, since they are verified to be in the correct position prior to locking, sealing, or securing. This SR also does not apply to valves that cannot be inadvertently misaligned, such as check valves. This Surveillance does not require any testing or valve manipulation; rather, it involves verification that those valves capable of being mispositioned are in the correct position.

The SR is modified by a Note that states one or more AFW trains may be considered OPERABLE during alignment and operation for steam generator level control, if it is capable of being manually (i.e., remotely or locally, as appropriate) realigned to the AFW mode of operation, provided it is not otherwise inoperable. This exception allows the system to be out of its normal standby alignment and temporarily incapable of automatic initiation without declaring the train(s) inoperable. Since AFW may be used during startup, shutdown, hot standby operations, and hot shutdown operations for steam generator level control, and these manual operations are an accepted function of the AFW system, OPERABILITY (i.e., the intended safety function) continues to be maintained.

The 31 day Frequency is based on engineering judgment, is consistent with the procedural controls governing valve operation, and ensures correct valve positions.

SR 3.7.5.2

Verifying that each AFW pump's developed head at the flow test point is greater than or equal to the required developed head ensures that AFW pump performance has not degraded during the cycle. Flow and differential head are normal tests of centrifugal pump performance required by Section XI of the ASME Code (Ref 3). This test confirms one point on the pump design curve and is indicative of overall performance. Such inservice tests confirm component OPERABILITY, trend performance, and detect incipient failures by indicating abnormal performance. Performance of inservice testing discussed in the ASME Code, Section XI (Ref. 3) (only required at 3 month intervals) satisfies this requirement.

This SR is modified by a Note indicating that performance of this SR for the turbine driven AFW pump is required to be completed within 24 hours after the unit exceeds 2% of RTP. This exception is required C RAI 3.7.5-1

BASES

SURVEILLANCE REQUIREMENTS (continued)

to prevent excessive RCS cooldowns as a result of steam draw from the steam generators during pump testing. This Note allows suitable test conditions to be established while allowing a reasonable time period to complete the SR during unit startups and low power operation.

SR 3.7.5.3

This SR verifies that AFW can be delivered to the appropriate steam generator in the event of any accident or transient that generates an ESFAS, by demonstrating that each motor driven AFW pump discharge motor operated valve (AF-4020, 4021, 4022, and 4023) actuate to their correct positions on an actual or simulated actuation signal. This Surveillance is not required for valves that are locked, sealed, or otherwise secured in the required position under administrative controls. The 18 month Frequency is based on the need to perform this Surveillance under the conditions that apply during a unit outage and the potential for an unplanned transient if the Surveillance were performed with the reactor at power. The 18 month Frequency is acceptable based on operating experience and the design reliability of the equipment.

The SR is modified by a Note that states one or more AFW trains may be considered OPERABLE during alignment and operation for steam generator level control, if it is capable of being manually (i.e., remotely or locally, as appropriate) realigned to the AFW mode of operation, provided it is not otherwise inoperable. This exception allows the system to be out of its normal standby alignment and temporarily incapable of automatic initiation without declaring the train(s) inoperable. Since AFW may be used during startup, shutdown, hot standby operations, and hot shutdown operations for steam generator level control, and these manual operations are an accepted function of the AFW system, OPERABILITY (i.e., the intended safety function) continues to be maintained.

This SR is modified by a Note that states the SR is not required in MODE 4. In MODE 4, AFW actuation does not need to be OPERABLE because either AFW or residual heat removal (RHR) will already be in operation to remove decay heat or sufficient time is available to manually place either system in operation.

SR 3.7.5.4

This SR verifies that the AFW pumps will start in the event of any accident or transient that generates an ESFAS by demonstrating that each AFW pump starts automatically on an actual or simulated actuation signal in MODES 1, 2, and 3. In MODE 4, AFW actuation does not need to be OPERABLE because either AFW or residual heat removal (RHR) will already be in operation to remove decay heat or

BASES

SURVEILLANCE REQUIREMENTS (continued)

sufficient time is available to manually place either system in operation. The 18 month Frequency is based on the need to perform this Surveillance under the conditions that apply during a unit outage and the potential for an unplanned transient if the Surveillance were performed with the reactor at power.

This SR is modified by two Notes. Note 1 indicates that the SR may be deferred until suitable test conditions are established. This deferral is required because there is insufficient steam pressure to perform the test. Note 2 states one or more AFW trains may be considered OPERABLE during alignment and operation for steam generator level control, if it is capable of being manually (i.e., remotely or locally, as appropriate) realigned to the AFW mode of operation, provided it is not otherwise inoperable. This exception allows the system to be out of its normal standby alignment and temporarily incapable of automatic initiation without declaring the train(s) inoperable. Since AFW may be used during startup, shutdown, hot standby operations, and hot shutdown operations for steam generator level control, and these manual operations are an accepted function of the AFW system, OPERABILITY (i.e., the intended safety function) continues to be maintained. C RAI 3.7.5-1

SR 3.7.5.5

This SR verifies that the AFW is properly aligned by verifying the flow paths from the CST to each steam generator supplied by the respective AFW pump system prior to exceeding 2% of RTP after more than 30 days in any combination of MODE 5 or 6 or defueled. OPERABILITY of AFW flow paths must be verified before sufficient core heat is generated that would require the operation of the AFW System during a subsequent shutdown. The Frequency is reasonable, based on engineering judgement and other administrative controls that ensure that flow paths remain OPERABLE. To further ensure AFW System alignment, flow path OPERABILITY is verified following extended outages to determine no misalignment of valves has occurred. This SR ensures that the flow path from the CST to the steam generators is properly aligned. C RAI 3.7.5-1

REFERENCES

1. FSAR, Section 10.2.
2. FSAR, Section 14.1.10.
3. ASME, Boiler and Pressure Vessel Code, Section XI.

No Significant Hazards Considerations - NUREG-1431 Section 3.07.05

16-Oct-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 3.07.05

16-Oct-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 clarifies application of the Required Actions for an inoperable AFW pump to the entire AFW pump system. This change does not result in the introduction of any new or different equipment. Through not introducing any new failure modes and mechanisms, this change does not result in a significant change in the probability of previously evaluated accidents. The consequences of previously evaluated accidents will remain the same because the loss of any pump system component (e.g. piping, valves, or actuation capability) is bounded and at worst, equivalent to the inoperability of the AFW pump itself. Accordingly, the consequences of previously evaluated accidents remain the same. 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 allow limited operation in a condition which is bounded by the exiting condition for an inoperable pump. 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? Expansion of the scope for which the Required Actions can be applied will continue to be enveloped by the loss of the pump itself. Application of the proposed Required Actions will continue to be limited to a single pump system, therefore the redundant pump systems will continue to be required operable. Based on the availability of redundant pump systems, in combination with the low probability of an event occurring in combination with the failure of a remaining operable pump systems, the margin of safety is not impacted.

NSHC Number	NSHC Text
L.02 Rev. A	The CTS only provides Actions for a single inoperable AFW pump during two unit operation, thereby requiring each unit to be placed into hot shutdown (equivalent to ITS Mode 3) within 7 hours, ultimately requiring at least one unit to be then cooled down to less than 350 before the Actions for a single unit operating can then be applied. The proposed ITS will allow the Actions for an inoperable turbine driven AFW pump to be applied to the affected unit alone, with no interdependence established on opposite unit equipment that cannot be shared. 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 changes do not result in any hardware changes, nor does the change significantly increase the probability of any analyzed events since the function of the equipment has remained unchanged. The turbine driven AFW pump systems are not shared between the two units. These pump systems are dedicated to their respective unit. As such, the availability of the opposite units turbine driven AFW pump system has no affect on the probability or consequences of previously evaluated accident. 2. Does the change create the possibility of a new or different kind of accident from any accident previously evaluated? The proposed change does not 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. 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 allow application of the Technical Specification Required Actions for an inoperable turbine driven AFW pump system to the affected unit only. The turbine driven AFW pump systems are not shared systems, therefore no dependency is established in any accident analysis on the opposite unit's turbine driven AFW pump system. Accordingly, this change do not represent a significant reduction in a margin of safety.

NSHC Number	NSHC Text
L.03 Rev. A	Both turbine driven AFW pump steam supply lines are required to be operable to consider the turbine driven AFW pump system to be operable. Therefore, the inoperability of a steam supply line results in entry into the Actions for an inoperability of a turbine driven AFW pump, which allows up to 72 hours to restore the pump to operable status. The proposed ITS will allow 7 days to restore a single inoperable steam supply line to operable status, thus extending the allowable outage time by 96 hours. In accordance with the criteria set forth in 10 CFR 50.92, PBNP has evaluated this proposed Technical Specifications change and determined it does not represent a significant hazards consideration. The following is provided in support of this conclusion. 1. Does the change involve a significant increase in the probability or consequences of an accident previously evaluated? This change does not result in any hardware changes. The AFW system is assumed to function in the mitigation of various design basis events, but is not assumed to be an initiator of any analyzed event. The change will not allow continuous operation such that a single failure will preclude the turbine driven AFW pump system from fulfilling its safety function. This change allows unit operation for an additional 96 hours with one of the two steam supplies to the turbine driven pump inoperable. The consequences of an event occurring during the additional 96 hours are the same as those currently allowed for 72 hours (inoperable turbine driven pump system). Therefore, the proposed change does not increase the probability or consequences of an accident previously evaluated. 2. Does the change create the possibility of a new or different kind of accident from any accident previously evaluated? The proposed change does not 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 not allow continuous unit operation with a steam supply line to the turbine driven AFW pump inoperable. 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 increased time allowed is acceptable based on the small probability of an event during this time frame which would affect the availability of the remaining steam supply while requiring the turbine driven AFW pump system for mitigation of the event. The requested Completion Time will provide a reasonable time to restore an inoperable steam supply to operable status. The condition of a turbine driven AFW pump system being inoperable due to the unavailability of a steam supply line is bounded by the Point Beach single failure evaluation. As such, this change does not significantly reduce the margin of safety.

NSHC Number	NSHC Text
L.04 Rev. A	The CTS only provides Actions for a single inoperable AFW pump, thereby requiring each unit to be placed into hot shutdown (equivalent to ITS Mode 3) within 7 hours in accordance with CTS 15.3.0.B, if more than one AFW pump system is inoperable. The proposed ITS Action for all three AFW pump systems inoperable suspends the requirements of LCO 3.0.3 and requires immediate initiation of action to restore one AFW pump system to operable status. 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 changes do not result in any hardware changes, nor does the change significantly increase the probability of any analyzed events since the function of the equipment has remained unchanged. The CTS requirement to place the unit(s) in a condition that requires AFW when no AFW is available is not appropriate and is being corrected by the proposed change. As such, the proposed change has no affect on the probability or consequences of previously evaluated accident. 2. Does the change create the possibility of a new or different kind of accident from any accident previously evaluated? The proposed change does not 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. 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 allow application of the Technical Specification Required Actions for the condition of all AFW pumping systems inoperable. This proposed change corrects an inconsistency within the CTS. Accordingly, this change does not represent a significant reduction in a margin of safety.
L.05 Rev. C	Not used.

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16-Oct-00

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

No Significant Hazards Considerations - NUREG-1431 Section 3.07.05

16-Oct-00

NSHC Number	NSHC Text
LB Rev. A	In accordance with the criteria set forth in 10 CFR 50.92, PBNP has evaluated this proposed Technical Specifications change and determined it does not represent a significant hazards consideration. The following is provided in support of this conclusion. 1. Does the change involve a significant increase in the probability or consequences of an accident previously evaluated? This change involves deletion of a Specifications/information which is duplicative of information contained in the Code of Federal Regulations (CFRs). This information is more appropriately addressed by the CFRs and serves no purpose in the Technical Specifications. Deletion of this information will not result in an increase in the probability of an accident. Regulatory requirements do not alter plant design or configuration; therefore, this does not alter any event precursor. Accordingly, there will be no effect on the consequences of any accident. 2. Does the change create the possibility of a new or different kind of accident from any accident previously evaluated? The proposed change does not require a physical alteration of the plant (no new or different type of equipment will be installed) or changes in parameters governing normal plant operation. The proposed change deletes materials from the Technical Specifications which are adequately addressed in the CFRs. Thus, this change does not create the possibility of a new or different kind of accident from any accident previously evaluated. 3. Does this change involve a significant reduction in a margin of safety? The proposed change deletes materials from the Technical Specifications which are duplicative of requirements contained in the CFRs. These items are not an input to any accident analysis and, therefore, have no impact on margin of safety.

No Significant Hazards Considerations - NUREG-1431 Section 3.07.05

16-Oct-00

NSHC Number	NSHC Text
M Rev. A	In accordance with the criteria set forth in 10 CFR 50.92, PBNP has evaluated this proposed Technical Specifications change and determined it does not represent a significant hazards consideration. The following is provided in support of this conclusion. 1. Does the change involve a significant increase in the probability or consequences of an accident previously evaluated? The proposed change provides more restrictive requirements for operation of the facility. These more stringent requirements do not result in operation that will increase the probability of initiating an analyzed event and do not alter the assumptions relative to the mitigation of an accident or transient event. These more restrictive requirements continue to ensure process variables, structures, systems and components are maintained consistent with the safety analyses. Therefore, this change does not increase the probability or consequences of an accident previously evaluated. 2. Does the change create the possibility of a new or different kind of accident from any accident previously evaluated? The proposed change does not require a physical alteration of the plant (no new or different type of equipment will be installed) or changes in parameters governing normal plant operation. The proposed change does impose different requirements. However, these changes are consistent with assumptions made in the safety analysis. Thus, this change does not create the possibility of a new or different kind of accident from any accident previously evaluated. 3. Does this change involve a significant reduction in a margin of safety? The imposition of more restrictive requirements either has no affect on or increases the margin of safety. Each change is providing additional restrictions to enhance plant safety. These changes are consistent with the safety analysis. Therefore, this change does not involve a reduction in a margin of safety.