F. MINIMUM CONDITIONS FOR CRITICALITY

<See LCO 3.1.4 >

Specification :

1. Except during low-power physics tests, the reactor shall not be made critical when the moderator temperature coefficient is more positive than 5 pcm/°F.

2. Reactor power shall not exceed 70 percent of Rated Power if the moderator temperature coefficient is positive.

3. During an approach to criticality, at least one (1) count per second, attributable to neutrons, shall register on a narrow range source range nuclear instrument. <See LCO 3.3.1 >

4. In no case shall the reactor be made critical (other than for the purpose of low level physics tests) to the left of the reactor core criticality curve presented in Figure 15.3.1-1. M.1

5. The reactor shall be maintained subcritical by at least 1% $\frac{\Delta k}{k}$ until normal water level is established in the pressurizer. <See LCO 3.4.9 >

Add Action A and SR 3.4.2.1. See Insert 3.4.2-1. M.2

A.2

Basis:

During the early part of the fuel cycle, the moderator temperature coefficient is calculated to be slightly positive at coolant temperatures below 70 percent of rated thermal power.(1)(2) The moderator coefficient at low temperatures will be most positive at the beginning of life of the fuel cycle, when the boron concentration in the coolant is the greatest. Later in the life of the fuel cycle, the boron concentrations in the coolant will be lower and the moderator coefficients will be either less positive or will be negative. At all times, the moderator coefficient is negative when ≥70 percent of rated thermal power. Suitable physics measurements of moderator coefficient of reactivity will be made as part of the startup program to verify analytic predictions.

LCO 3.4.2 CTS Mark up Insert

Insert 3.4.2-1:

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. T_{avg} in one or more RCS loops not within limit.	A.1 Be in MODE 2 with $k_{eff} < 1.0$.	30 minutes

SURVEILLANCE	FREQUENCY
SR 3.4.2.1 Verify RCS T_{avg} in each loop $\geq 540°F$.	12 hours

Description of Changes - NUREG-1431 Section 3.04.02

13-Nov-99

DOC Number	DOC Text
M.02	ITS LCO 3.4.2, Action A is a proposed addition to CTS 15.3.1.F.4. If Tavg in one or more RCS loops is not within the limits of ITS 3.4.2, Action A is entered and the plant must be brought to MODE 2 with keff < 1.0 within 30 minutes. The allowed time is reasonable, based on operating experience, to reach MODE 2 with keff < 1.0 in an orderly manner and without challenging plant systems. SR 3.4.2.1 is another proposed addition to CTS 15.3.1.F.4. SR 3.4.2.1 requires the RCS loop average temperature to be verified at or above 540°F every 30 minutes when the low low Tavg alarm is not reset and any RCS loop Tavg < 547°F. When any RCS loop average temperature is < 547°F and the low low Tavg alarm is alarming, RCS loop average temperatures could fall below the LCO requirement without additional warning. The SR to verify RCS loop average temperatures every 30 minutes is frequent enough to prevent the inadvertent violation of the LCO. Therefore, since these changes place additional requirements on plant operation, they are more restrictive.

CTS:	**ITS:**
NEW	LCO 3.04.02 COND A
	LCO 3.04.02 COND A RA A.1
	SR 3.04.02.01

Justification For Deviations - NUREG-1431 Section 3.04.02

13-Nov-99

JFD Number	JFD Text	
01	The brackets have been removed and the proper plant specific information has been provided. In some instances, even though the information was designated as being site specific information in the LCO (bracketed), the corresponding Bases information was not bracketed. These cases are self evident, corresponding to the bracketed information in the LCO, and have had the appropriate site specific information provided.	
	ITS:	**NUREG:**
	B 3.04.02	B 3.04.02
	LCO 3.04.02	LCO 3.04.02
	SR 3.04.02.01	SR 3.04.02.01
02	The mode of applicability for LCO 3.4.2 is Mode 1 and Mode 2 with Keff >= 1.0. Action A requires the plant to be placed in Mode 3. This is outside the modes of applicability. Therefore it is revised to require that the plant be placed in Mode 2 with Keff < 1.0. This change is consistent with TSTF 26, which has been approved for incorporation into revision two of NUREG 1431.	
	ITS:	**NUREG:**
	B 3.04.02	B 3.04.02
	LCO 3.04.02 COND A RA A.1	LCO 3.04.02 COND A RA A.1
03	With the incorporation of TSTF-9 (relocation of SDM to COLR), the differences between LCO 3.1.1 and LCO 3.1.2 are removed and LCO 3.1.2 is incorporated into LCO 3.1.1, therefore subsequent Section 3.1 LCOs have been renumbered. Accordingly, the reference to LCOs 3.1.10 within the Bases has been revised, to reflect this change. This change is consistent with TSTF 136, which has been approved for incorporation into revision two of NUREG 1431.	
	ITS:	**NUREG:**
	B 3.04.02	B 3.04.02

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.2 RCS Minimum Temperature for Criticality.4.2 RCS Minimum Temperature for Criticality

540 1

LCO 3.4.2 Each RCS loop average temperat ure (T_{avg}) shall be ≥ [541]°F.

APPLICABILITY: MODE 1,
MODE 2 with $k_{eff} \geq 1.0$.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. T_{avg} in one or more RCS loops not within limit.	A.1 Be in MODE 3	30 minutes

2 with $k_{eff} < 1.0$ 2

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.4.2.1	Verify RCS T_{avg} in each loop ≥ [541]°F. 540 1	~~-----NOTE----- Only required if [T_{avg} - T_{ref} deviation, low low T_{avg}] alarm not reset and any RCS loop T_{avg} < [547]°F ---------------- 30 minutes thereafter~~ 12 hours

Approved TSTF 27, Rev. 3

B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.2 RCS Minimum Temperature for Criticality

BASES

BACKGROUND

This LCO is based upon meeting several major considerations before the reactor can be made critical and while the reactor is critical.

3 3

The first consideration is moderator temperature coefficient (MTC), LCO 3.1.4, "Moderator Temperature Coefficient (MTC)." In the transient and accident analyses, the MTC is assumed to be in a range from slightly positive to negative and the operating temperature is assumed to be within the nominal operating envelope while the reactor is critical. The LCO on minimum temperature for criticality helps ensure the plant is operated consistent with these assumptions.

The second consideration is the protective instrumentation. Because certain protective instrumentation (e.g., excore neutron detectors) can be affected by moderator temperature, a temperature value within the nominal operating envelope is chosen to ensure proper indication and response while the reactor is critical.

The third consideration is the pressurizer operating characteristics. The transient and accident analyses assume that the pressurizer is within its normal startup and operating range (i.e., saturated conditions and steam bubble present). It is also assumed that the RCS temperature is within its normal expected range for startup and power operation. Since the density of the water, and hence the response of the pressurizer to transients, depends upon the initial temperature of the moderator, a minimum value for moderator temperature within the nominal operating envelope is chosen.

The fourth consideration is that the reactor vessel is above its minimum nil ductility reference temperature when the reactor is critical.

APPLICABLE SAFETY ANALYSES

Although the RCS minimum temperature for criticality is not itself an initial condition assumed in Design Basis Accidents (DBAs), the closely aligned temperature for hot

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

zero power (HZP) is a process variable that is an initial condition of DBAs, such as the rod cluster control assembly (RCCA) withdrawal, RCCA ejection, and main steam line break accidents performed at zero power that either assumes the failure of, or presents a challenge to, the integrity of a fission product barrier.

All low power safety analyses assume initial RCS loop temperatures ≥ the HZP temperature of 547 °F (Ref. 1). The minimum temperature for criticality limitation provides a small band, 6 °F, for critical operation below HZP. This band allows critical operation below HZP during plant startup and does not adversely affect any safety analyses since the MTC is not significantly affected by the small temperature difference between HZP and the minimum temperature for criticality.

1 7

The RCS minimum temperature for criticality satisfies Criterion 2 of the NRC Policy Statement.

LCO

Compliance with the LCO ensures that the reactor will not be made or maintained critical ($k_{eff} \geq 1.0$) at a temperature less than a small band below the HZP temperature, which is assumed in the safety analysis. Failure to meet the requirements of this LCO may produce initial conditions inconsistent with the initial conditions assumed in the safety analysis.

APPLICABILITY

In MODE 1 and MODE 2 with $k_{eff} \geq 1.0$, LCO 3.4.2 is applicable since the reactor can only be critical ($k_{eff} \geq 1.0$) in these MODES .

9 3

The special test exception of LCO 3.1.10, "MODE 2 PHYSICS TESTS Exceptions," permits PHYSICS TESTS to be performed at ≤ 5% RTP with RCS loop average temperatures slightly lower than normally allowed so that fundamental nuclear characteristics of the core can be verified. In order for nuclear characteristics to be accurately measured, it may be necessary to operate outside the normal restrictions of this LCO. For example, to measure the MTC at beginning of cycle, it is necessary to allow RCS loop average temperatures to fall below $T_{no\ load}$, which may cause RCS loop average

(continued)

BASES

APPLICABILITY (continued)

temperatures to fall below the temperature limit of this LCO.

ACTIONS

A.1

2

2 with k_{eff} < 1.0

If the parameters that are outside the limit cannot be restored, the plant must be brought to a MODE in which the LCO does not apply. To achieve this status, the plant must be brought to MODE 3 within 30 minutes. Rapid reactor shutdown can be readily and practically achieved within a 30 minute period. The allowed time is reasonable, based on operating experience, to reach MODE 3 in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.4.2.1

Approved TSTF 27, Rev. 3

Replace with Insert B3.4.2-1.

RCS loop average temperature is required to be verified at or above [541]°F every 30 minutes when [T_{avg}-T_{ref} deviation, low low T_{avg}] alarm not reset and any RCS loop T_{avg} < [547]°F.

The Note modifies the SR. When any RCS loop average temperature is < [547]°F and the [T_{avg} - T_{ref} deviation, low low T_{avg}] alarm is alarming, RCS loop average temperatures could fall below the LCO requirement without additional warning. The SR to verify RCS loop average temperatures every 30 minutes is frequent enough to prevent the inadvertent violation of the LCO.

REFERENCES

1. FSAR, Section [15.0.3]

14, Table 14.0-1

1

LCO 3.4.2 NUREG Mark up Insert

Insert B3.4.2-1:

$\boxed{540} \leftarrow \boxed{1}$

RCS loop average temperature is required to be verified at or above $\boxed{[541]}$°F every 12 hours. The SR to verify RCS loop average temperatures every 12 hours takes into account indications and alarms that are continuously available to the operator in the control room and is consistent with other routine Surveillances which are typically performed once per shift. In addition, operators are trained to be sensitive to RCS temperature during approach to criticality and will ensure that the minimum temperature for criticality is met as criticality is approached.

13-Nov-99

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

No Significant Hazards Considerations - NUREG-1431 Section 3.04.02

13-Nov-99

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

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.2 RCS Minimum Temperature for Criticality

LCO 3.4.2 Each RCS loop average temperature (T_{avg}) shall be ≥ 540°F.

APPLICABILITY: MODE 1,
MODE 2 with k_{eff} ≥ 1.0.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. T_{avg} in one or more RCS loops not within limit.	A.1 Be in MODE 2 with k_{eff} < 1.0.	30 minutes

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
SR 3.4.2.1 Verify RCS T_{avg} in each loop ≥ 540°F.	12 hours

B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.2 RCS Minimum Temperature for Criticality

BASES

BACKGROUND

This LCO is based upon meeting several major considerations before the reactor can be made critical and while the reactor is critical.

The first consideration is moderator temperature coefficient (MTC), LCO 3.1.3, "Moderator Temperature Coefficient (MTC)." In the transient and accident analyses, the MTC is assumed to be in a range from slightly positive to negative and the operating temperature is assumed to be within the nominal operating envelope while the reactor is critical. The LCO on minimum temperature for criticality helps ensure the plant is operated consistent with these assumptions.

The second consideration is the protective instrumentation. Because certain protective instrumentation (e.g., excore neutron detectors) can be affected by moderator temperature, a temperature value within the nominal operating envelope is chosen to ensure proper indication and response while the reactor is critical.

The third consideration is the pressurizer operating characteristics. The transient and accident analyses assume that the pressurizer is within its normal startup and operating range (i.e., saturated conditions and steam bubble present). It is also assumed that the RCS temperature is within its normal expected range for startup and power operation. Since the density of the water, and hence the response of the pressurizer to transients, depends upon the initial temperature of the moderator, a minimum value for moderator temperature within the nominal operating envelope is chosen.

The fourth consideration is that the reactor vessel is above its minimum nil ductility reference temperature when the reactor is critical.

APPLICABLE SAFETY ANALYSES

Although the RCS minimum temperature for criticality is not itself an initial condition assumed in Design Basis Accidents (DBAs), the closely aligned temperature for hot zero power (HZP) is a process variable that is an initial condition of DBAs, such as the rod cluster control assembly

BASES

APPLICABLE SAFETY ANALYSES (continued)

(RCCA) withdrawal, RCCA ejection, and main steam line break accidents performed at zero power that either assumes the failure of, or presents a challenge to, the integrity of a fission product barrier.

All low power safety analyses assume initial RCS loop temperatures ≥ the HZP temperature of 547 °F (Ref. 1). The minimum temperature for criticality limitation provides a small band, 7 °F, for critical operation below HZP. This band allows critical operation below HZP during plant startup and does not adversely affect any safety analyses since the MTC is not significantly affected by the small temperature difference between HZP and the minimum temperature for criticality.

The RCS minimum temperature for criticality satisfies Criterion 2 of the NRC Policy Statement.

LCO

Compliance with the LCO ensures that the reacto r will not be made or maintained critical ($k_{eff} \geq 1.0$) at a temperature less than a small band below the HZP temperature, which is assumed in the safety analysis. Failure to meet the requirements of this LCO may produce initial conditions inconsistent with the initial conditions assumed in the safety analysis.

APPLICABILITY

In MODE 1 and MODE 2 with $k_{eff} \geq 1.0$, LCO 3.4.2 is applicable since the reactor can only be critical ($k_{eff} \geq 1.0$) in these MODES.

The special test exception of LCO 3.1.9, "MODE 2 PHYSICS TESTS Exceptions," permits PHYSICS TESTS to be performed at ≤ 5% RTP with RCS loop average temperatures slightly lower than normally allowed so that fundamental nuclear characteristics of the core can be verified. In order for nuclear characteristics to be accurately measured, it may be necessary to operate outside the normal restrictions of this LCO. For example, to measure the MTC at beginning of cycle, it is necessary to allow RCS loop average temperatures to fall below $T_{no\ load}$, which may cause RCS loop average temperatures to fall below the temperature limit of this LCO.

BASES

ACTIONS

A.1

If the parameters that are outside the limit cannot be restored, the plant must be brought to a MODE in which the LCO does not apply. To achieve this status, the plant must be brought to MODE 2 with $k_{eff} < 1.0$ within 30 minutes. Rapid reactor shutdown can be readily and practically achieved within a 30 minute period. The allowed time is reasonable, based on operating experience, to reach MODE 2 with $k_{eff} < 1.0$ in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.4.2.1

RCS loop average temperature is required to be verified at or above 540°F every 12 hours. The SR to verify RCS loop average temperatures every 12 hours takes into account indications and alarms that are continuously available to the operator in the control room and is consistent with other routine Surveillances which are typically performed once per shift. In addition, operators are trained to be sensitive to RCS temperature during approach to criticality and will ensure that the minimum temperature for criticality is met as criticality is approached.

REFERENCES

1. FSAR, Section 14, Table 14.0-1.

Cross-Reference Report - NUREG-1431 Section 3.04.03
ITS to CTS

13-Nov-99

ITS	CTS	DOC
B 3.04.03	BASES	A.03
LCO 3.04.03	15.03.01.B.01	A.01
	15.03.01.B.01	A.02
LCO 3.04.03 COND A	NEW	L.01
LCO 3.04.03 COND A RA A.1	NEW	L.01
LCO 3.04.03 COND A RA A.2	NEW	L.01
LCO 3.04.03 COND B	NEW	L.01
LCO 3.04.03 COND B RA B.1	NEW	L.01
LCO 3.04.03 COND B RA B.2	NEW	L.01
LCO 3.04.03 COND C	NEW	M.01
LCO 3.04.03 COND C RA C.1	NEW	M.01
LCO 3.04.03 COND C RA C.2	NEW	M.01
PTLR	15.03.01 F 15.03.01-01	LA.02
	15.03.01 F 15.03.01-02	LA.02
	15.03.01.B.01.A	LA.01
	15.03.01.B.01.B	LA.01
	15.03.01.B.01.C	LA.01
	15.03.01.B.04	LA.01
SR 3.04.03.01	NEW	M.01
SR 3.04.03.01 NOTE	NEW	M.01

Cross-Reference Report - NUREG-1431 Section 3.04.03

CTS to ITS

13-Nov-99

CTS	ITS	DOC
15.03.01 F 15.03.01-01	PTLR	LA.02
15.03.01 F 15.03.01-02	PTLR	LA.02
15.03.01.B.01	LCO 3.04.03	A.02
	LCO 3.04.03	A.01
15.03.01.B.01.A	PTLR	LA.01
15.03.01.B.01.B	PTLR	LA.01
15.03.01.B.01.C	PTLR	LA.01
15.03.01.B.02	FSAR	R.01
15.03.01.B.03	FSAR	R.02
15.03.01.B.04	PTLR	LA.01
15.04.01 T 15.04.01-01 10 (16)	FSAR	R.01
BASES	B 3.04.03	A.03

Description of Changes - NUREG-1431 Section 3.04.03

13-Nov-99

DOC Number	DOC Text
A.01	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:** 15.03.01.B.01 **ITS:** LCO 3.04.03
A.02	CTS 15.3.1.B.1 requires that the RCS temperature and pressure be limited in accordance with the limit lines shown in Figures 15.3.1-1 and 15.3.1-2 during heatup, cooldown, criticality, and inservice leak and hydrostatic testing. Proposed ITS 3.4.3 requires RCS pressure, RCS temperature, and RCS heatup and cooldown rates be maintained within the limits specified in the PTLR. This change is necessary since the pressure/temperature limit curves have been moved to the PTLR. Changes to the PTLR will be controlled by the PTLR process in Section 5 of the proposed ITS. This approach provides an effective level of regulatory control and provides for a more appropriate change control process. The level of safety of facility operation in unaffected by the change, because there is no change in the overall operational requirements. **CTS:** 15.03.01.B.01 **ITS:** LCO 3.04.03
A.03	The Bases of the current Technical Specifications for this section have been completely replaced by revised Bases that reflect the format and applicable content of PBNP ITS Chapter 3.4, consistent with the Standard Technical Specifications for Westinghouse Plants, NUREG-1431. The revised Bases are as shown in the PBNP ITS Bases. **CTS:** BASES **ITS:** B 3.04.03

DOC Number	DOC Text
L.01	CTS 15.3.1.B.1 is revised to adopt ITS LCO 3.4.3, Actions A and B, to provide requirements such that the reactor vessel is not operated outside the bounds of the stress analysis, and that stresses are not increased in other RCPB components. No explicit actions are currently provided for non-compliance with the reactor coolant system pressure and temperature limits of CTS 15.3.1.B.1. As a result, CTS 15.3.0 applies which requires placing the unit in a non-applicable condition. If the requirements of ITS LCO 3.4.3 are not met in MODES 1, 2, 3 or 4, proposed Condition A is entered. Required Action A.1 allows 30 minutes to restore parameter(s) to within limits, so that the RCPB is returned to a condition that has been verified by stress analyses. The 30 minute Completion Time reflects the urgency of restoring the parameters to within the analyzed range. Most violations will not be severe, and the activity can be accomplished in this time in a controlled manner. In addition to restoring operation within limits, Required Action A.2 requires an evaluation be completed within 72 hours to determine if RCS operation can continue. The Completion Time is reasonable to accomplish the evaluation. If the required restoration activity cannot be accomplished within 30 minutes, Required Action B.1 and Required Action B.2 must be implemented to reduce pressure and temperature. If the required evaluation for continued operation cannot be accomplished within 72 hours or the results are indeterminate or unfavorable, action must proceed to reduce pressure and temperature as specified in Required Action B.1 and Required Action B.2. Pressure and temperature are reduced by bringing the plant to MODE 3 within 6 hours and to MODE 5, with RCS pressure < 500 psig, 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. The addition of Required Actions A.1 and A.2 results in less restrictive requirements by allowing continued operation for up to 72 hours while an evaluation of the RCS is performed. Allowing continued operation after exceeding the RCS pressure, RCS temperature, or RCS heatup and cooldown rates is acceptable, because it is only allowed if the parameter(s) of concern can be restored to within limits within 30 minutes. Furthermore, if operation was restored to within the limits of the LCO within 30 minutes, the violation was most likely not severe. Therefore continued operation while an evaluation is performed should not result in any degradation of the RCPB. If the Required Action and associated Completion Time of Condition A are not met, the plant is required to be placed in a lower MODE, because either the RCS remained in an unacceptable P/T region for an extended period of increased stress or a sufficiently severe event caused entry into an unacceptable region. Either possibility indicates a need for more careful examination of the event, best accomplished with the RCS at reduced pressure and temperature. In reduced pressure and temperature conditions, the possibility of propagation with undetected flaws is decreased. **CTS:** NEW **ITS:** LCO 3.04.03 COND A LCO 3.04.03 COND A RA A.1

DOC Number	DOC Text	
	NEW	LCO 3.04.03 COND A RA A.2 LCO 3.04.03 COND B LCO 3.04.03 COND B RA B.1 LCO 3.04.03 COND B RA B.2
LA.01	CTS 15.3.1.B.1.a, b, c and 15.3.1.B.4 provide limitations on the use of, and instructions for updating the pressure/temperature limit curves. These details have been moved to the Pressure/Temperature Limits Report (PTLR). This information provides details or design or process which are not directly pertinent to the actual requirement, i.e., Limiting Condition of Operation or Surveillance Requirement, but rather describe an acceptable method of compliance. Since these details are not necessary to adequately describe the actual regulatory requirement, they can be moved to other documents without impact on safety. Changes to the PTLR will be controlled by the PTLR process in Section 5 of the proposed ITS.	
	CTS:	**ITS:**
	15.03.01.B.01.A	PTLR
	15.03.01.B.01.B	PTLR
	15.03.01.B.01.C	PTLR
	15.03.01.B.04	PTLR
LA.02	CTS Figures 15.3.1-1 and 15.3.1-2, Heatup and Cooldown Limitations Curves, have been moved to the Pressure/Temperature Limits Report (PTLR). Changes to the PTLR will be controlled by the PTLR process in proposed ITS Specification 5.6.6. Proposed ITS Specification 5.6.6 requires: a) RCS pressure and temperature limits for heatup, cooldown, low temperature operation, criticality, and hydrostatic testing as well as heatup and cooldown rates shall be established and documented in the PTLR for LCO 3.4.3, 3.4.10 and 3.4.12. b) the analytical methods used to determine the RCS pressure and temperature limits shall be those previously reviewed and approved by the NRC, specifically those described in Regulatory Guide 1.99, Revision 2, and ASME Code Section III (1974 Edition), Appendix G and WCAP-14040, Rev. 1. c) the PTLR shall be provided to the NRC upon issuance for each reactor vessel fluence period and for any revision or supplement thereto. This change is considered acceptable based on the fact that any changes to any of these operational limits must be calculated in accordance with NRC approved methodologies. This change represents a relaxation of existing requirements, but is consistent with NUREG 1431.	
	CTS:	**ITS:**
	15.03.01 F 15.03.01-01	PTLR
	15.03.01 F 15.03.01-02	PTLR

DOC Number	DOC Text
M.01	CTS 15.3.1.B.1 is revised by adopting ITS LCO 3.4.3, Action C, and SR 3.4.3.1. No explicit actions are currently provided for non-compliance with the reactor coolant system pressure and temperature limits of CTS 15.3.1.B.1 with RCS temperature less than or equal to 200 F. Therefore, Action C and a Surveillance Requirement are provided consistent with NUREG-1431.

If the requirements of ITS LCO 3.4.3 are not met any time other than MODES 1, 2, 3 or 4, proposed Condition C is entered. Required Action C.1 specifies that actions must be initiated immediately to correct operation outside of the P/T limits, so that the RCPB is returned to a condition that has been verified by stress analysis. The immediate Completion Time reflects the urgency of initiating action to restore the parameters to within the analyzed range. Most violations will not be severe, and the activity can be accomplished in this time in a controlled manner. Besides restoring operation within limits, Required Action C.2, requires an evaluation to determine if RCS operation can continue. The evaluation must verify that the RCPB integrity remains acceptable and must be completed prior to entry into MODE 4.

SR 3.4.3.1 requires verification that operation is within the limits of the PTLR every 30 minutes when RCS pressure and temperature conditions are undergoing planned changes. This Frequency is considered reasonable in view of the control room indication available to monitor RCS status. Also, since temperature rate of change limits are specified in hourly increments, 30 minutes permits assessment and correction for minor deviations within a reasonable time. Since this change imposes new requirements, it is more restrictive and has no adverse impact on safety.

CTS:	ITS:
NEW	LCO 3.04.03 COND C
	LCO 3.04.03 COND C RA C.1
	LCO 3.04.03 COND C RA C.2
	SR 3.04.03.01
	SR 3.04.03.01 NOTE

13-Nov-99

DOC Number	DOC Text
R.01	Wisconsin Electric Power Company has utilized the selection criteria provided in the 10 CFR 50.36.ii, and has concluded that the Steam Generator Pressure/Temperature Limits can be relocated to licensee control. The basis for this conclusion is as follows:

The limitation on steam generator pressures and temperatures ensures that pressure-induced stresses on the steam generators do not exceed the maximum allowable fracture toughness limits. These pressure and temperature limits are based on maintaining a steam generator reference temperature-nil ductility temperature (RTndt) sufficient to prevent brittle fracture. As such, the Technical Specification places limits on variables consistent with structural analysis results. However, these limits are not initial condition assumptions of a DBA or transient. These limits represent operating restrictions and Criterion 2 includes operating restrictions. However, it should be noted that in the Final Policy Statement the Criterion 2 discussion specified only those operating restrictions required to preclude unanalyzed accidents and transients be included in Technical Specifications.

Comparison to Screening Criteria:

1. The steam generator P/T limits are not used for, nor capable of, detecting a significant abnormal degradation of the reactor coolant pressure boundary prior to a design basis accident (DBA).
2. Steam generator P/T limits are not a process variable, design feature, or operating restriction that is an initial condition of a DBA or transient.
3. Steam generator P/T limits are not part of a primary success path in the mitigation of a DBA or transient.
4. As discussed in Section 4.0 (Appendix A, page A-55) and summarized in Table 1 of WCAP-11618, the steam generator P/T limits were found to be non-significant risk contributors to core damage frequency and offsite releases. This is, in large part, due to SGTR events being negligible contributors in past PWR PRAs. For Point Beach Station, SGTR sequences are important in the Point Beach IPE. However, this Point Beach plant-specific PSA does not evaluate conditions below 70 °F. Thus, this requirement does not meet Criterion 2 for inclusion in Technical Specification since it is not an operating restriction which is assumed in a DBA or transient which is monitored and controlled during power operation. In addition, it is also recognized that the likelihood of pressurizing the SG secondary side when RCS temperature is below 70 °F is small.

Conclusion:

Since the screening criteria have not been satisfied, the Steam Generator P/T Limits LCO and Surveillances may be relocated to other plant controlled documents outside the Technical Specifications.

CTS:	ITS:
15.03.01.B.02	FSAR
15.04.01 T 15.04.01-01 10 (16)	FSAR

DOC Number	DOC Text
R.02	Wisconsin Electric Power Company has utilized the selection criteria provided in the 10 CFR 50.36.ii, and has concluded that the Pressurizer Heatup and Cooldown Limits can be relocated to licensee control. The basis for this conclusion is as follows: The heatup and cooldown rate limits are placed on the pressurizer to prevent non-ductile failure and assure compatibility of operation with the fatigue analysis performed. The limits meet the requirements given in the ASME Boiler and Pressure Vessel Code, Section III, Appendix G. These limitations are consistent with structural analysis results. However, these limits are not initial condition assumptions of a DBA or transient. These limits represent operating restrictions and Criterion 2 includes operating restrictions. However, it should be noted that in the Final Policy Statement the Criterion 2 discussion specified only those operating restrictions required to preclude unanalyzed accidents and transients be included in Technical Specifications. Comparison to Screening Criteria: 1. Pressurizer heat-up and cool-down and spray water differential temperature limits are not used for, nor capable of, detecting a significant abnormal degradation of the reactor coolant pressure boundary prior to a design basis accident (DBA). 2. Pressurizer heat-up and cool-down and spray water differential temperature limits are not a process variable, design feature, or operating restriction that is an initial condition of a DBA or transient. 3. Pressurizer heat-up and cool-down and spray water differential temperature limits are not a part of a primary success path in the mitigation of a DBA or transient. 4. As discussed in Section 4.0 (Appendix A, page A-41) and summarized in Table 1 of WCAP-11618, the pressurizer heat-up and cool-down limits were found to be non-significant risk contributors to core damage frequency and offsite releases. Wisconsin Electric Power Company has reviewed this evaluation and considers it applicable to Point Beach Station. The Point Beach IPE offers no additional information concerning rates of heat-up and cool-down, as such events are not within its scope. Conclusion: Since the screening criteria have not been satisfied, the Pressurizer Heat-up and Cool-down and spray water differential temperature Limits LCO and Surveillances may be relocated to other plant controlled documents outside the Technical Specifications. **CTS:** 15.03.01.B.03 **ITS:** FSAR

B. Pressure/Temperature Limits

Specification:

1. The Reactor Coolant System temperature and pressure shall be limited in accordance with the limit lines shown in Figure 15.3.1-1 and 15.3.1-2 during heatup, cooldown, criticality, and inservice leak and hydrostatic testing with: A.2

   a. A maximum heatup of 100°F in any one hour,
   b. A maximum cooldown of 100°F in any one hour, and
   c. An average temperature change of ≤10°F per hour during inservice leak and hydrostatic testing operations. LA.1

2. The secondary side of the steam generator will not be pressurized above 200 psig if the temperature of the steam generator vessel shell is below 70°F. R.1

3. The pressurizer temperature shall be limited to:
   a. A maximum heatup of 100°F in any one hour and a maximum cooldown of 200°F in any one hour, and
   b. A maximum spray water temperature differential between the pressurizer and spray fluid of not greater than 320°F. R.2

4. The reactor vessel irradiation surveillance specimens are removed and examined, according to NRC approved schedules, to determine changes in material properties. The results of these examinations shall be considered in the evaluation of the prediction method to be used to update Figures 15.3.1-1 and 15.3.1-2. Revised figures shall be provided to the Commission at least sixty (60) days before the calculated exposure of the applicable reactor vessel exceeds the exposure for which the figures apply LA.1

Add COND A & Cond B.
See Insert 3.4.2-1. L.1

Add COND C and SR 3.4.3.1.
See Insert 3.4.2-2. M.1

A.3

Basis

All components in the Reactor Coolant System are designed to withstand the effects of cyclic loads due to reactor system temperature and pressure changes.[1] These cyclic loads are introduced by normal unit load transients, reactor trips, and startup and shutdown operation. The number of thermal and loading cycles used for design purposes are shown in Table 4.1-8 of the FSAR. During unit startup and shutdown, the rates of temperature and pressure changes are limited. The maximum plant heatup and cooldown rate of 100°F per hour is consistent with the design number of cycles and satisfies stress limits for cyclic operation.

The ASME Code, Section III, Non-mandatory Appendix G contains procedures for the development of heatup and cooldown curves for protection against nonductile failure. The ASME Code requires that a 1/4 wall thickness flaw, either on the inside or outside, depending upon the location of concern, be assumed to exist in the structure. As the Code of Federal Regulations, Title 10, Chapter 50, Appendix G, invokes the ASME Code, Appendix G, the ASME Code procedures are utilized in developing the heatup and cooldown limitation curves.

During heatup, the thermal gradients in the reactor vessel wall produce thermal stresses which vary from compressive at the inner wall to tensile at the outer wall. These thermal induced compressive stresses tend to alleviate the tensile stresses induced by the internal pressure. Therefore, a pressure-temperature curve based on steady-state conditions (i.e., no thermal stresses) represents a lower bound of all similar curves for finite heatup rates when the inner wall of the vessel is treated as the governing location.

The heatup analysis also covers the determination of pressure-temperature limitations for the case in which the outer wall of the vessel becomes the controlling location. The thermal gradients established during heatup produce tensile stresses at the outer wall of the vessel. These stresses are additive to the pressure induced tensile stresses which are already present. The thermal induced stresses at the outer wall of the vessel are tensile and are dependent on both the rate of heatup and the time along the heatup ramp; therefore, a lower bound curve similar to that described for the heatup of the inner wall cannot be defined. Subsequently, for the cases in which the outer wall of the vessel becomes the stress controlling location, each heatup rate of interest must be analyzed on an individual basis.

During cooldown the controlling location is always the inside wall where the cooldown thermal gradients tend to produce tensile stresses while producing compressive stress at the outside wall.

The heatup and cooldown curves are composite curves which are prepared by determining the most conservative case with either the inside or outside wall controlling for any heatup or cooldown rate up to 100°F in any one hour.

In developing these curves, an initial unirradiated RT_{NDT} of -6°F was utilized as reported in BAW-1803 dated January 1984. (Reference 5) This value is based upon a statistical evaluation of Linde 80 weld material test data consisting of measured reference temperatures, drop weight data, and related pre-irradiated Charpy data. A standard deviation (σ_I) of 19°F was also calculated for this data set. Both the initial RT_{NDT} and standard deviation values in BAW-1803 may be revised as additional data are obtained.

As a result of fast neutron irradiation, there will be an increase in the RT_{NDT} with nuclear operation. The maximum integrated fast neutron exposure

A.3

of the vessel is computed to be 2.5 x 10^{19} neutrons/cm^2 for 40 years of operation at 1518.5 MWt and 80 percent load factor.[2] This maximum fluence is the exposure expected at the inner reactor vessel wall. However, the neutron fluence used to predict the ΔRT_{NDT} shift is the one-quarter shell thickness neutron exposure. The relationship between fluence at the vessel ID wall and the fluence at the one-quarter and three-quarter shell thickness locations is as presented in Regulatory Guide 1.99 Revision 2, "Radiation Damage to Reactor Vessel Materials." (Reference 6)

Once the fluence is determined, the adjusted reference temperature used in revising the heatup and cooldown curves is obtained by utilizing the method in Section 1.1 of Regulatory Guide 1.99 Revision 2 (Reference 6) for the limiting weld material of both Unit 1 and Unit 2.

The heatup and cooldown curves presented in Figure 15.3.1-1 and 15.3.1-2 were calculated based on the above information and the methods of ASME Code Section III (1974 Edition), Appendix G, "Protection Against Nonductile Failure", and are applicable up to the operational exposure indicated on the figures.

The regulations governing the pressure-temperature limits (10 CFR 50 - Appendix G and ASME Code Section III - Appendix G) do not require additional margins for instrumentation uncertainties be added to the heatup and cooldown curves. This is because the inclusion of instrumentation uncertainties, in addition to other conservatisms in the methods for calculating the pressure temperature limits, is not necessary to protect the vessel from damage.

The actual temperature shift of the vessel material will be established periodically during operation by removing and evaluating reactor vessel material irradiation surveillance specimens installed near the inside wall of the reactor vessel in the core area. Since the neutron spectra at the irradiation samples and vessel inside radius are identified by a specified lead factor, the measured temperature shift for a sample is an excellent indicator of the effects of power operation on the adjacent section of the reactor vessel. If the experimental temperature shift (at the 30 ft-lb level) does not substantiate the predicted shift, new prediction curves and heatup and cooldown curves must be developed.

The pressure-temperature limit lines shown on Figure 15.3.1-1 for reactor criticality and for inservice leak and hydrostatic testing have been provided to assure compliance with the minimum temperature requirements of Appendix G to 10 CFR 50 for reactor criticality and for inservice leak and hydrostatic testing.

The spray should not be used if the temperature difference between the pressurizer and spray fluid is greater than 320°F. This limit is imposed to maintain the thermal stresses at the pressurizer spray line nozzle below the design limit.

The temperature requirements for the steam generator correspond with the measured NDT for the shell.

The reactor vessel materials surveillance capsule removal schedules have been developed based upon the requirements of the Code of Federal Regulations, Title 10, Part 50, Appendix H and with consideration of ASTM Standard E-185-82. When the capsule lead factors are considered, the scheduled removal dates accommodate the weld data needs of all the participants in the Babcock and Wilcox Master Integrated Reactor Vessel Surveillance Program. Additionally, the schedule will provide plate/forging material data as well as fluence data corresponding to the expiration of the current licenses and of any future license extension.

References

(1) FSAR, Section 4.1.5
(2) Westinghouse Electric Corporation, WCAP-12794, Rev. 3/12795, Rev. 3
(3) Westinghouse Electric Corporation, WCAP-8743
(4) Westinghouse Electric Corporation, WCAP-8738
(5) Babcock & Wilcox, BAW 1803
(6) Regulatory Guide 1.99, Revision 2

< See LCO 3.1.4 >

The limitations of the moderator temperature coefficient are provided to ensure that the assumptions used in the accident and transient analyses remain valid through each fuel cycle. This requirement is waived during low power physics tests to permit measurement of reactor moderator coefficient and other physics design parameters of interest. During physics tests, special operating precautions will be taken. In addition, the strong negative Doppler coefficient[3] and the small integrated Dk/k would limit the magnitude of a power excursion resulting from a reduction of moderator density.

Requiring that the source range instrumentation is registering a count rate attributable to neutrons of at least one (1) count per second insures that the source range instrumentation is functioning properly. A functional source range instrument permits the operator to monitor neutron flux levels and to observe the subcritical neutron multiplication during the positive reactivity addition of the reactor startup. < See LCO 3.3.1 >

The requirement that the reactor is not to be made critical below the Reactor Core Criticality Curve provides assurance that a proper relationship between reactor coolant pressure and temperature will be maintained during system heatup and pressurization. Heatup to this temperature will be accomplished by operating the reactor coolant pumps. However, as provided in 10 CFR Part 50, Appendix G, Section IV.A.3, the reactor core may be taken critical below this curve for the purpose of low-level physics tests. A.3

If the specified shutdown margin is maintained (Section 15.3.10), there is no possibility of an accidental criticality as a result of an increase of moderator temperature or a decrease of coolant pressure.[1] < See LCO 3.1.4 >

The requirement for bubble formation in the pressurizer when the reactor has passed the threshold of 1 percent subcriticality will assure that the reactor coolant system will not be solid when criticality is achieved.

A.3

References:

(1) FSAR Table 3.2.1-1

(2) FSAR Table 3.2.1-9

(3) FSAR Figure 3.2.1-10 < See LCO 3.1.4 >

Figure 15.3.1-1/PBNP Units 1 & 2
Heatup Limitations Applicable to
23.6 Effective Full Power Years
(Approximately January 2001)

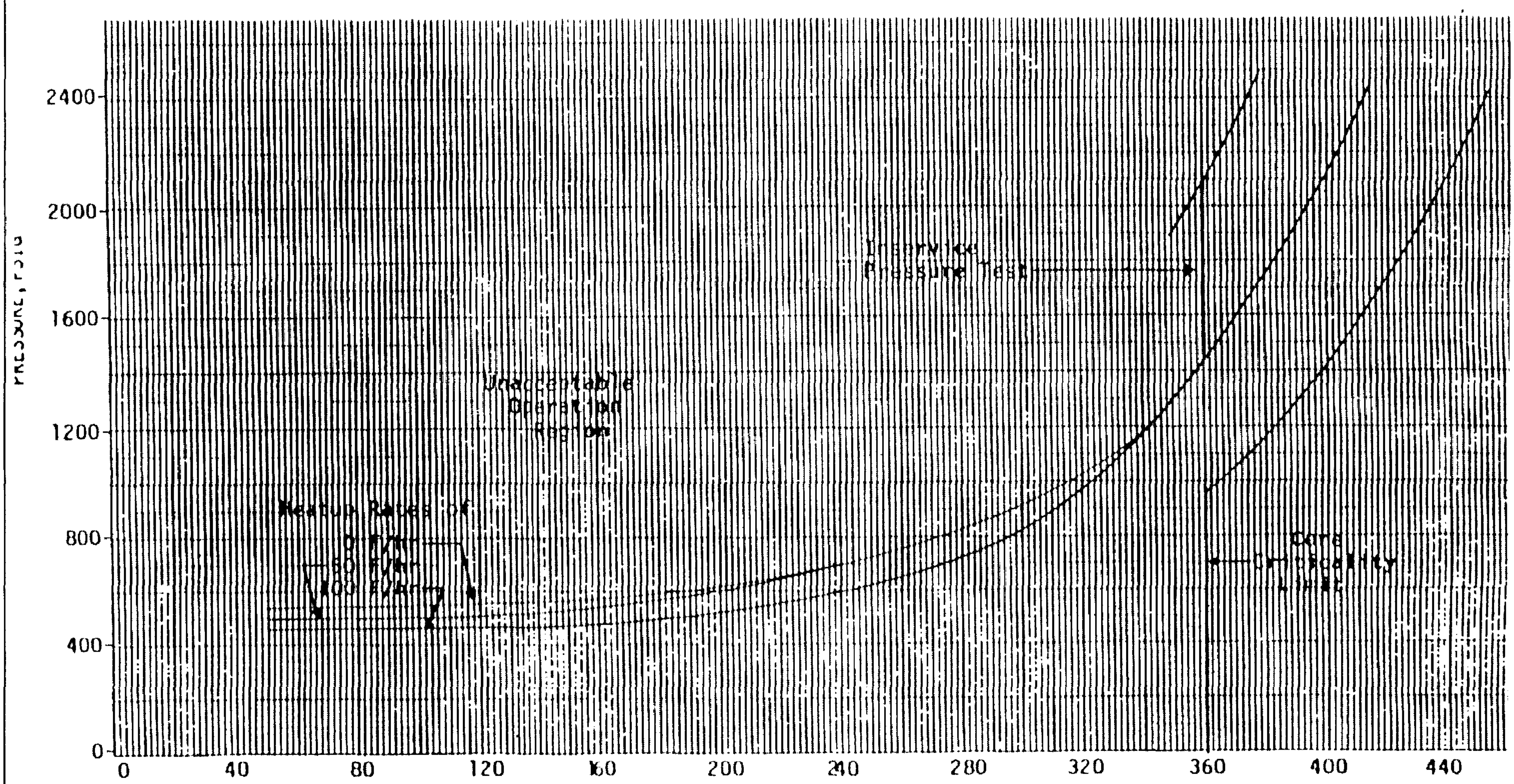

Figure 15.3.1-2/PBNP Units 1 & 2
Cooldown Limitations Applicable to
23.6 Effective Full Power Years
(Approximately January 2001)

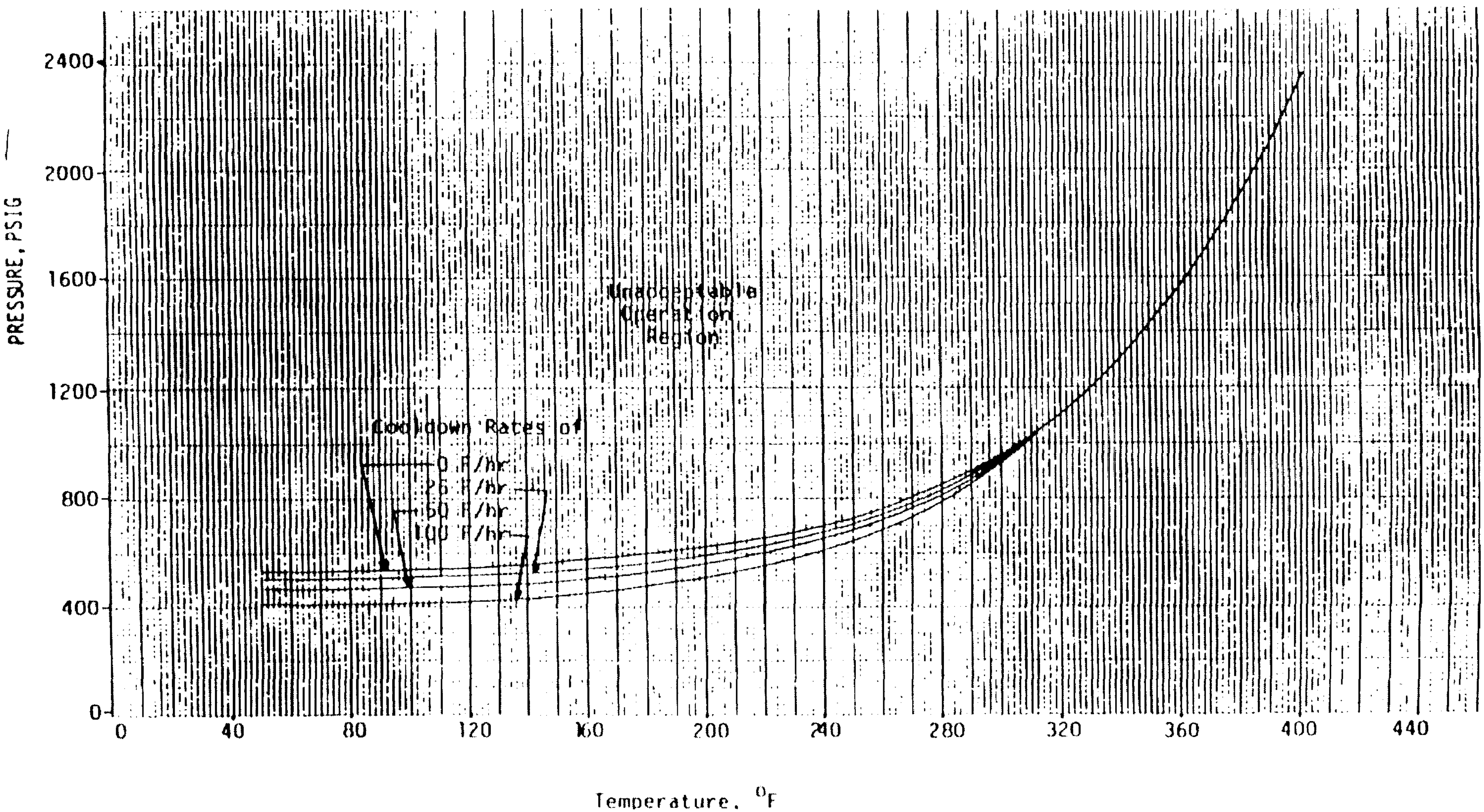

TABLE 15.4.1-1 (continued)

NO.	CHANNEL DESCRIPTION	CHECK	CALIBRATE	TEST	PLANT CONDITIONS WHEN REQUIRED	
9.	Steam Generator Flow Mismatch	S(22)	R	Q(1)	ALL	< See Section 3.3 >
10.	Steam Generator Pressure	S(16)	R	Q(1)	ALL	< See Section 3.3 >
		R.1				
11.	4KV Bus Undervoltage (A01 & A02)					
	-AFW pump actuation	-	R	M(1)	ALL	
	-Reactor Protection actuation	-	R	M(1,2)	ALL	
12.	4KV Bus Underfrequency (A01 & A02)					
	-to Reactor Coolant Pump trip	-	R	-	ALL	< See Section 3.3 >
13.	Safeguards Bus Voltage					
	-Loss of 4KV	S	R	M	ALL	
	-Degraded 4KV	S	R	M	ALL	
	-Loss of 480V	S	R	M	ALL	
14.	120 Vac Instrument Buses	W(6)	-	-	ALL	< See Section 3.8 >
15.	Reactor Trip Signal From Turbine					
	-Turbine Autostop	-	-	M(1)	ALL	
	-Turbine Stop Valve	-	-	M(1)	ALL	
16.	Reactor Trip Signal From SI	-	-	M(1)	ALL	< See Section 3.3 >
17.	Feedwater Isolation on SI					
	-MFP Trip on Safety Injection	-	-	R	ALL	
	-MFRV Shutting on Safety Injection	-	-	R	ALL	
18.	Accumulator Level and Pressure	S	R	-	ALL	< See Section 3.5 >
19.	Analog Rod Position	S(8,22)	R	-	ALL	< See Section 3.1 >
	-with step counters	S(22)	-	-	ALL	
	-Monitoring by On-Line Computer	(18)	-	-	PWR, HOT S/D	

NOTES USED IN TABLE 15.4.1-1 (continued)

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

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

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

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

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

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

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

< See LCO 3.3.3 >

(16) A check of one pressure channel per steam generator is required whenever the steam generator could be pressurized. R.1

(17) Includes test of logic for reactor trip on low-low level, automatic actuation logic for auxiliary feedwater pumps, and test of logic for feedwater isolation on high steam generator level. < See LCO 3.3.1 and 3.3.2 >

< See Section 3.1 >

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

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

(20) To confirm that hot channel factor limits are being satisfied, the requirements of TS 15.3.10.B must be met. < See 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 Section 3.1 and LCO 3.3.1 >

(23) Each train tested at least every 62 days on a staggered basis. < See LCO 3.3.1 and 3.3.2 >

(24) Neutron detectors excluded from calibration. < See LCO 3.3.1 >

Insert 3.4.3-1:

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. ---------NOTE--------- Required Action A.2 shall be completed whenever this Condition is entered. --------------------- Requirements of LCO not met in MODE 1, 2, 3, or 4.	A.1 Restore parameter(s) to within limits. AND A.2 Determine RCS is acceptable for continued operation.	30 minutes 72 hours
B. Required Action and associated Completion Time of Condition A not met.	B.1 Be in MODE 3. AND B.2 Be in MODE 5 with RCS pressure < 500 psig.	6 hours 36 hours

Section 3.4.3 CTS Markup Inserts (continued)

Insert 3.4.3-2:

CONDITION	REQUIRED ACTION	COMPLETION TIME
C. ---------NOTE--------- Required Action C.2 shall be completed whenever this Condition is entered. ----------------------	C.1 Initiate action to restore parameter(s) to within limits.	Immediately
	AND	
Requirements of LCO not met any time in other than MODE 1, 2, 3, or 4.	C.2 Determine RCS is acceptable for continued operation.	Prior to entering MODE 4

SURVEILLANCE		FREQUENCY
SR 3.4.3.1	------------------NOTE------------------- Only required to be performed during RCS heatup and cooldown operations and RCS inservice leak and hydrostatic testing with $k_{eff} < 1.0$. --	
	Verify RCS pressure, RCS temperature, and RCS heatup and cooldown rates are within the limits specified in the PTLR.	30 minutes

Justification For Deviations - NUREG-1431 Section 3.04.03

13-Nov-99

JFD Number	JFD Text	
01	The brackets have been removed and the proper plant specific information has been provided.	
	ITS:	**NUREG:**
	B 3.04.03	B 3.04.03
	LCO 3.04.03 COND B RA B.2	LCO 3.04.03 COND B RA B.2
02	The term "with keff < 1" is being added to the Note for SR 3.4.3.1 clarify that this SR is only required when the reactor has a keff < 1. NUREG 1431 states that the applicability of this LCO is "at all times." However, this SR should not be required when keff >= 1 (i.e when the reactor is critical), because as stated in the bases for SR 3.4.3.1, LCO 3.4.2 establishes a more restrictive requirement for RCS temperature when keff >=1.	
	ITS:	**NUREG:**
	SR 3.04.03.01 NOTE	SR 3.04.03.01 NOTE

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.3 RCS Pressure and Temperature (P/T) Limits

LCO 3.4.3 RCS pressure, RCS temperature, and RCS heatup and cooldown rates shall be maintained within the limits specified in the PTLR.

APPLICABILITY: At all times.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. ---------NOTE--------- Required Action A.2 shall be completed whenever this Condition is entered. --------------------- Requirements of LCO not met in MODE 1, 2, 3, or 4.	A.1 Restore parameter(s) to within limits. <u>AND</u> A.2 Determine RCS is acceptable for continued operation.	30 minutes 72 hours
B. Required Action and associated Completion Time of Condition A not met.	B.1 Be in MODE 3. <u>AND</u> B.2 Be in MODE 5 with RCS pressure < $\boxed{[500]}$ psig.	6 hours 36 hours

(continued)

500

1

ACTIONS (continued)

CONDITION	REQUIRED ACTION		COMPLETION TIME
C. ---------NOTE--------- Required Action C.2 shall be completed whenever this Condition is entered. ----------------------	C.1	Initiate action to restore parameter(s) to within limits.	Immediately
	<u>AND</u>		
Requirements of LCO not met any time in other than MODE 1, 2, 3, or 4.	C.2	Determine RCS is acceptable for continued operation.	Prior to entering MODE 4

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.4.3.1	-------------------NOTE------------------- Only required to be performed during RCS heatup and cooldown operations and RCS inservice leak and hydrostatic testing. --	
	Verify RCS pressure, RCS temperature, and RCS heatup and cooldown rates are within the limits specified in the PTLR.	30 minutes

with $k_{eff} < 1.0$

2

B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.3 RCS Pressure and Temperature (P/T) Limits

BASES

BACKGROUND

All components of the RCS are designed to withstand effects of cyclic loads due to system pressure and temperature changes. These loads are introduced by startup (heatup) and shutdown (cooldown) operations, power transients, and reactor trips. This LCO limits the pressure and temperature changes during RCS heatup and cooldown, within the design assumptions and the stress limits for cyclic operation.

The PTLR contains P/T limit curves for heatup, cooldown, inservice leak and hydrostatic (ISLH) testing, and data for the maximum rate of change of reactor coolant temperature (Ref. 1).

Each P/T limit curve defines an acceptable region for normal operation. The usual use of the curves is operational guidance during heatup or cooldown maneuvering, when pressure and temperature indications are monitored and compared to the applicable curve to determine that operation is within the allowable region.

The LCO establishes operating limits that provide a margin to brittle failure of the reactor vessel and piping of the reactor coolant pressure boundary (RCPB) The vessel is the component most subject to brittle failure, and the LCO limits apply mainly to the vessel. The limits do not apply to the pressurizer, which has different design characteristics and operating functions.

10 CFR 50, Appendix G (Ref. 2), requires the establishment of P/T limits for specific material fracture toughness requirements of the RCPB materials. Reference 2 requires an adequate margin to brittle failure during normal operation, anticipated operational occurrences, and system hydrostatic tests. It mandates the use of the American Society of Mechanical Engineers (ASME) Code, Section III, Appendix G (Ref. 3).

The neutron embrittlement effect on the material toughness is reflected by increasing the nil ductility reference temperature (RT_{NDT}) as exposure to neutron fluence increases.

(continued)

BASES

BACKGROUND (continued)

The actual shift in the RT_{NDT} of the vessel material will be established periodically by removing and evaluating the irradiated reactor vessel material specimens, in accordance with ASTM E 185 (Ref. 4) and Appendix H of 10 CFR 50 (Ref. 5). The operating P/T limit curves will be adjusted, as necessary, based on the evaluation findings and the recommendations of Regulatory Guide 1.99 (Ref. 6).

The P/T limit curves are composite curves established by superimposing limits derived from stress analyses of those portions of the reactor vessel and head that are the most restrictive. At any specific pressure, temperature, and temperature rate of change, one location within the reactor vessel will dictate the most restrictive limit. Across the span of the P/T limit curves, different locations are more restrictive, and, thus, the curves are composites of the most restrictive regions.

The heatup curve represents a different set of restrictions than the cooldown curve because the directions of the thermal gradients through the vessel wall are reversed. The thermal gradient reversal alters the location of the tensile stress between the outer and inner walls.

The criticality limit curve includes the Referenc e 2 requirement that it be ≥ 40°F above the heatup curve or the cooldown curve, and not less than the minimum permissible temperature for ISLH testing. However, the criticality curve is not operationally limiting; a more restrictive limit exists in LCO 3.4.2, "RCS Minimum Temperature for Criticality."

The consequence of violating the LCO limits is that the RCS has been operated under conditions that can result in brittle failure of the RCPB, possibly leading to a nonisolable leak or loss of coolant accident. In the event these limits are exceeded, an evaluation must be performed to determine the effect on the structural integrity of the RCPB components. The ASME Code, Section XI, Appendix E (Ref. 7), provides a recommended methodology for evaluating an operating event that causes an excursion outside the limits.

(continued)

BASES (continued)

APPLICABLE SAFETY ANALYSES

The P/T limits are not derived from Design Basis Accident (DBA) analyses. They are prescribed during normal operation to avoid encountering pressure, temperature, and temperature rate of change conditions that might cause undetected flaws to propagate and cause nonductile failure of the RCPB, an unanalyzed condition. Reference 1 establishes the methodology for determining the P/T limits. Although the P/T limits are not derived from any DBA, the P/T limits are acceptance limits since they preclude operation in an unanalyzed condition.

RCS P/T limits satisfy Criterion 2 of the NRC Policy Statement.

LCO

The two elements of this LCO are:

a. The limit curves for heatup, cooldown, and ISLH testing; and

b. Limits on the rate of change of temperature.

The LCO limits apply to all components of the RCS, except the pressurizer. These limits define allowable operating regions and permit a large number of operating cycles while providing a wide margin to nonductile failure.

The limits for the rate of change of temperature control the thermal gradient through the vessel wall and are used as inputs for calculating the heatup, cooldown, and ISLH testing P/T limit curves. Thus, the LCO for the rate of change of temperature restricts stresses caused by thermal gradients and also ensures the validity of the P/T limit curves.

Violating the LCO limits places the reactor vessel outside of the bounds of the stress analyses and can increase stresses in other RCPB components. The consequences depend on several factors, as follows:

a. The severity of the departure from the allowable operating P/T regime or the severity of the rate of change of temperature;

(continued)

BASES (continued)

LCO (continued)

b. The length of time the limits were violated (longer violations allow the temperature gradient in the thick vessel walls to become more pronounced); and

c. The existences, sizes, and orientations of flaws in the vessel material.

APPLICABILITY

The RCS P/T limits LCO provides a definition of acceptable operation for prevention of nonductile failure in accordance with 10 CFR 50, Appendix G (Ref. 2). Although the P/T limits were developed to provide guidance for operation during heatup or cooldown (MODES 3, 4, and 5) or ISLH testing, their Applicability is at all times in keeping with the concern for nonductile failure. The limits do not apply to the pressurizer.

During MODES 1 and 2, other Technical Specifications provide limits for operation that can be more restrictive than or can supplement these P/T limits. LCO 3.4.1, "RCS Pressure, Temperature, and Flow Departure from Nucleate Boiling (DNB) Limits"; LCO 3.4.2, "RCS Minimum Temperature for Criticality"; and Safety Limit 2.1, "Safety Limits," also provide operational restrictions for pressure and temperature and maximum pressure. Furthermore, MODES 1 and 2 are above the temperature range of concern for nonductile failure, and stress analyses have been performed for normal maneuvering profiles, such as power ascension or descent.

ACTIONS

A.1 and A.2

Operation outside the P/T limits during MODE 1, 2, 3, or 4 must be corrected so that the RCPB is returned to a condition that has been verified by stress analyses.

The 30 minute Completion Time reflects the urgency of restoring the parameters to within the analyzed range. Most violations will not be severe, and the activity can be accomplished in this time in a controlled manner.

Besides restoring operation within limits, an evaluation is required to determine if RCS operation can continue. The

(continued)

BASES

ACTIONS

A.1 and A.2 (continued)

evaluation must verify the RCPB integrity remains acceptable and must be completed before continuing operation. Sever al methods may be used, including comparison with pre-analyzed transients in the stress analyses, new analyses, or inspection of the components.

ASME Code, Section XI, Appendix E (Ref. 7), may be used to support the evaluation. However, its use is restricted to evaluation of the vessel beltline.

The 72 hour Completion Time is reasonable to accomplish the evaluation. The evaluation for a mild violation is possible within this time, but more severe violations may require special, event specific stress analyses or inspections. A favorable evaluation must be completed before continuing to operate.

Condition A is modified by a Note requiring Required Action A.2 to be completed whenever the Condition is entered. The Note emphasizes the need to perform the evaluation of the effects of the excursion outside the allowable limits. Restoration alone per Required Action A.1 is insufficient because higher than analyzed stresses may have occurred and may have affected the RCPB integrity.

B.1 and B.2

If a Required Action and associated Completion Time of Condition A are not met, the plant must be placed in a lower MODE because either the RCS remained in an unacceptable P/T region for an extended period of increased stress or a sufficiently severe event caused entry into an unacceptable region. Either possibility indicates a need for more careful examination of the event, best accomplished with the RCS at reduced pressure and temperature. In reduced pressure and temperature conditions, the possibility of propagation with undetected flaws is decreased.

If the required restoration activity cannot be accomplished within 30 minutes, Required Action B.1 and Required Action B.2 must be implemented to reduce pressure and temperature.

(continued)

BASES

ACTIONS

B.1 and B.2 (continued)

If the required evaluation for continued operation cannot be accomplished within 72 hours or the results are indeterminate or unfavorable, action must proceed to reduce pressure and temperature as specified in Required Action B.1 and Required Action B.2. A favorable evaluation must be completed and documented before returning to operating pressure and temperature conditions.

Pressure and temperature are reduced by bringing the plant to MODE 3 within 6 hours and to MODE 5 with RCS pressure < [500] psig within 36 hours.

1 500

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.

C.1 and C.2

Actions must be initiated immediately to correct operation outside of the P/T limits at times other than when in MODE 1, 2, 3, or 4, so that the RCPB is returned to a condition that has been verified by stress analysis.

The immediate Completion Time reflects the urgency of initiating action to restore the parameters to within the analyzed range. Most violations will not be severe, and the activity can be accomplished in this time in a controlled manner.

Besides restoring operation within limits, an evaluation is required to determine if RCS operation can continue. The evaluation must verify that the RCPB integrity remains acceptable and must be completed prior to entry into MODE 4. Several methods may be used, including comparison with pre-analyzed transient s in the stress analyses, or inspection of the components.

ASME Code, Section XI, Appendix E (Ref. 7), may be used to support the evaluation. However, its use is restricted to evaluation of the vessel beltline.

(continued)

BASES

ACTIONS

C.1 and C.2 (continued)

Condition C is modified by a Note requiring Required Action C.2 to be completed whenever the Condition is entered. The Note emphasizes the need to perform the evaluation of the effects of the excursion outside the allowable limits. Restoration alone per Required Action C.1 is insufficient because higher than analyzed stresses may have occurred and may have affected the RCPB integrity.

SURVEILLANCE REQUIREMENTS

SR 3.4.3.1

Verification that operation is within the PTLR limits is required every 30 minutes when RCS pressure and temperature conditions are undergoing planned changes. This Frequency is considered reasonable in view of the control room indication available to monitor RCS status. Also, since temperature rate of change limits are specified in hourly increments, 30 minutes permits assessment and cor rection for minor deviations within a reasonable time.

Surveillance for heatup, cooldown, or ISLH testing may be discontinued when the definition given in the relevant plant procedure for ending the activity is satisfied.

This SR is modified by a Note that only requires this SR to be performed during system heatup, cooldown, and ISLH testing. No SR is given for criticality operations because LCO 3.4.2 contains a more restrictive requirement.

REFERENCES

1. WCAP-7924-A, April 1975.
2. 10 CFR 50, Appendix G.
3. ASME, Boiler and Pressure Vessel Code, Section III, Appendix G.
4. ASTM E 185-82, July 1982.
5. 10 CFR 50, Appendix H.

(continued)

BASES

REFERENCES (continued)	6.	Regulatory Guide 1.99, Revision 2, May 1988.
	7.	ASME, Boiler and Pressure Vessel Code, Section XI, Appendix E.

No Significant Hazards Considerations - NUREG-1431 Section 3.04.03

13-Nov-99

NSHC Number	NSHC Text
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	In accordance with the criteria set forth in 10 CFR 50.92, PBNP has evaluated this proposed Technical Specifications change and determined it does not represent a significant hazards consideration. The following is provided in support of this conclusion.

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

The proposed change does not involve any physical alteration of plant systems, structures or components, changes in parameters governing normal plant operation, or methods of operation. This change adopts Required Actions for non-compliance with the reactor coolant system pressure and temperature limits, which allows continued operation for up to 72 hours while an evaluation of the RCS is performed. Allowing continued operation after exceeding the RCS pressure, RCS temperature, or RCS heatup and cooldown rates is acceptable, because it is only allowed if the parameter(s) of concern can be restored to within limits within 30 minutes. Furthermore, if operation was restored to within the limits of the LCO within 30 minutes, the violation was most likely not severe. Therefore, continued operation while an evaluation is performed should not result in any degradation of the RCPB. If the required restoration activity cannot be accomplished within 30 minutes, the required evaluation for continued operation cannot be accomplished within 72 hours, or the results are indeterminate or unfavorable, the plant is required to be placed in a lower MODE, because either the RCS remained in an unacceptable P/T region for an extended period of increased stress or a sufficiently severe event caused entry into an unacceptable region. Either possibility indicates a need for more careful examination of the event, best accomplished with the RCS at reduced pressure and temperature. In reduced pressure and temperature conditions, the possibility of propagation with undetected flaws is decreased. Therefore, this change does not involve an increase in the probability or consequences of an accident previously evaluated.

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

The proposed change does not involve any physical alteration of plant systems, structures or components, nor does it alter parameters governing normal plant operation. The proposed change does not introduce a new mode of operation or alter the method of normal plant operation. Therefore, the possibility of a new or different kind of accident from any accident previously evaluated is not created.

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

There are no margins of safety related to safety analyses that are dependent upon the proposed change. This change adopts Required Actions for non-compliance with the reactor coolant system pressure and temperature limits, which allows continued operation for up to 72 hours while an evaluation of the RCS is performed. This is only allowed if the parameter(s) of concern can be restored to within limits within 30 minutes (the violation was

NSHC Number	NSHC Text
	most likely not severe). Furthermore, if the required restoration activity cannot be accomplished within 30 minutes, the required evaluation for continued operation cannot be accomplished within 72 hours, or the results are indeterminate or unfavorable, the plant is required to be placed in a lower MODE. Therefore, this change does not involve a reduction in a margin of safety.
LA	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	In accordance with the criteria set forth in 10 CFR 50.92, PBNP has evaluated this proposed Technical Specifications change and determined it does not represent a significant hazards consideration. The following is provided in support of this conclusion. 1. Does the change involve a significant increase in the probability or consequences of an accident previously evaluated? The proposed change provides more restrictive requirements for operation of the facility. These more stringent requirements do not result in operation that will increase the probability of initiating an analyzed event and do not alter the assumptions relative to the mitigation of an accident or transient event. These more restrictive requirements continue to ensure process variables, structures, systems and components are maintained consistent with the safety analyses. Therefore, this change does not increase the probability or consequences of an accident previously evaluated. 2. Does the change create the possibility of a new or different kind of accident from any accident previously evaluated? The proposed change does not require a physical alteration of the plant (no new or different type of equipment will be installed) or changes in parameters governing normal plant operation. The proposed change does impose different requirements. However, these changes are consistent with assumptions made in the safety analysis. Thus, this change does not create the possibility of a new or different kind of accident from any accident previously evaluated. 3. Does this change involve a significant reduction in a margin of safety? The imposition of more restrictive requirements either has no affect on or increases the margin of safety. Each change is providing additional restrictions to enhance plant safety. These changes are consistent with the safety analysis. Therefore, this change does not involve a reduction in a margin of safety.

NSHC Number	NSHC Text
R	In accordance with the criteria set forth in 10 CFR 50.92, PBNP has evaluated this proposed Technical Specifications change and determined it does not represent a significant hazards consideration. The following is provided in support of this conclusion. 1. Does the change involve a significant increase in the probability or consequences of an accident previously evaluated? The proposed change relocates requirements and surveillances for structures, systems, components or variables which did not meet the criteria for inclusion in Technical Specifications as identified in the 10CFR 50.36 Technical Specification Selection Criteria. The affected structures, systems, components or variables are not assumed to be initiators of analyzed events and are not assumed to mitigate accident or transient events. The requirements and surveillances for these affected structures, systems, components or variables will be relocated from the Technical Specifications to an appropriate administratively controlled document and maintained pursuant to 10CFR 50.59. Therefore, this change does not increase the probability or consequences of an accident previously evaluated. 2. Does the change create the possibility of a new or different kind of accident from any accident previously evaluated? The proposed change does not require a physical alteration of the plant (no new or different type of equipment will be installed) or change in parameters governing normal plant operation. The proposed change will not impose any different requirements and adequate control of information will be maintained. Thus, this change does not create the possibility of a new or different kind of accident from any accident previously evaluated. 3. Does this change involve a significant reduction in a margin of safety? The proposed change will not reduce a margin of safety because it has no impact on any safety analysis assumptions. In addition, the affected requirement will be relocated to an owner controlled document for which future changes will be evaluated pursuant to the requirements of 10CFR 50.59. Therefore, this change does not involve a reduction in a margin of safety.

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.3 RCS Pressure and Temperature (P/T) Limits

LCO 3.4.3 RCS pressure, RCS temperature, and RCS heatup a nd cooldown rates shall be maintained within the limits specified in the PTLR.

APPLICABILITY: At all times.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. ---------NOTE--------- Required Action A.2 shall be completed whenever this Condition is entered. --------------------- Requirements of LCO not met in MODE 1, 2, 3, or 4.	A.1 Restore parameter(s) to within limits. <u>AND</u> A.2 Determine RCS is acceptable for continued operation.	30 minutes 72 hours
B. Required Action and associated Completion Time of Condition A not met.	B.1 Be in MODE 3. <u>AND</u> B.2 Be in MODE 5 with RCS pressure < 500 psig.	6 hours 36 hours

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
C. ---------NOTE--------- Required Action C.2 shall be completed whenever this Condition is entered. --------------------- Requirements of LCO not met any time in other than MODE 1, 2, 3, or 4.	C.1 Initiate action to restore parameter(s) to within limits. AND C.2 Determine RCS is acceptable for continued operation.	Immediately Prior to entering MODE 4

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
SR 3.4.3.1 -------------------NOTE------------------- Only required to be performed during RCS heatup and cooldown operations and RCS inservice leak and hydrostatic testing with $k_{eff} < 1.0$. --- Verify RCS pressure, RCS temperature, and RCS heatup and cooldown rates are within the limits specified in the PTLR.	30 minutes

B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.3 RCS Pressure and Temperature (P/T) Limits

BASES

BACKGROUND

All components of the RCS are designed to withstand effects of cyclic loads due to system pressure and temperature changes. These loads are introduced by startup (heatup) and shutdown (cooldown) operations, power transients, and reactor trips. This LCO limits the pressure and temperature changes during RCS heatup and cooldown, within the design assumptions and the stress limits for cyclic operation.

The PTLR contains P/T limit curves for heatup, cooldown, inservice leak and hydrostatic (ISLH) testing, and data for the maximum rate of change of reactor coolant temperature (Ref. 1).

Each P/T limit curve defines an acceptable region for normal operation. The usual use of the curves is operational guidance during heatup or cooldown maneuvering, when pressure and temperature indications are monitored and compared to the applicable curve to determine that operation is within the allowable region.

The LCO establishes operating limits that provide a margin to brittle failure of the reactor vessel and piping of the reactor coolant pressure boundary (RCPB). The vessel is the component most subject to brittle failure, and the LCO limits apply mainly to the vessel. The limits do not apply to the pressurizer, which has different design characteristics and operating functions.

10 CFR 50, Appendix G (Ref. 2), requires the establishment of P/T limits for specific material fracture toughness requirements of the RCPB materials. Reference 2 requires an adequate margin to brittle failure during normal operation, anticipated operational occurrences, and system hydrostatic tests. It mandates the use of the American Society of Mechanical Engineers (ASME) Code, Section III, Appendix G (Ref. 3).

The neutron embrittlement effect on the material toughness is reflected by increasing the nil ductility reference temperature (RT_{NDT}) as exposure to neutron fluence increases.

BASES

BACKGROUND (continued)

The actual shift in the RT_{NDT} of the vessel material will be established periodically by removing and evaluating the irradiated reactor vessel material specimens, in accordance with ASTM E 185 (Ref. 4) and Appendix H of 10 CFR 50 (Ref. 5). The operating P/T limit curves will be adjusted, as necessary, based on the evaluation findings and the recommendations of Regulatory Guide 1.99 (Ref. 6).

The P/T limit curves are composite curves established by superimposing limits derived from stress analyses of those portions of the reactor vessel and head that are the most restrictive. At any specific pressure, temperature, and temperature rate of change, one location within the reactor vessel will dictate the most restrictive limit. Across the span of the P/T limit curves, different locations are more restrictive, and, thus, the curves are composites of the most restrictive regions.

The heatup curve represents a different set of restrictions than the cooldown curve because the directions of the thermal gradients through the vessel wall are reversed. The thermal gradient reversal alters the location of the tensile stress between the outer and inner walls.

The criticality limit curve includes the Reference 2 requirement that it be ≥ 40°F above the heatup curve or the cooldown curve, and not less than the minimum permissible temperature for ISLH testing. However, the criticality curve is not operationally limiting; a more restrictive limit exists in LCO 3.4.2, "RCS Minimum Temperature for Criticality."

The consequence of violating the LCO limits is that the RCS has been operated under conditions that can result in brittle failure of the RCPB, possibly leading to a nonisolable leak or loss of coolant accident. In the event these limits are exceeded, an evaluation must be performed to determine the effect on the structural integrity of the RCPB components. The ASME Code, Section XI, Appendix E (Ref. 7), provides a recommended methodology for eval uating an operating event that causes an excursion outside the limits.

BASES

APPLICABLE SAFETY ANALYSES

The P/T limits are not derived from Design Basis Accident (DBA) analyses. They are prescribed during normal operation to avoid encountering pressure, temperature, and temperature rate of change conditions that might cause undetected flaws to propagate and cause nonductile failure of the RCPB, an unanalyzed condition. Reference 1 establishes the methodology for determining the P/T limits. Although the P/T limits are not derived from any DBA, the P/T limits are acceptance limits since they preclude operation in an unanalyzed condition.

RCS P/T limits satisfy Criterion 2 of the NRC Policy Statement.

LCO

The two elements of this LCO are:

a. The limit curves for heatup, cooldown, and ISLH testing; and

b. Limits on the rate of change of temperature.

The LCO limits apply to all components of the RCS, except the pressurizer. These limits define allowable operating regions and permit a large number of operating cycles while providing a wide margin to nonductile failure.

The limits for the rate of change of temperature control the thermal gradient through the vessel wall and are used as inputs for calculating the heatup, cooldown, and ISLH testing P/T limit curves. Thus, the LCO for the rate of change of temperature restricts stresses caused by thermal gradients and also ensures the validity of the P/T limit curves.

Violating the LCO limits places the reactor vessel outside of the bounds of the stress analyses and can increase stresses in other RCPB components. The consequences depend on several factors, as follows:

a. The severity of the departure from the allowable operating P/T regime or the severity of the rate of change of temperature;

LCO (continued)

b. The length of time the limits were violated (longer violations allow the temperature gradient in the thick vessel walls to become more pronounced); and

c. The existences, sizes, and orientations of flaws in the vessel material.

APPLICABILITY

The RCS P/T limits LCO provides a definition of acceptable operation for prevention of nonductile failure in accordance with 10 CFR 50, Appendix G (Ref. 2). Although the P/T limits were developed to provide guidance for operation during heatup or cooldown (MODES 3, 4, and 5) or ISLH testing, their Applicability is at all times in keeping with the concern for nonductile failure. The limits do not apply to the pressurizer.

During MODES 1 and 2, other Technical Specifications provide limits for operation that can be more restrictive than or can supplement these P/T limits. LCO 3.4.1, "RCS Pressure, Temperature, and Flow Departure from Nucleate Boiling (DNB) Limits"; LCO 3.4.2, "RCS Minimum Temperature for Criticality"; and Safety Limit 2.1, "Safety Limits," also provide operational restrictions for pressure and temperature and maximum pressure. Furthermore, MODES 1 and 2 are above the temperature range of concern for nonductile failure, and stress analyses have been performed for normal maneuvering profiles, such as power ascension or descent.

ACTIONS

A.1 and A.2

Operation outside the P/T limits during MODE 1, 2, 3, or 4 must be corrected so that the RCPB is returned to a condition that has been verified by stress analyses.

The 30 minute Completion Time reflects the urgency of restoring the parameters to within the analyzed range. Most violations will not be severe, and the activity can be accomplished in this time in a controlled manner.

Besides restoring operation within limits, an evaluation is required to determine if RCS operation can continue. The

BASES

ACTIONS (continued)

evaluation must verify the RCPB integrity remains acceptable and must be completed before continuing operation. Several methods may be used, including comparison with pre -analyzed transients in the stress analyses, new analyses, or inspection of the components.

ASME Code, Section XI, Appendix E (Ref. 7), may be used to support the evaluation. However, its use is restricted to evaluation of the vessel beltline.

The 72 hour Completion Time is reasonable to accomplish the evaluation. The evaluation for a mild violation is possible within this time, but more severe violations may require special, event specific stress analyses or inspections. A favorable evaluation must be completed before continuing to operate.

Condition A is modified by a Note requiring Required Action A.2 to be completed whenever the Condition is entered. The Note emphasizes the need to perform the evaluation of the effects of the excursion outside the allowable limits. Restoration alone per Required Action A.1 is insufficient because higher than analyzed stresses may have occurred and may have affected the RCPB integrity.

B.1 and B.2

If a Required Action and associated Completion Time of Condition A are not met, the plant must be placed in a lower MODE because either the RCS remained in an unacceptable P/T region for an extended period of increased stress or a sufficiently severe event caused entry into an unacceptable region. Either possibility indicates a need for more careful examination of the event, best accomplished with the RCS at reduced pressure and temperature. In reduced pressure and temperature conditions, the possibility of propagation with undetected flaws is decreased.

If the required restoration activity cannot be accomplished within 30 minutes, Required Action B.1 and Required Action B.2 must be implemented to reduce pressure and temperature.

BASES

ACTIONS (continued)

If the required evaluation for continued operation cannot be accomplished within 72 hours or the results are indeterminate or unfavorable, action must proceed to reduce pressure and temperature as specified in Required Action B.1 and Required Action B.2. A favorable evaluation must be completed and documented before returning to operating pressure and temperature conditions.

Pressure and temperature are reduced by bringing the plant to MODE 3 within 6 hours and to MODE 5 with RCS pressure < 500 psig 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.

C.1 and C.2

Actions must be initiated immediately to correct operation outside of the P/T limits at times other than when in MODE 1, 2, 3, or 4, so that the RCPB is returned to a condition that has been verified by stress analysis.

The immediate Completion Time reflects the urgency of initiating action to restore the parameters to within the analyzed range. Most violations will not be severe, and the activity can be accomplished in this time in a controlled manner.

Besides restoring operation within limits, an evaluation is required to determine if RCS operation can continue. The evaluation must verify that the RCPB integrity remains acceptable and must be completed prior to entry into MODE 4. Several methods may be used, including comparison with pre-analyzed transients in the stress analyses, or inspection of the components.

ASME Code, Section XI, Appendix E (Ref. 7), may be used to support the evaluation. However, its use is restricted to evaluation of the vessel beltline.

BASES

ACTIONS (continued)

Condition C is modified by a Note requiring Required Action C.2 to be completed whenever the C ondition is entered. The Note emphasizes the need to perform the evaluation of the effects of the excursion outside the allowable limits. Restoration alone per Required Action C.1 is insufficient because higher than analyzed stresses may have occurred and may have affected the RCPB integrity.

SURVEILLANCE REQUIREMENTS

SR 3.4.3.1

Verification that operation is within the PTLR limits is required every 30 minutes when RCS pres sure and temperature conditions are undergoing planned changes. This Frequency is considered reasonable in view of the control room indication available to monitor RCS status. Also, since temperature rate of change limits are specified in hourly increments, 30 minutes permits assessment and correction for minor deviations within a reasonable time.

Surveillance for heatup, cooldown, or ISLH testing may be discontinued when the definition given in the relevant plant procedure for ending the activity is satisfied.

This SR is modified by a Note that only requires this SR to be performed during system heatup, cooldown, and ISLH testing. No SR is given for criticality operations because LCO 3.4.2 contains a more restrictive requirement.

REFERENCES

1. WCAP-7924-A, April 1975.
2. 10 CFR 50, Appendix G.
3. ASME, Boiler and Pressure Vessel Code, Section III, Appendix G.
4. ASTM E 185-82, July 1982.
5. 10 CFR 50, Appendix H.
6. Regulatory Guide 1.99, Revision 2, May 1988.
7. ASME, Boiler and Pressure Vessel Code, Section XI, Appendix E.

Cross-Reference Report - NUREG-1431 Section 3.04.04

ITS to CTS

13-Nov-99

ITS	CTS	DOC
B 3.04.04	BASES	A.03
LCO 3.04.04	15.03.01.A.01.A	A.01
LCO 3.04.04 COND A	15.03.01.A.01.A.01	A.02
LCO 3.04.04 COND A RA A.1	15.03.01.A.01.A.01	A.02
SR 3.04.04.01	NEW	M.01

Cross-Reference Report - NUREG-1431 Section 3.04.04
CTS to ITS

13-Nov-99

CTS	ITS	DOC
15.03.01.A.01.A	FSAR	R.01
	LCO 3.04.04	A.01
15.03.01.A.01.A.01	LCO 3.04.04 COND A	A.02
	LCO 3.04.04 COND A RA A.1	A.02
BASES	B 3.04.04	A.03

Description of Changes - NUREG-1431 Section 3.04.04

13-Nov-99

DOC Number	DOC Text	
A.01	In the conversion of Point Beach current Technical Specifications (CTS) to the proposed plant specific Improved Technical Specifications (ITS), certain wording preferences or conventions are adopted which do not result in technical changes (either actual or interpretational). Editorial changes, reformatting, and revised numbering are adopted to make the ITS consistent with the Standard Technical Specifications, Westinghouse Plants, NUREG-1431, Revision 1 (i.e., Improved Standard Technical Specifications (ISTS)).	
	CTS:	**ITS:**
	15.03.01.A.01.A	LCO 3.04.04
A.02	CTS 15.3.1.A.1.a(1) requires the reactor to be placed in hot shutdown within 6 hours, if one or both reactor coolant pump(s) cease to operate. CTS defines hot shutdown as a condition when the reactor is subcritical, by an amount greater than or equal to the shutdown margin requirement of CTS 15.3.10, and Tavg is at or greater than 540 degrees F. Therefore, this action places the reactor in a condition whereby the requirements of CTS 15.3.1.A.1.a(1) are not applicable. Proposed ITS 3.4.4, Action A, requires the reactor to be placed in MODE 3 within 6 hours, if the requirements of LCO 3.4.4 are not met (Two RCS loops OPERABLE and in operation.) ITS defines MODE 3 as a condition where keff is < 0.99 and average reactor coolant temperature is greater than or equal to 350 degrees F. Therefore, this action places the reactor in a condition whereby the requirements of ITS 3.4.4 are not applicable. To ensure that the assumptions of the accident analyses remain valid, CTS 15.3.1.A.1.a(1) and ITS 3.4.4 require both RCS loops to be in operation with the reactor critical. When these requirements are not met, both CTS and the proposed ITS actions require lowering reactor power level to a subcritical condition to reduce the core heat removal needs and minimize the possibility of violating DNB limits. The temperature requirements of CTS "hot shutdown" and ITS "MODE 3", although different, are used to provide a range of plant conditions over which each of these terms applies. Although ITS MODE 3 covers a broader range of plant conditions than CTS hot shutdown, the entry point from plant operation with a critical reactor to each of these defined plant conditions is the same. Therefore this change is administrative.	
	CTS:	**ITS:**
	15.03.01.A.01.A.01	LCO 3.04.04 COND A
		LCO 3.04.04 COND A RA A.1
A.03	The Bases of the current Technical Specifications for this section have been completely replaced by revised Bases that reflect the format and applicable content of PBNP ITS Chapter 3.4, consistent with the Standard Technical Specifications for Westinghouse Plants, NUREG-1431. The revised Bases are as shown in the PBNP ITS Bases.	
	CTS:	**ITS:**
	BASES	B 3.04.04

DOC Number	DOC Text
M.01	CTS 15.3.1.A.1.a is revised to adopt ITS SR 3.4.4.1. This proposed surveillance requires verification that each RCS loop is in operation every 12 hours, providing adequate forced reactor coolant flow for core heat removal. Verification includes flow rate, temperature, or pump status monitoring. The frequency of 12 hours is sufficient considering the indications and alarms available to the operator in the control room to monitor RCS loop performance. Since this change imposes new requirements, it is more restrictive and has no adverse impact on safety.

CTS:	ITS:
NEW	SR 3.04.04.01

DOC Number	DOC Text
R.01	Wisconsin Electric Power Company has utilized the selection criteria provided in the 10 CFR 50.36.ii, and has concluded that the Reactor Vessel Head Vent System LCO and Surveillances can be relocated to licensee control. The basis for this conclusion is as follows:

The reactor vessel head vents are provided to exhaust non-condensable gases and/or steam from the RCS which could inhibit natural circulation core cooling following any event involving a loss of offsite power and requiring long term cooling, such as a loss-of-coolant accident (LOCA). Their function, capabilities, and testing requirements are consistent with the requirements of Item II.B.1 of NUREG-0737, "Clarification of TMI Action Plan Requirements," however, the operation of reactor vessel head vents is not assumed in the safety analysis. This is because the operation of the vents is not part of the primary success path. The operation of these vents is an operator action after the event has occurred, and is only required when there is indication that natural circulation is not occurring.

Comparison to Screening Criteria:

1. Reactor vessel head vent system is not used for, nor capable of, detecting a significant abnormal degradation of the reactor coolant pressure boundary prior to a design basis accident (DBA).

2. Reactor vessel head vent system is not a process variable, design feature, or operating restriction that is an initial condition of a DBA or transient.

3. Reactor vessel head vent system is not part of a primary success path in the mitigation of a DBA or transient.

4. As discussed in Section 4.0 (Appendix A, page A-44) and summarized in Table 1 of WCAP-11618, the reactor vessel head vent system was found to be a non-significant risk contributor to core damage frequency and offsite releases. Wisconsin Electric Power Company has reviewed this evaluation and considers it applicable to Point Beach Station. Reactor head vent valves are not important for any scenarios modeled in the Point Beach IPE.

Conclusion:

Since the screening criteria have not been satisfied, the Reactor Vessel Head Vent System LCO and Surveillances may be relocated to other plant controlled documents outside the Technical Specifications

CTS:	**ITS:**
15.03.01.A.01.A	FSAR

15.3 LIMITING CONDITIONS FOR OPERATION

15.3.1 REACTOR COOLANT SYSTEM

Applicability

Applies to the operating status of the Reactor Coolant System.

Objective

To specify those limiting conditions for operation of the Reactor Coolant System which must be met to ensure safe reactor operation.

Specification

A. OPERATIONAL COMPONENTS

1. Coolant Pumps*
   a. When the reactor is critical, both reactor coolant pumps shall be in operation.
      (1) If one or both reactor coolant pump(s) cease operating, the reactor shall be placed in hot shutdown within 6 hours. A.2
   b. When the reactor is subcritical and the average reactor coolant temperature is greater than 350°F, except for tests, at least one reactor coolant pump shall be in operation.
      (1) Both reactor coolant pumps may be deenergized provided:
         a. No operations are permitted that would cause dilution of the reactor coolant system boron concentration,
         b. Core outlet temperature is maintained at least 10°F below saturation temperature, and
         c. The reactor trip breakers are open. < See LCO 3.4.5 >
   c. At least one reactor coolant pump or residual heat removal system shall be in operation when a reduction is made in the boron concentration of the reactor coolant. < See 3.4.6, 3.4.7 & 3.4.8 >

   Add SR 3.4.4.1. See Insert 3.4.4-1. M.1

2. Steam Generator*
   a. One steam generator shall be operable whenever the average reactor coolant temperature is above 350°F. < See LCO 3.4.5 >

3. Components Required for Redundant Decay Heat Removal Capability*
   a. Reactor coolant temperature less than 350°F and greater than 140°F.
      (1) At least two of the decay heat removal methods listed shall be operable.
         (a) Reactor Coolant Loop A, its associated steam generator and either reactor coolant pump
         (b) Reactor Coolant Loop B, its associated steam generator and either reactor coolant pump < See LCO 3.4.6 >

* Applicable only when one or more fuel assemblies are in the reactor vessel.

(5) If both block valves are inoperable, restore the block valves to OPERABLE status within one hour or place the associated PORVs in manual control. Restore at least one block valve to OPERABLE status within the next hour. If these conditions cannot be met, then place the unit in a HOT SHUTDOWN condition within the next six hours. < See LCO 3.4.11 >

6. The pressurizer shall be operable with at least 100 KW of pressurizer heaters available and a water level greater than 10% and less than 95% during steady-state power operation. At least one bank of pressurizer heaters shall be supplied by an emergency bus power supply. < See LCO 3.4.9 >

7. Reactor Coolant Gas Vent System

R.1

These Specifications are not applicable during cold or refueling shutdown conditions:

a. At least one Reactor Coolant Gas Vent System vent path to the pressurizer relief tank (PRT) or containment atmosphere shall be operable from each of the following locations:

(1) Reactor vessel head

(2) Pressurizer

Each vent path from these locations to the common header includes two closed valves in parallel powered from emergency buses. The common header vents to the PRT and the containment atmosphere each contain a closed valve powered from an emergency bus which provides series isolation.

b. When unable to vent from the common header to the PRT or the containment atmosphere, reactor startup and/or power operations may continue provided that the series isolation valve in the inoperable vent path is maintained closed with power removed from the valve actuator.

c. If a vent path from the reactor vessel head or the pressurizer to the common header becomes inoperable, reactor startup and/or power operations may continue provided that the paralleled isolation valves in the inoperable vent path from that location to the common header are maintained closed with power removed from the valve actuator. This does not necessitate removing power from the PRT or

R.1

containment atmosphere isolation valves. The inoperable vent path shall be restored to operable status within thirty days, or the reactor shall be placed in hot shutdown within six hours and in cold shutdown within the following thirty hours.

d. If the vent paths from both the reactor vessel head and the pressurizer to the common header are inoperable or the vent paths from the common header to both the PRT and the containment atmosphere are inoperable, then maintain all the inoperable vent path valves closed with power removed from the valve actuators of all the valves in the inoperable vent paths. Restore at least one of the vent paths from the reactor vessel head or pressurizer to the containment atmosphere or the PRT to operable status within 72 hours or be in hot shutdown within six hours and in cold shutdown within the following thirty hours.

A.3

Basis

When the boron concentration of the reactor coolant system is to be reduced, the process must be uniform to prevent sudden reactivity changes in the reactor. Mixing of the reactor coolant will be sufficient to maintain a uniform boron concentration if at least one reactor coolant pump or one residual heat removal pump is running while the change is taking place. The residual heat removal pump will circulate the primary system volume in approximately one-half hour. The pressurizer is of little concern because of the lower pressurizer volume and because pressurizer boron concentration normally will be higher than that of the rest of the reactor coolant.

Specification 15.3.1.A.1 requires that at least one reactor coolant pump must be operating whenever the average reactor coolant temperature is above 350°F unless the listed restrictions are established. This is required so that the FSAR zero power transients (rod withdrawal from subcritical and rod ejection) are addressed from conservative conditions. With the reactor subcritical, with required shutdown margin, and with the trip breakers open, a single rod ejection will not result in criticality being reached. With the reactor subcritical and the average reactor coolant temperature above 350°F, a single reactor coolant pump provides sufficient decay heat removal capability. Heat transfer analyses[1] show that reactor heat equivalent to 3.5% of the rated power can be removed with natural circulation only.

Items 15.3.1.A.1.a. permits an orderly reduction in power if a reactor coolant pump is lost during operation at less than or equal to 50% of rated power.

A.3

Above 50% power, an automatic reactor trip will occur if either pump is lost. The power-to-flow ratio will be maintained equal to or less than 1.0, which ensures that the minimum DNB ratio increases at lower flow since the maximum enthalpy rise does not increase above its normal full-flow maximum value.[2]

Specification 15.3.1.A.3 provides limiting conditions for operation to ensure that redundancy in decay heat removal methods is provided. A single reactor coolant loop with its associated steam generator and a reactor coolant pump or a single residual heat removal loop provides sufficient heat removal capacity for removing the reactor core decay heat; however, single failure considerations require that at least two decay heat removal methods be available. Operability of a steam generator for decay heat removal includes two sources of water, water level indication in the steam generator, a vent path to atmosphere, and the Reactor Coolant System filled and vented so thermal convection cooling of the core is possible. If the steam generators are not available for decay heat removal, this Specification requires both residual heat removal loops to be operable unless the reactor system is in the refueling shutdown condition with the refueling cavity flooded and no operations in progress which could cause an increase in reactor decay heat load or a decrease in boron concentration. In this condition, the reactor vessel is essentially a fuel storage pool and removing a RHR loop from service provides conservative conditions should operability problems develop in the other RHR loop. Also, one residual heat removal loop may be temporarily out of service due to surveillance testing, calibration, or inspection requirements. The surveillance procedures follow administrative controls which allow for timely restoration of the residual heat removal loop to service if required.

Additionally, with reactor coolant temperature between 350°F and 140°F, all operating decay heat removal pumps (either reactor coolant pumps or residual heat removal pumps) are allowed to be deenergized for a short time (1 hour) with the stipulation that boron dilution activities are not allowed and that core outlet temperature remain 10°F below saturation.

The operation of one reactor coolant pump or one RHR pump provides adequate flow to ensure mixing, prevent stratification and produce gradual reactivity changes during boron concentration reductions in the reactor coolant system. The reactivity change rate associated with boron reduction will, therefore, be within the capability of operator recognition and control.

Each of the pressurizer safety valves is designed to relieve 288,000 lbs per hour of saturated steam at setpoint. If no residual heat is removed by any of the means available, the amount of steam which could be generated at safety valve relief pressure would be less than half the valves' capacity. One valve, therefore, provide adequate defense against overpressurization. Below 350°F and 400 psig in the Reactor Coolant System, the residual heat removal system can remove decay heat and thereby control system temperature and pressure.

A PORV is defined as OPERABLE if leakage past the valve is less than that allowed in Specification 15.3.1.D and the most recent associated channel test, as specified in Table 15.4.1-1. is acceptable. Additionally, the PORV must have the capability of operating manually to relieve reactor coolant system pressure increases.

A block valve is defined as OPERABLE if the valve can operate manually and if it can control identified PORV leakage.

When a PORV is INOPERABLE due to excessive seat leakage, the block valve is shut with power maintained to the block valve so that the block valve(s) is readily available and may be used to allow the PORV to control reactor pressure. Excessive primary system leakage is defined in specification 15.3.1.D. The block valve may remain shut to isolate the leaking PORV for a limited period of time not to exceed the next refueling shutdown. When a PORV is INOPERABLE for reasons other than excessive seat leakage, the block valve is shut with power removed; this precludes any inadvertent opening of the block valve.

When a block valve is INOPERABLE, the associated PORV is placed in manual control; this precludes the undesired automatic opening of the PORV.

The requirement that 100 KW of pressurizer heaters and their associated controls be capable of being supplied electrical power from an emergency bus provides assurance that these heaters can be energized during a loss of offsite power condition to maintain pressure control and natural circulation at hot shutdown.

The requirement to have a reactor coolant system gas vent operable from the reactor vessel or the pressurizer steam space assures that non-condensible gases can be released from the Reactor Coolant System if necessary. The Reactor Coolant Gas Vent System (RCGVS) provides an orificed vent path from the pressurizer steam space and an orificed vent path from the reactor vessel. Both vent paths include two parallel solenoid-operated isolation valves which are powered from emergency buses and vent to a common header. From the common header, gases may be vented via separate lines, each with a single solenoid operated isolation valve powered from the emergency bus to the pressurizer relief tank or containment atmosphere. The orifice in these vent lines restricts leakage so that, in the event of a pipe break or isolation valve failure, makeup water for the leakage can be provided by a single coolant charging pump. If a RCGVS vent path from either the pressurizer or reactor vessel head is inoperable, Specification 15.3.1.A.7.c requires the remotely operable valves in that inoperable path to be shut with power removed. If a vent path from the common header to the pressurizer relief tank or containment atmosphere is inoperable, the isolation valve in that path must be shut but reactor operations may continue. If both vent paths to or both vent paths from the common header are inoperable, the RCGVS is inoperable and the steps in specification 15.3.1.A.7.d must be taken.

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(1) FSAR Section 14.1.11.

(2) FSAR Section 7.2.3.

Section 3.4.4 CTS Markup Insert

Insert 3.4.4-1:

SURVEILLANCE		FREQUENCY
SR 3.4.4.1	Verify each RCS loop is in operation.	12 hours

Justification For Deviations - NUREG-1431 Section 3.04.04

13-Nov-99

JFD Number	JFD Text	
01	The brackets have been removed and the proper plant specific information has been provided.	
	ITS:	**NUREG:**
	B 3.04.04	B 3.04.04
	LCO 3.04.04	LCO 3.04.04
02	NUREG-1431 LCO 3.4.4 Bases have been modified to reflect the Point Beach current licensing basis values for Power Range Neutron Flux - High trip analysis setpoint (118%), and the maximum assumed power level used to generate the pressure temperature Safety Limit (120%).	
	ITS:	**NUREG:**
	B 3.04.04	B 3.04.04
03	LCO 3.9.2 "Unborated Water Source Isolation Valves" was not adopted, based on the Point Beach design. Accordingly, the references to LCO 3.9.5 and 6 within the Bases for LCO 3.4.4 have been revised to reflect the renumbering that has occurred in Section 3.9 of the ITS.	
	ITS:	**NUREG:**
	B 3.04.04	B 3.04.04
04	NUREG-1431, LCO 3.4.4, Bases discussion of the LCO has been modified such that the "definition" of what constitutes an OPERABLE RCP loop applies only in MODES 1 and 2. Although LCO 3.4.4 is only applicable in MODES 1 and 2, this prevents the misapplication of this information in other LCOs.	
	ITS:	**NUREG:**
	B 3.04.04	B 3.04.04

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.4 RCS Loops – MODES 1 and 2.4.4 RCS Loops – MODES 1 and 2

LCO 3.4.4 [Four] RCS loops shall be OPERABLE and in operation.

Two 1

APPLICABILITY: MODES 1 and 2.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Requirements of LCO not met.	A.1 Be in MODE 3.	6 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
SR 3.4.4.1 Verify each RCS loop is in operation.	12 hours

B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.4 RCS Loops – MODES 1 and 2

BASES

BACKGROUND

The primary function of the RCS is removal of the heat generated in the fuel due to the fission process, and transfer of this heat, via the steam generators (SGs), to the secondary plant.

The secondary functions of the RCS include:

a. Moderating the neutron energy level to the thermal state, to increase the probability of fission;

b. Improving the neutron economy by acting as a reflector;

c. Carrying the soluble neutron po ison, boric acid;

d. Providing a second barrier against fission product release to the environment; and

e. Removing the heat generated in the fuel due to fission product decay following a unit shutdown. two 1

The reactor coolant is circulated through [four] loops connected in parallel to the reactor vessel, each containing an SG, a reactor coolant pump (RCP), and appropriate flow and temperature instrumentation for both control and protection. The reactor vessel contains the clad fuel. The SGs provide the heat sink to the isolated secondary coolant. The RCPs circulate the coolant through the reactor vessel and SGs at a sufficient rate to ensure proper heat transfer and prevent fuel damage. This forced circulation of the reactor coolant ensures mixing of the coolant for proper boration and chemistry control.

APPLICABLE SAFETY ANALYSES

Safety analyses contain various assumptions for the design bases accident initial conditions including RCS pressure, RCS temperature, reactor power level, core parameters, and safety system setpoints. The important aspect for this LCO is the reactor coolant forced flow rate, which is represented by the number of RCS loops in service.

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

1 → two

Both transient and steady state analyses have been performed to establish the effect of flow on the departure from nucleate boiling (DNB). The transient and accident analyses for the plant have been performed assuming [four] RCS loops are in operation. The majority of the plant safety analyses are based on initial conditions at high core power or zero power. The accident analyses that are most important to RCP operation are the [four] pump coastdown, single pump locked rotor, single pump (broken shaft or coastdown), and rod withdrawal events (Ref. 1).

1 → two

Steady state DNB analysis has been performed for the [four] RCS loop operation. For [four] RCS loop operation, the steady state DNB analysis, which generates the pressure and temperature Safety Limit (SL) (i.e., the departure from nucleate boiling ratio (DNBR) limit) assumes a maximum power level of 109% RTP. This is the design overpower condition for [four] RCS loop operation. The value for the accident analysis setpoint of the nuclear overpower (high flux) trip is 107% and is based on an analysis assumption that bounds possible instrumentation errors. The DNBR limit defines a locus of pressure and temperature points that result in a minimum DNBR greater than or equal to the critical heat flux correlation limit.

2 → 120

1 → two

2 → 118

The plant is designed to operate with all RCS loops in operation to maintain DNBR above the SL, during all normal operations and anticipated transients. By ensuring heat transfer in the nucleate boiling region, adequate heat transfer is provided between the fuel cladding and the reactor coolant.

RCS Loops – MODES 1 and 2 satisfy Criterion 2 of the NRC Policy Statement.

LCO

The purpose of this LCO is to require an adequate forced flow rate for core heat removal. Flow is represented by the number of RCPs in operation for removal of heat by the SGs. To meet safety analysis acceptance criteria for DNB, [four] pumps are required at rated power.

two ← 1

An OPERABLE RCS loop consists of an OPERABLE RCP in operation providing forced flow for heat transport and an

In MODES 1 and 2.

4

(continued)

BASES

LCO (continued)

OPERABLE SG in accordance with the Steam Generator Tube Surveillance Program.

APPLICABILITY

In MODES 1 and 2, the reactor is critical and thus has the potential to produce maximum THERMAL POWER. Thus, to ensure that the assumptions of the accident analyses remain valid, all RCS loops are required to be OPERABLE and in operation in these MODES to prevent DNB and core damage.

The decay heat production rate is much lower than the full power heat rate. As such, the forced circulation flow and heat sink requirements are reduced for lower, noncritical MODES as indicated by the LCOs for MODES 3, 4, and 5.

Operation in other MODES is covered by:

LCO 3.4.5, "RCS Loops – MODE 3";
LCO 3.4.6, "RCS Loops – MODE 4";
LCO 3.4.7, "RCS Loops – MODE 5, Loops Filled";
LCO 3.4.8, "RCS Loops – MODE 5, Loops Not Filled";
LCO 3.9.5, "Residual Heat Removal (RHR) and Coolant Circulation – High Water Level" (MODE 6); and
LCO 3.9.6, "Residual Heat Removal (RHR) and Coolant Circulation – Low Water Level" (MODE 6).

ACTIONS

A.1

If the requirements of the LCO are not met, the Required Action is to reduce power and bring the plant to MODE 3. This lowers power level and thus reduces the core heat removal needs and minimizes the possibility of violating DNB limits.

The Completion Time of 6 hours is reasonable, based on operating experience, to reach MODE 3 from full power conditions in an orderly manner and without challenging safety systems.

BASES (continued)

SURVEILLANCE REQUIREMENTS

SR 3.4.4.1

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

REFERENCES

1. FSAR, Section $\boxed{[\ \ ]}$.

14

1

No Significant Hazards Considerations - NUREG-1431 Section 3.04.04

13-Nov-99

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

NSHC Number	NSHC Text
R	In accordance with the criteria set forth in 10 CFR 50.92, PBNP has evaluated this proposed Technical Specifications change and determined it does not represent a significant hazards consideration. The following is provided in support of this conclusion. 1. Does the change involve a significant increase in the probability or consequences of an accident previously evaluated? The proposed change relocates requirements and surveillances for structures, systems, components or variables which did not meet the criteria for inclusion in Technical Specifications as identified in the 10CFR 50.36 Technical Specification Selection Criteria. The affected structures, systems, components or variables are not assumed to be initiators of analyzed events and are not assumed to mitigate accident or transient events. The requirements and surveillances for these affected structures, systems, components or variables will be relocated from the Technical Specifications to an appropriate administratively controlled document and maintained pursuant to 10CFR 50.59. Therefore, this change does not increase the probability or consequences of an accident previously evaluated. 2. Does the change create the possibility of a new or different kind of accident from any accident previously evaluated? The proposed change does not require a physical alteration of the plant (no new or different type of equipment will be installed) or change in parameters governing normal plant operation. The proposed change will not impose any different requirements and adequate control of information will be maintained. Thus, this change does not create the possibility of a new or different kind of accident from any accident previously evaluated. 3. Does this change involve a significant reduction in a margin of safety? The proposed change will not reduce a margin of safety because it has no impact on any safety analysis assumptions. In addition, the affected requirement will be relocated to an owner controlled document for which future changes will be evaluated pursuant to the requirements of 10CFR 50.59. Therefore, this change does not involve a reduction in a margin of safety.

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.4 RCS Loops — MODES 1 and 2

LCO 3.4.4 Two RCS loops shall be OPERABLE and in operation.

APPLICABILITY: MODES 1 and 2.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Requirements of LCO not met.	A.1 Be in MODE 3.	6 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
SR 3.4.4.1 Verify each RCS loop is in operation.	12 hours

B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.4 RCS Loops – MODES 1 and 2

BASES

BACKGROUND

The primary function of the RCS is removal of the heat generated in the fuel due to the fission process, and transfer of this heat, via the steam generators (SGs), to the secondary plant.

The secondary functions of the RCS include:

a. Moderating the neutron energy level to the thermal state, to increase the probability of fission;

b. Improving the neutron economy by acting as a reflector;

c. Carrying the soluble neutron poison, boric acid;

d. Providing a second barrier against fis sion product release to the environment; and

e. Removing the heat generated in the fuel due to fission product decay following a unit shutdown.

The reactor coolant is circulated through two loops connected in parallel to the reactor vessel, each containing an SG, a reactor coolant pump (RCP), and appropriate flow and temperature instrumentation for both control and protection. The reactor vessel contains the clad fuel. The SGs provide the heat sink to the isolated secondary coolant. The RCPs circulate the coolant through the reactor vessel and SGs at a sufficient rate to ensure proper heat transfer and prevent fuel damage. This forced circulation of the reactor coolant ensures mixing of the coolant for proper boration and chemistry control.

APPLICABLE SAFETY ANALYSES

Safety analyses contain various assumptions for the design bases accident initial conditions including RCS pressure, RCS temperature, reactor power level, core parameters, and safety system setpoints. The important aspect for this LCO is the reactor coolant forced flow rate, which is represented by the number of RCS loops in service.

Both transient and steady state analyses have been performed to establish the effect of flow on the departure from

BASES

APPLICABLE SAFETY ANALYSES (continued)

nucleate boiling (DNB). The transient and accident analyses for the plant have been performed assuming two RCS loops are in operation. The majority of the plant safety analyses are based on initial conditions at high core power or zero power. The accident analyses that are most important to RCP operation are the two pump coastdown, single pump locked rotor, single pump (broken shaft or coastdown), and rod withdrawal events (Ref. 1).

Steady state DNB analysis has been performed for the two RCS loop operation. For two RCS loop operation, the steady state DNB analysis, which generates the pressure and temperature Safety Limit (SL) (i.e., the departure from nucleate boiling ratio (DNBR) limit) assumes a maximum power level of 120% RTP. This is the design overpower condition for two RCS loop operation. The value for the accident analysis setpoint of the nuclear overpower (high flux) trip is 118% and is based on an analysis assumption that bounds possible instrumentation errors. The DNBR limit defines a locus of pressure and temperature points that result in a minimum DNBR greater than or equal to the critical heat flux correlation limit.

The plant is designed to operate with all RCS loops in operation to maintain DNBR above the SL, during all normal operations and anticipated transients. By ensuring heat transfer in the nucleate boiling region, adequate heat transfer is provided between the fuel cladding and the reactor coolant.

RCS Loops - MODES 1 and 2 satisfy Criterion 2 of the NRC Policy Statement.

LCO

The purpose of this LCO is to require an adequate forced flow rate for core heat removal. Flow is represented by the number of RCPs in operation for removal of heat by the SGs. To meet safety analysis acceptance criteria for DNB, two pumps are required at rated power.

In MODES 1 and 2, an OPERABLE RCS loop consists of an OPERABLE RCP in operation providing forced flow for heat transport and an OPERABLE SG in accordance with the Steam Generator Tube Surveillance Program.

BASES

APPLICABILITY

In MODES 1 and 2, the reactor is critical and thus has the potential to produce maximum THERMAL POWER. Thus, to ensure that the assumptions of the accident analyses remain valid, all RCS loops are required to be OPERABLE and in operation in these MODES to prevent DNB and core damage.

The decay heat production rate is much lower than the full power heat rate. As such, the forced circulation flow and heat sink requirements are reduced for lower, noncritical MODES as indicated by the LCOs for MODES 3, 4, and 5.

Operation in other MODES is covered by:

LCO 3.4.5, "RCS Loops - MODE 3";
LCO 3.4.6, "RCS Loops - MODE 4";
LCO 3.4.7, "RCS Loops - MODE 5, Loops Filled";
LCO 3.4.8, "RCS Loops - MODE 5, Loops Not Filled";
LCO 3.9.4, "Residual Heat Removal (RHR) and Coolant Circulation - High Water Level" (MODE 6); and
LCO 3.9.5, "Residual Heat Removal (RHR) and C oolant Circulation - Low Water Level" (MODE 6).

ACTIONS

A.1

If the requirements of the LCO are not met, the Required Action is to reduce power and bring the plant to MODE 3. This lowers power level and thus reduces the core heat removal needs and minimizes the possibility of violating DNB limits.

The Completion Time of 6 hours is reasonable, based on operating experience, to reach MODE 3 from full power conditions in an orderly manner and without challenging safety systems.

SURVEILLANCE REQUIREMENTS

SR 3.4.4.1

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

BASES

REFERENCES	1.	FSAR, Section 14 .

Cross-Reference Report - NUREG-1431 Section 3.04.05

ITS to CTS

13-Nov-99

ITS	CTS	DOC
LCO 3.04.05	15.03.01.A.01 *	A.02
	15.03.01.A.01.B	M.01
	15.03.01.A.02 *	A.02
	15.03.01.A.02.A	M.01
LCO 3.04.05 COND A	NEW	M.03
LCO 3.04.05 COND A RA A.1	15.03.03.A.03	M.04
	NEW	M.03
LCO 3.04.05 COND B	NEW	M.03
LCO 3.04.05 COND B RA B.1	15.03.03.A.03	M.04
	NEW	M.03
LCO 3.04.05 COND C	NEW	M.03
LCO 3.04.05 COND C RA C.1	NEW	M.03
LCO 3.04.05 COND C RA C.2	NEW	M.03
LCO 3.04.05 COND C RA C.3	NEW	M.03
LCO 3.04.05 NOTE	15.03.01.A.01.B.01	M.02
	15.03.01.A.01.B.01.A	A.01
	15.03.01.A.01.B.01.B	A.01
	15.03.01.A.01.B.01.C	LA.01
SR 3.04.05.01	NEW	M.03
SR 3.04.05.02	NEW	M.03
SR 3.04.05.03	NEW	M.03

Cross-Reference Report - NUREG-1431 Section 3.04.05
CTS to ITS

13-Nov-99

CTS	ITS	DOC
15.03.01.A.01 *	LCO 3.04.05	A.02
15.03.01.A.01.B	LCO 3.04.05	M.01
15.03.01.A.01.B.01	LCO 3.04.05 NOTE	M.02
15.03.01.A.01.B.01.A	LCO 3.04.05 NOTE	A.01
15.03.01.A.01.B.01.B	LCO 3.04.05 NOTE	A.01
15.03.01.A.01.B.01.C	LCO 3.04.05 NOTE	LA.01
15.03.01.A.02 *	LCO 3.04.05	A.02
15.03.01.A.02.A	LCO 3.04.05	M.01
15.03.03.A.03	LCO 3.04.05 COND A RA A.1	M.04
	LCO 3.04.05 COND B RA B.1	M.04

Description of Changes - NUREG-1431 Section 3.04.05

13-Nov-99

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

CTS:	ITS:
15.03.01.A.01.B.01.A	LCO 3.04.05 NOTE
15.03.01.A.01.B.01.B	LCO 3.04.05 NOTE

DOC Number	DOC Text
A.02	CTS 15.3.1.A.1 and 15.3.1.A.2 are both modified by Note *. This Note states, "Applicable only when one or more fuel assemblies are in the reactor vessel." Proposed ITS LCO 3.4.5 is applicable in MODE 3. ITS section 1.1, Definitions, states "A MODE shall correspond to any one inclusive combination of core reactivity condition, power level, average reactor coolant temperature, and reactor vessel head closure bolt tensioning specified in Table 1.1-1 with fuel in the reactor vessel." As specified in CTS 15.3.1.A.1 and 15.3.1.A.2, Note *, ITS 3.4.5 only applies with fuel in the reactor vessel. Therefore this change is administrative.

CTS:	ITS:
15.03.01.A.01 *	LCO 3.04.05
15.03.01.A.02 *	LCO 3.04.05

DOC Number	DOC Text
LA.01	CTS 15.3.1.A.1.b.(1).c requires the reactor trip breakers be open when both RCPs are deenergized per CTS 15.3.1.A.1.b.(1). This precludes inadvertent control rod withdrawal and the potential heat input to the reactor coolant. Proposed ITS 3.4.5 Note C relaxes this requirement by requiring the Rod Control System to not be capable of control rod withdrawal. The specific method of preventing control rod withdrawal is relocated to the Bases. This detail is not required to be in the technical specifications to provide adequate protection to the public health and safety. The requirement that the control rods are inserted and are not capable of being withdrawn is maintained.

CTS:	ITS:
15.03.01.A.01.B.01.C	LCO 3.04.05 NOTE

Description of Changes - NUREG-1431 Section 3.04.05

13-Nov-99

DOC Number	DOC Text
M.01	CTS 15.3.1.A.1.b requires at least one RCP to be in operation when the reactor is subcritical and the average reactor coolant temperature is greater than 350 F. CTS 15.3.1.A.2.a requires one steam generator to be operable when the average reactor coolant temperature is above 350 F. Proposed ITS 3.4.5 requires two RCS loops to be operable and one RCS loop to be in operation in Mode 3. ITS defines Mode 3 as a condition where the reactivity of the reactor core is < 0.99 (subcritical) and the average reactor coolant temperature is greater than or equal to 350 F. ITS 3.4.5 bases describe an operable RCS loop as consisting of one operable RCP and one operable SG. Therefore proposed ITS 3.4.5 requires two operable RCPs and two operable SGs in Mode 3. In this plant condition RCPs are used to provide forced circulation for decay heat removal and to ensure adequate mixing of boron. The decay heat removal requirements are low enough that a single RCS loop with a single RCP running is sufficient to remove core decay heat and provide adequate mixing of boron to prevent stratification. However, two RCS loops are required to be operable to ensure redundant capability for decay heat removal. Since this proposed change imposes additional requirements on plant operation in Mode 3, it is more restrictive.

CTS:	ITS:
15.03.01.A.01.B	LCO 3.04.05
15.03.01.A.02.A	LCO 3.04.05

M.02

CTS 15.3.1.A.1.b(1) allows both RCPs to be deenergized if:
a. No operations are permitted that would cause dilution of the reactor coolant system boron concentration,
b. Core outlet temperature is maintained at least 10 F below saturation temperature, and
c. The reactor trip breakers are open.

ITS 3.4.5 is modified by a Note that allows all RCPs to not be in operation for less than or equal to 1 hour per 8 hour period provided:
a. No operations are permitted that would cause reduction of the RCS boron concentration,
b. Core outlet temperature is maintained at least 10°F below saturation temperature, and
c. The Rod Control System is not capable of rod withdrawal.

This proposed Note is more restrictive, since it limits the time both RCPs can be deenergized to permit testing. This change is acceptable because unlimited operation with no RCPs operating could permit boron stratification. The one hour allowed time period is adequate to perform the desired tests. Operating experience has shown that boron stratification is not a problem during this short time period with no forced flow.

CTS:	ITS:
15.03.01.A.01.B.01	LCO 3.04.05 NOTE

DOC Number	DOC Text
M.03	CTS 15.3.1.A.1.b is revised to adopt ITS LCO 3.4.5, Actions A, B, C, SR 3.4.5.1, SR 3.4.5.2 and SR 3.4.5.3, to require that decay heat removal capability be available and in operation when the plant is in Mode 3. LCO 3.4.5 requires that at least two RCS loops be operable and one RCS loop be in operation to ensure that the safety limit criteria will be met for all of the postulated accidents. If one required RCS loop is inoperable, redundancy for heat removal is lost. Action A requires restoration of the required RCS loop to operable status within 72 hours. This time allowance is a justified period to be without the redundant, non-operating loop because a single loop in operation has a heat transfer capability greater than that needed to remove the decay heat produced in the reactor core and because of the low probability of a failure in the remaining loop occurring during this period. If restoration is not possible within 72 hours, Action B requires bringing the unit to MODE 4 within 12 hours. In MODE 4, the unit may be placed on the RHR System for decay heat removal. The additional Completion Time of 12 hours is compatible with required operations to achieve cooldown and depressurization from the existing plant conditions in an orderly manner and without challenging plant systems. If two RCS loops are inoperable or no RCS loop is in operation, except as during conditions permitted by the Note in the LCO section, Action C.1 requires all CRDMs to be de-energized by opening the RTBs or de energizing the MG sets. All operations involving a reduction of RCS boron concentration must be suspended, and action to restore one of the RCS loops to operable status and operation must be initiated. Boron dilution requires forced circulation for proper mixing, and opening the RTBs or de energizing the MG sets removes the possibility of an inadvertent rod withdrawal. The immediate Completion Time reflects the importance of maintaining operation for heat removal. SR 3.4.5.1 requires verification every 12 hours that the required RCS loops are in operation, providing forced flow for decay heat removal. The 12 hour frequency is sufficient considering other indications and alarms available to the operator in the control room to monitor RCS loop performance. SR 3.4.5.2 requires verification of SG operability by ensuring that the secondary side narrow range water level is greater than or equal to 30% for required RCS loops. If the water level is < 30%, the tubes may become uncovered and the associated loop may not be capable of providing the heat sink for removal of the decay heat. The 12 hour frequency is adequate in view of other indications available in the control room to alert the operator to a loss of SG level. SR 3.4.5.3 requires verification that the required RCPs are operable to ensure that safety analyses limits are met. The requirement also ensures that an additional RCP can be placed in operation, if needed, to maintain decay heat removal and reactor coolant circulation. Verification is performed by verifying proper breaker alignment and power availability to the required RCPs. Since this change imposes new requirements, it is more restrictive.

CTS:	ITS:
NEW	LCO 3.04.05 COND A
	LCO 3.04.05 COND A RA A.1
	LCO 3.04.05 COND B
	LCO 3.04.05 COND B RA B.1
	LCO 3.04.05 COND C
	LCO 3.04.05 COND C RA C.1

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DOC Number	DOC Text	
	NEW	LCO 3.04.05 COND C RA C.2 LCO 3.04.05 COND C RA C.3 SR 3.04.05.01 SR 3.04.05.02 SR 3.04.05.03

M.04

CTS 15.3.3.A.3 allows relaxation of the SI and RHR requirements specified in CTS 15.3.3.A.1.d and e. However, limitations on continued operation exist when the inoperable RHR component exceeds the specified allowed outage time. In the event the reactor is shutdown, the remaining methods of decay heat removal (DHR) are evaluated. If both RCS loops are available, the reactor can remain in hot shutdown. However, if one RHR loop is being relied upon to provide redundancy for DHR, the reactor is required to be maintained between 350 F and 140 F. Relying on RHR to provide redundancy for DHR implies another method of DHR is available. This other method must be a RCS loop, because this specification was entered when a RHR component became inoperable. Therefore the conditions would be, reactor coolant temperature is > 350 F with one operable RCS loop. This is consistent with the DHR requirements of CTS 15.3.1.A.1.b. Therefore the actions of CTS 15.3.3.A.3 that require the reactor be maintained between 350 F and 140 F would not be required.

However, proposed ITS 3.4.5 requires two operable RCS loops, with one RCS loop in operation in MODE 3. In the event only one RCS loop is operable, ITS LCO 3.4.5, Action A.1, requires restoration of the required RCS loop to operable status in 72 hours, otherwise be in Mode 4 (< 350 F) in 12 hours, per Required Action B.1. Requiring the reactor be cooled down to < 350 F under these conditions, is consistent with the requirements of CTS 15.3.3.A.3.

Although proposed ITS 3.4.5 allows 72 hours to restore the inoperable RCS loop, this is a more restrictive requirement, because CTS only requires one RCS loop to be in operation in hot shutdown. Additionally, specifying the time required to cool the reactor below 350 F places additional requirements on plant operations and is also more restrictive.

CTS:	ITS:
15.03.03.A.03	LCO 3.04.05 COND A RA A.1 LCO 3.04.05 COND B RA B.1

15.3 LIMITING CONDITIONS FOR OPERATION

15.3.1 REACTOR COOLANT SYSTEM

Applicability

Applies to the operating status of the Reactor Coolant System.

Objective

To specify those limiting conditions for operation of the Reactor Coolant System which must be met to ensure safe reactor operation.

Specification

A. OPERATIONAL COMPONENTS

1. Coolant Pumps* A.2

< See LCO 3.4.4 >

a. When the reactor is critical, both reactor coolant pumps shall be in operation.

(1) If one or both reactor coolant pump(s) cease operating, the reactor shall be placed in hot shutdown within 6 hours. M.1

LCO 3.4.5 b. When the reactor is subcritical and the average reactor coolant temperature is greater than 350°F, except for tests, at least one reactor coolant pump shall be in operation.

LCO 3.4.5 Note

(1) Both reactor coolant pumps may be deenergized provided: M.2

a. No operations are permitted that would cause dilution of the reactor coolant system boron concentration,

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

c. The reactor trip breakers are open. LA.1

c. At least one reactor coolant pump or residual heat removal system shall be in operation when a reduction is made in the boron concentration of the reactor coolant.

< See LCO 3.4.6, 3.4.7 & 3.4.8 >

2. Steam Generator* A.2

LCO 3.4.5 a. One steam generator shall be operable whenever the average reactor coolant temperature is above 350°F. M.1

Add Action A, B, C, SR 3.4.5.1, SR 3.4.5.2 and SR 3.4.5.3. See Insert 3.4.5-1. M.3

3. Components Required for Redundant Decay Heat Removal Capability*

a. Reactor coolant temperature less than 350°F and greater than 140°F.

(1) At least two of the decay heat removal methods listed shall be operable.

(a) Reactor Coolant Loop A, its associated steam generator and either reactor coolant pump

(b) Reactor Coolant Loop B, its associated steam generator and either reactor coolant pump

< See LCO 3.4.6 >

* Applicable only when one or more fuel assemblies are in the reactor vessel.

A.2

< See Section 3.5 >

the requirements of 15.3.3.A.1 within the time specified, the reactor shall be placed in the hot shutdown condition within six hours. The reactor shall be maintained in a condition with reactor coolant temperatures greater than 350°F, unless one residual heat removal loop is being relied upon to provide redundancy for decay heat removal. In this case the reactor shall be maintained between 350°F and 140°F. M.4

a. One residual heat removal pump may be out of service, provided the pump is restored to operable status within 72 hours. The other residual heat removal pump shall be operable. < See Section 3.5 >

b. One residual heat exchanger may be out of service for a period of no more than 72 hours.

c. Any valve in the system, required to function 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.

Section 3.4.5 CTS Markup Insert

Insert 3.4.5-1:

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One required RCS loop inoperable.	A.1 Restore required RCS loop to OPERABLE status.	72 hours
B. Required Action and associated Completion Time of Condition A not met.	B.1 Be in MODE 4.	12 hours
C. Two RCS loops inoperable. OR No RCS loop in operation.	C.1 Place the Rod Control System in a condition incapable of rod withdrawal. AND C.2 Suspend all operations involving a reduction of RCS boron concentration. AND C.3 Initiate action to restore one RCS loop to OPERABLE status and operation.	Immediately Immediately Immediately

Section 3.4.5 CTS Markup Insert

Insert 3.4.5-1 (continued):

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

Justification For Deviations - NUREG-1431 Section 3.04.05

13-Nov-99

JFD Number	JFD Text
01	The brackets have been removed and the proper plant specific information has been provided. Since the ISTS LCO 3.4.5 Condition C and associated Required Actions C.1 and C.2 were not used as a part of Point Beach's ITS, Condition D and associated Required Actions D.1, D.2 and D.3 of the ISTS have been relabeled as Condition C and Required Actions C.1, C.2 and C.3.

ITS:	NUREG:
B 3.04.05	B 3.04.05
LCO 3.04.05	LCO 3.04.05
LCO 3.04.05 COND C	LCO 3.04.05 COND D
LCO 3.04.05 COND C RA C.1	LCO 3.04.05 COND D RA D.1
LCO 3.04.05 COND C RA C.2	LCO 3.04.05 COND D RA D.2
LCO 3.04.05 COND C RA C.3	LCO 3.04.05 COND D RA D.3
N/A	LCO 3.04.05 COND C LCO 3.04.05 COND C RA C.1 LCO 3.04.05 COND C RA C.2
SR 3.04.05.02	SR 3.04.05.02

JFD Number	JFD Text
02	Only one RCS loop is required to be in operation in Mode 3 to provide sufficient flow to ensure adequate boron mixing and decay heat removal. Two RCS loops are required to be OPERABLE to provide redundant capability for decay heat removal. With the RTB's in the closed position and Rod Control System capable of rod withdrawal, accidental control rod withdrawal from subcritical is postulated and requires one RCS loop to be OPERABLE and in operation to ensure that the accident analysis limits are met. This analysis is therefore bounded by the decay heat removal redundancy requirements. Accordingly, TSTF-87, Rev.2 replacement discussion on RTB's was not adopted. The Mode 3 Limiting Condition for Operation becomes, "Two RCS loops shall be OPERABLE, and one RCS loop shall be in operation."

ITS:	NUREG:
B 3.04.05	B 3.04.05
LCO 3.04.05	LCO 3.04.05
N/A	LCO 3.04.05 A LCO 3.04.05 B
SR 3.04.05.01	SR 3.04.05.01

Justification For Deviations - NUREG-1431 Section 3.04.05

13-Nov-99

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

JFD Number	JFD Text
07	A sentence has been added to the LCO 3.4.5 Bases to clarify that the OPERABLE RCP and SG must be in the same loop for the RCS loop to be considered OPERABLE. This sentence was added because the NUREG-1431 Bases did not specify this condition for an OPERABLE RCS loop, and this condition was considered to be a necessary attribute for Point Beach. **ITS:** B 3.04.05 **NUREG:** B 3.04.05

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.5 RCS Loops - MODE 3

1 Two

one RCS loop shall be in operation.

2

LCO 3.4.5 [Two] RCS loops shall be OPERABLE, and either:

~~a. [Two] RCS loops shall be in operation when the Rod Control System is capable of rod withdrawal; or~~

~~b. One RCS loop shall be in operation when the Rod Control System is not capable of rod withdrawal .~~

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

All reactor coolant pumps may be de -energized for ≤ 1 hour per 8 hour period provided:

3 be not in operation

not be in operation

Approved TSTF 153

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

4

b. Core outlet temperature is maintain ed at least 10°F below saturation temperature / ; and

c. The Rod Control System is not capable of rod withdrawal.

APPLICABILITY: MODE 3.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One required RCS loop inoperable.	A.1 Restore required RCS loop to OPERABLE status.	72 hours
B. Required Action and associated Completion Time of Condition A not met.	B.1 Be in MODE 4.	12 hours

(continued)

CONDITION	REQUIRED ACTION		COMPLETION TIME
[~~C. One required RCS loop not in operation, and reactor trip breakers closed and Rod Control System capable of rod withdrawal.~~	~~C.1~~	~~Restore required RCS loop to operation.~~	~~1 hour~~
	~~OR~~		
	~~C.2~~	~~De-energize all control rod drive mechanisms (CRDMs).~~	~~1 hour~~]
C D. Two [Two] RCS loops inoperable.	C D.1	~~De-energize all CRDMs.~~ Place the Rod Control System in a condition incapable of rod withdrawal.	Immediately
OR	AND		
No RCS loop in operation.	D.2	Suspend all operations involving a reduction of RCS boron concentration.	Immediately
	AND		
	D.3	Initiate action to restore one RCS loop to OPERABLE status and operation.	Immediately

1

1

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SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.4.5.1	Verify required RCS loops are one RCS loop is in operation.	12 hours

2

(continued)

SURVEILLANCE REQUIREMENTS (continued)

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

B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.5 RCS Loops – MODE 3

BASES

BACKGROUND

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

two 1

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

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

two 1

APPLICABLE SAFETY ANALYSES

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

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TSTF-87 R.2

Therefore, in MODE 3 with ~~RTBs in~~ the ~~closed position and~~ Rod Control System capable of rod withdrawal, accidental control rod withdrawal from subcritical is postulated and requires at least [two] RCS loop~~s~~ to be OPERABLE and in operation to ensure that the accident analyses limits are

one 1

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

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

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

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

1 two one the

LCO

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

1 one is

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When

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

3 be not in operation

not be in operation

Approved TSTF 153

The Note permits all RCPs to be de-energized for ≤ 1 hour per 8 hour period. The purpose of the Note is to perform tests that are designed to validate various accident analyses values. ~~One of these tests is validation of the pump coastdown curve used as input to a number of accident analyses including a loss of flow accident. This test is generally performed in MODE 3 during the initial startup testing program, and as such should only be performed once. If, however, changes are made to the RCS that would cause a change to the flow characteristics of the RCS, the input~~

5

(continued)

BASES

LCO (continued)

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

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

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

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

b. Core outlet temperature is maintained at least 10 °F below saturation temperature, so that no vapor bubble may form and possibly cause a natural circulation flow obstruction~~.~~; and 4

4

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

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

7

APPLICABILITY

In MODE 3, this LCO ensures forced circulation of the reactor coolant to remove decay heat from the core and to provide proper boron mixing. ~~The most stringent condition of the LCO, that is, two RCS loops OPERABLE and two RCS loops in operation, applies to MODE 3 with RTBs in the~~ One RCS loop provides sufficient circulation for these purposes. However, one additional RCS loop is required to be OPERABLE to ensure redundant capability for decay heat removal. 2

(continued)

BASES

APPLICABILITY (continued)

~~closed position. The least stringent condition, that is, two RCS loops OPERABLE and one RCS loop in operation, applies to MODE 3 with the RTBs open.~~ 2

Operation in other MODES is covered by:

LCO 3.4.4, "RCS Loops – MODES 1 and 2";
LCO 3.4.6, "RCS Loops – MODE 4";
LCO 3.4.7, "RCS Loops – MODE 5, Loops Filled";
LCO 3.4.8, "RCS Loops – MODE 5, Loops Not Filled";
LCO 3.9.5 4 6 "Residual Heat Removal (RHR) and Coolant Circulation – High Water Level" (MODE 6); and
LCO 3.9.6 5 "Residual Heat Removal (RHR) and Coolant Circulation – Low Water Level" (MODE 6).

ACTIONS

A.1

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

B.1

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

~~C.1 and C.2~~

~~If the required RCS loop is not in operation, and the RTBs are closed and Rod Control System capable of rod withdrawal,~~ 1

(continued)

BASES

~~C.1 and C.2 (continued)~~

~~the Required Action is either to restore the required RCS loop to operation or to de-energize all CRDMs by opening the RTBs or de-energizing the motor generator (MG) sets. When the RTBs are in the closed position and Rod Control System capable of rod withdrawal, it is postulated that a power excursion could occur in the event of an inadvertent control rod withdrawal. This mandates having the heat transfer capacity of two RCS loops in operation. If only one loop is in operation, the RTBs must be opened. The Completion Times of 1 hour to restore the required RCS loop to operation or de-energize all CRDMs is adequate to perform these operations in an orderly manner without exposing the unit to risk for an undue time period.~~

1

C

D 1, D 2, and D 3

two

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

place the Rod Control System in a condition incapable of rod withdrawal (e.g.)

Approved TSTF-87 R.2

SURVEILLANCE REQUIREMENTS

SR 3.4.5.1

2

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

one RCS loop is

2

(continued)

BASES

SURVEILLANCE REQUIREMENTS (continued)

1 30

SR 3.4.5.2

SR 3.4.5.2 requires verification of SG OPERABILITY. SG OPERABILITY is verified by ensuring that the secondary side narrow range water level is ≥ [17]% for required RCS loops. If the SG secondary side narrow range water level is < [17]%, the tubes may become uncovered and the associated loop may not be capable of providing the heat sink for removal of the decay heat. The 12 hour Frequency is considered adequate in view of other indications available in the control room to alert the operator to a loss of SG level.

SR 3.4.5.3

Verification that the required RCPs are OPERABLE ensures that safety analyses limits are met. The requirement also ensures that an additional RCP can be placed in operation, if needed, to maintain decay heat removal and reactor coolant circulation. Verification is performed by verifying proper breaker alignment and power availability to the required RCPs.

REFERENCES

None.

No Significant Hazards Considerations - NUREG-1431 Section 3.04.05

13-Nov-99

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

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.5 RCS Loops – MODE 3

LCO 3.4.5 Two RCS loops shall be OPERABLE, and one RCS loop shall be in operation.

-----------------------------NOTE-----------------------------
All reactor coolant pumps may be not in operation for ≤ 1 hour per 8 hour period provided:

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

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

c. The Rod Control System is not capable of rod withdrawal.

APPLICABILITY: MODE 3.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One required RCS loop inoperable.	A.1 Restore required RCS loop to OPERABLE status.	72 hours
B. Required Action and associated Completion Time of Condition A not met.	B.1 Be in MODE 4.	12 hours

(continued)

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

SURVEILLANCE REQUIREMENTS

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