B 3.9 REFUELING OPERATIONS

B 3.9.1 Boron Concentration

BASES

BACKGROUND

The limit on the boron concentrations of the Reactor Coolant System (RCS), the refueling canal, and the refueling cavity during refueling ensures that the reactor remains subcritical during MODE 6. Refueling boron concentration is the soluble boron concentration in the coolant in each of these volumes having direct access to the reactor core during refueling.

The soluble boron concentration offsets the core reactivity and is measured by chemical analysis of a representative sample of the coolant in each of the volumes. The refueling boron concentration limit is specified in the COLR. Plant procedures ensure the specified boron concentration in order to maintain an overall core reactivity of $k_{eff} \leq 0.95$ during fuel handling, with control rods and fuel assemblies assumed to be in the most adverse configuration (least negative reactivity) allowed by plant procedures.

1

Point Beach design criteria

GDC 26 of 10 CFR 50, Appendix A, require that two independent reactivity control systems of different design principles be provided (Ref. 1). One of these systems must be capable of holding the reactor core subcritical under cold conditions. The Chemical and Volume Control System (CVCS) is the system capable of maintaining the reactor subcritical in cold conditions by maintaining the boron concentration.

2

The refueling cavity is then flooded with borated water from the Refueling Water Storage Tank (RWST) through the open reactor vessel by gravity feeding or by use of the Residual Heat Removal (RHR) System pumps. The refueling canal is flooded with borated water from the Spent Fuel Pool, RWST or Chemical and Volume Control System holdup tanks.

The reactor is brought to shutdown conditions before beginning operations to open the reactor vessel for refueling. After the RCS is cooled and depressurized and the vessel head is unbolted, the head is slowly removed to form the refueling cavity. The refueling canal and the refueling cavity are then flooded with borated water from the refueling water storage tank through the open reactor vessel by gravity feeding or by the use of the Residual Heat Removal (RHR) System pumps.

The pumping action of the RHR System in the RCS and the natural circulation due to thermal driving heads in the reactor vessel and refueling cavity mix the added concentrated boric acid with the water in the refueling

(continued)

BASES

BACKGROUND (continued)

canal. The RHR System is in operation during refueling (see LCO 3.9.5 "Residual Heat Removal (RHR) and Coolant Circulation-High Water Level," and LCO 3.9.6 "Residual Heat Removal (RHR) and Coolant Circulation -Low Water Level") to provide forced circulation in the RCS and assist in maintaining the boron concentrations in the RCS, the refueling canal, and the refueling cavity above the COLR limit.

APPLICABLE SAFETY ANALYSES

During refueling operations, the reactivity condition of the core is consistent with the initial conditions a ssumed for the boron dilution accident in the accident analysis and is conservative for MODE 6. The boron concentration limit specified in the COLR is based on the core reactivity at the beginning of each fuel cycle (the end of refueling) and includes an uncertainty allowance.

The required boron concentration and the plant refueling procedures that verify the correct fuel loading plan (including full core mapping) ensure that the k_{eff} of the core will remain ≤ 0.95 during the refueling operation. Hence, at least a 5% $\Delta k/k$ margin of safety is established during refueling.

During refueling, the water volume in the spent fuel pool, the transfer canal, the refueling canal, the refueling cavity, and the reactor vessel form a single mass. As a result, the soluble boron concentration is relatively the same in each of these volumes.

The limiting boron dilution accident analyzed occurs in MODE 5 (Ref. 2). A detailed discussion of this event is provided in Bases B 3.1.2 "SHUTDOWN MARGIN (SDM) - T_{avg} ~~≤ 200°F~~ "

The RCS boron concentration satisfies Criterion 2 of the NRC Policy Statement.

LCO

The LCO requires that a minimum boron concentr ation be maintained in the RCS, the refueling canal, and the refueling cavity while in MODE 6. The boron concentration limit specified in the COLR ensures that a core k_{eff} of

(continued)

BASES

LCO (continued)

≤ 0.95 is maintained during fuel handling operations. Violation of the LCO could lead to an inadvertent criticality during MODE 6.

APPLICABILITY

This LCO is applicable in MODE 6 to ensure that the fuel in the reactor vessel will remain subcritical. The required boron concentration ensures a $k_{eff} \leq 0.95$. Above MODE 6, LCO 3.1.1, "SHUTDOWN MARGIN (SDM) ~~- T_{avg} > 200°F," and LCO 3.1.2, "SHUTDOWN MARGIN (SDM) - T_{avg} ≤ 200°F,"~~ 4 ensure that an adequate amount of negative reactivity is available to shut down the reactor and maintain it subcritical.

ACTIONS

A.1 and A.2

Continuation of CORE ALTERATIONS or positive reactivity additions (including actions to reduce boron concentration) is contingent upon maintaining the unit in compliance with the LCO. If the boron concentration of any coolant volume in the RCS, the refueling canal, or the refueling cavity is less than its limit, all operations involving CORE ALTERATIONS or positive reactivity additions must be suspended immediately.

Suspension of CORE ALTERATIONS and positive reactivity additions shall not preclude moving a component to a safe position.

A.3

In addition to immediately suspending CORE ALTERATIONS or positive reactivity additions, boration to restore the concentration must be initiated immediately.

In determining the required combination of boration flow rate and concentration, no unique Design Basis Event must be satisfied. The only requirement is to restore the boron concentration to its required value as soon as possible. In order to raise the boron concentration as soon as possible, the operator should begin boration with the best source available for unit conditions.

(continued)

BASES

ACTIONS

A.3 (continued)

Once actions have been initiated, they must be continued until the boron concentration is restored. The restoration time depends on the amount of boron that must be injected to reach the required concentration.

SURVEILLANCE REQUIREMENTS

SR 3.9.1.1

This SR ensures that the coolant boron concentra tion in the RCS, the refueling canal, and the refueling cavity is within the COLR limits. The boron concentration ~~of the coolant in~~ 5 ~~each volume~~ 5 is determined periodically by chemical analysis.

A minimum Frequency of once every 72 hours is a reasonable amount of time to verify the boron concentration of representative samples. The Frequency is based on operating experience, which has shown 72 hours to be adequate.

REFERENCES

1. 10 CFR 50, Appendix A, GDC 26. FSAR Sections 1.3.5, 3.1 and 9.3. 1

2. FSAR, Chapter [15] 14.1.4 6

No Significant Hazards Considerations - NUREG-1431 Section 3.09.01

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. The proposed change removes the requirement to continuously monitor radiation levels in fuel handling areas, the containment and spent fuel storage pool, and the associated actions to be taken when this requirement is not met. These monitors indicate when the radiation in the respective area has exceeded a setpoint. There are no safety related automatic functions assumed in accident analyses that are performed by these monitors, and the monitors are not used to mitigate a design basis accident or transient. Therefore, this change does not involve an increase in the probability or consequences of an accident previously evaluated. 2. Does the change create the possibility of a new or different kind of accident from any accident previously evaluated? The proposed change does not involve any physical alteration of plant systems, structures or components, nor does it alter parameters governing normal plant operation. The proposed change does not introduce a new mode of operation or alter the method of normal plant operation. Therefore, the possibility of a new or different kind of accident from any accident previously evaluated is not created. 3. Does this change involve a significant reduction in a margin of safety? There are no margins of safety related to safety analyses that are dependent upon the proposed change. The requirements will continue to assure that limiting conditions for refueling are properly maintained. Therefore, this change does not involve a reduction in a margin of safety.

No Significant Hazards Considerations - NUREG-1431 Section 3.09.01

13-Nov-99

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.

No Significant Hazards Considerations - NUREG-1431 Section 3.09.01

13-Nov-99

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.

3.9 REFUELING OPERATIONS

3.9.1 Boron Concentration

LCO 3.9.1 Boron concentrations of the Reactor Coolant System, the refueling canal, and the refueling cavity shall be maintained within the limit specified in the COLR.

APPLICABILITY: MODE 6.

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. Boron concentration not within limit.	A.1	Suspend CORE ALTERATIONS.	Immediately
	AND		
	A.2	Suspend positive reactivity additions.	Immediately
	AND		
	A.3	Initiate action to restore boron concentration to within limit.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.9.1.1	Verify boron concentration is within the limit specified in the COLR.	72 hours

B 3.9 REFUELING OPERATIONS

B 3.9.1 Boron Concentration

BASES

BACKGROUND

The limit on the boron concentrations of the Reactor Coolant System (RCS), the refueling canal, and the refueling cavity during refueling ensures that the reactor remains subcritical during MODE 6. Refueling boron concentration is the soluble boron concentration in the coolant in each of these volumes having direct access to the reactor core during refueling.

The soluble boron concentration offsets the core reactivity and is measured by chemical analysis of a representative sample of the coolant in each of the volumes. The refueling boron concentration limit is specified in the COLR. Plant procedures ensure the specified boron concentration in order to maintain an overall core reactivity of $k_{eff} \leq 0.95$ during fuel handling, with control rods and fuel assemblies assumed to be in the most adverse configuration (least negative reactivity) allowed by plant procedures.

Point Beach design criteria require that two independent reactivity control systems of different design principles be provided (Ref. 1). One of these systems must be capable of holding the reactor core subcritical under cold conditions. The Chemical and Volume Control System (CVCS) is the system capable of maintaining the reactor subcritical in cold conditions by maintaining the boron concentration.

The reactor is brought to shutdown conditions before beginning operations to open the reactor vessel for refueling. After the RCS is cooled and depressurized and the vessel head is unbolted, the head is slowly removed to form the refueling cavity. The refueling cavity is then flooded with borated water from the Refueling Water Storage Tank (RWST) through the open reactor vessel by gravity feeding or by use of the Residual Heat Removal (RHR) System pumps. The refueling canal is flooded with borated water from the Spent Fuel Pool, RWST or Chemical and Volume Control System holdup tanks.

The pumping action of the RHR System in the RCS and the natural circulation due to thermal driving heads in the reactor vessel and refueling cavity mix the added concentrated boric acid with the water in the refueling canal. The RHR System is in operation during refueling (see

BASES

BACKGROUND (continued)

LCO 3.9.4, "Residual Heat Removal (RHR) and Coolant Circulation - High Water Level," and LCO 3.9.5, "Residual Heat Removal (RHR) and Coolant Circulation - Low Water Level") to provide forced circulation in the RCS and assist in maintaining the boron concentrations in the RCS, the refueling canal, and the refueling cavity above the COLR limit.

APPLICABLE SAFETY ANALYSES

During refueling operations, the reactivity condition of the core is consistent with the initial conditions assumed for the boron dilution accident in the accident analysis and is conservative for MODE 6. The boron concentration limit specified in the COLR is based on the core reactivity at the beginning of each fuel cycle (the end of refueling) and includes an uncertainty allowance.

The required boron concentration and the plant refueling procedures that verify the correct fuel loading plan (including full core mapping) ensure that the k_{eff} of the core will remain ≤ 0.95 during the refueling operation. Hence, at least a 5% $\Delta k/k$ margin of safety is established during refueling.

During refueling, the water volume in the spent fuel pool, the transfer canal, the refueling canal, the refueling cavity, and the reactor vessel form a single mass. As a result, the soluble boron concentration is relatively the same in each of these volumes.

The limiting boron dilution accident analyzed occurs in MODE 5 (Ref. 2). A detailed discussion of this event is provided in Bases B 3.1.1, "SHUTDOWN MARGIN (SDM)."

The RCS boron concentr ation satisfies Criterion 2 of the NRC Policy Statement.

LCO

The LCO requires that a minimum boron concentration be maintained in the RCS, the refueling canal, and the refueling cavity while in MODE 6. The boron concentration limit specified in the COLR ensures that a core k_{eff} of

BASES

LCO (continued)

$\leq$ 0.95 is maintained during fuel handling operations. Violation of the LCO could lead to an inadvertent criticality during MODE 6.

APPLICABILITY

This LCO is applicable in MODE 6 to ensure that the fuel in the reactor vessel will remain subcritical. The required boron concentration ensures a $k_{eff} \leq 0.95$. Above MODE 6, LCO 3.1.1, "SHUTDOWN MARGIN (SDM)," ensure that an adequate amount of negative reactivity is available to shut down the reactor and maintain it subcritical.

ACTIONS

A.1 and A.2

Continuation of CORE ALTERATIONS or positive reactivity additions (including actions to reduce boron concentration) is contingent upon maintaining the unit in compliance with the LCO. If the boron concentration of any coolant volume in the RCS, the refueling canal, or the refueling cavity is less than its limit, all operations involving CORE ALTERATIONS or positive reactivity additions must be suspended immediately.

Suspension of CORE ALTERATIONS and positive reactivity additions shall not preclude moving a component to a safe position.

A.3

In addition to immediately suspending CORE ALTERATIONS or positive reactivity additions, boration to restore the concentration must be initiated immediately.

In determining the required combination of boration flow rate and concentration, no unique Design Basis Event must be satisfied. The only requirement is to restore the boron concentration to its required value as soon as possible. In order to raise the boron concentration as soon as possible, the operator should begin boration with the best source available for unit conditions.

BASES

ACTIONS (continued)

Once actions have been initiated, they must be continued until the boron concentration is restored. The restoration time depends on the amount of boron that must be injected to reach the required concentration.

SURVEILLANCE REQUIREMENTS

SR 3.9.1.1

This SR ensures that the coolant boron concentration in the RCS, the refueling canal, and the refueling cavity is within the COLR limits. The boron concentration is determined periodically by chemical analysis.

A minimum Frequency of once every 72 hours is a reasonable amount of time to verify the boron concentration of representative samples. The Frequency is based on operating experience, which has shown 72 hours to be adequate.

REFERENCES

1. FSAR Sections 1.3.5, 3.1 and 9.3.
2. FSAR, Chapter 14.1.4.

Justification For Deviations - NUREG-1431 Section 3.09.02

13-Nov-99

JFD Number	JFD Text
01	NUREG 3.9.2, "Unborated Water Source Isolation Valves", is not applicable. A boron dilution event has been analyzed in FSAR, Section 14.1.4. Point Beach meets the applicable acceptance criteria, based on detection and termination of the dilution prior to a loss of shutdown margin. Isolation of unborated water sources is not assumed. Therefore, this LCO has not been adopted.

ITS:	NUREG:
N/A	LCO 3.09.02
	LCO 3.09.02 COND A
	LCO 3.09.02 COND A NOTE
	LCO 3.09.02 COND A RA A.1
	LCO 3.09.02 COND A RA A.2
	LCO 3.09.02 COND A RA A.3
	SR 3.09.02.01

1

3.9 REFUELING OPERATIONS

3.9.2 Unborated Water Source Isolation Valves

LCO 3.9.2 Each valve used to isolate unborated water sources shall be secured in the closed position.

APPLICABILITY: MODE 6.

ACTIONS

----------NOTE----------
Separate Condition entry is allowed for each unborated water source isolation valve.

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. ---------NOTE--------- Required Action A.3 must be completed whenever Condition A is entered. ---------- One or more valves not secured in closed position.	A.1	Suspend CORE ALTERATIONS.	Immediately
	AND		
	A.2	Initiate actions to secure valve in closed position.	Immediately
	AND		
	A.3	Perform SR 3.9.1.1.	4 hours

Cross-Reference Report - NUREG-1431 Section 3.09.03

ITS to CTS

13-Nov-99

ITS	CTS	DOC
LCO 3.09.02	15.03.08.03	A.02
	15.03.08.03	M.01
	15.03.08.03	M.02
	15.03.08.03	M.03
	15.03.08.03	M.04
	15.03.08.03	A.01
LCO 3.09.02 COND A	15.03.08.09	M.05
LCO 3.09.02 COND A RA A.1	15.03.08.09	M.05
LCO 3.09.02 COND A RA A.2	15.03.08.09	M.05
LCO 3.09.02 COND B	15.03.08.09	M.06
LCO 3.09.02 COND B RA B.1	15.03.08.09	M.06
LCO 3.09.02 COND B RA B.2	15.03.08.09	M.06
LCO 3.09.02 COND C	NEW	M.07
LCO 3.09.02 COND C RA C.1	NEW	M.07
SR 3.09.02.01	15.04.01 T 15.04.01-01 03.A	A.03
	15.04.01 T 15.04.01-01 03.A	L.02
SR 3.09.02.02	15.04.01 T 15.04.01-01 03	A.01
	15.04.01 T 15.04.01-01 03	A.03
SR 3.09.02.02 NOTE	NEW	L.01

Cross-Reference Report - NUREG-1431 Section 3.09.03

CTS to ITS

13-Nov-99

CTS	ITS	DOC
15.03.08.03	LCO 3.09.02	A.02
	LCO 3.09.02	M.01
	LCO 3.09.02	M.02
	LCO 3.09.02	M.03
	LCO 3.09.02	M.04
	LCO 3.09.02	A.01
15.03.08.09	LCO 3.09.02 COND A	M.05
	LCO 3.09.02 COND A RA A.1	M.05
	LCO 3.09.02 COND A RA A.2	M.05
	LCO 3.09.02 COND B	M.06
	LCO 3.09.02 COND B RA B.1	M.06
	LCO 3.09.02 COND B RA B.2	M.06
15.04.01 T 15.04.01-01 03	SR 3.09.02.02	A.01
	SR 3.09.02.02	A.03
15.04.01 T 15.04.01-01 03.A	SR 3.09.02.01	L.02
	SR 3.09.02.01	A.03

Description of Changes - NUREG-1431 Section 3.09.03

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:** 15.03.08.03; 15.04.01 T 15.04.01-01 03 **ITS:** LCO 3.09.02; SR 3.09.02.02
A.02	CTS 15.3.8.3 requires one audible indication in the containment available whenever core geometry is being changed. Proposed ITS LCO 3.9.2 requires one Source Range audible count rate function be OPERABLE in MODE 6. The purpose of the audible count rate is to alert operators to inadvertent reactivity additions. FSAR Section 14.1.4 credits the audible count rate in alerting operators to take mitigative actions in the event of a boron dilution event. LCO 3.9.2 Bases will stipulate that the audible count rate be audible in the control room to meet the OPERABILITY requirements of LCO 3.9.2. **CTS:** 15.03.08.03 **ITS:** LCO 3.09.02
A.03	CTS 15.4.1, Table 15.4.1-1, Item 3, requires a check and calibration of the neutron source range instrument channels in ALL plant conditions (check only required when instrument is not blocked). Proposed ITS SR 3.9.2.1 and SR 3.9.2.2 require the performance of a CHANNEL CHECK and a CHANNEL CALIBRATION, respectively, in MODE 6. Requirements for performing these tests in other modes are located in the surveillances associated with proposed ITS LCO 3.3.1. This is an administrative change, because ITS SR 3.9.2.1 and SR 3.9.2.2 are required to be performed under the same plant conditions as the source range surveillance requirements listed in CTS 15.4.1, Table 15.4.1-1, (i.e., when the plant is shutdown and any reactor vessel head bolt is less than fully tensioned). **CTS:** 15.04.01 T 15.04.01-01 03; 15.04.01 T 15.04.01-01 03.A **ITS:** SR 3.09.02.02; SR 3.09.02.01
L.01	CTS Table 15.4.1-1, item 3, CHANNEL CALIBRATION requirement for the neutron source range instrument channels is modified in ITS SR 3.9.2.2 by a Note, that excludes the neutron detectors from the calibration. This is a relaxation of requirements and is less restrictive. This is acceptable because the neutron detectors are passive devices with minimal drift and because of the difficulty associated with simulating a signal. **CTS:** NEW **ITS:** SR 3.09.02.02 NOTE

Description of Changes - NUREG-1431 Section 3.09.03

13-Nov-99

DOC Number	DOC Text	
L.02	CTS Table 15.4.1-1, item 3, requires the performance of a CHECK of the neutron monitors "once per shift". ITS SR 3.9.2.1 requires a CHANNEL CHECK to be performed every 12 hours. The nominal Point Beach shift duration is 8 hours. Therefore this change extends the nominal time between performances of these surveillances by 4 hours, resulting in a relaxation of the current requirement. This is acceptable based on other less formal, but more frequent, checks of channels during normal operational use of the displays associated with the LCO required channels, and the low probability of equipment malfunction during the additional (nominal 4 hour) time interval.	
	CTS:	**ITS:**
	15.04.01 T 15.04.01-01 03.A	SR 3.09.02.01
M.01	CTS 15.3.8.3 is applicable during refueling operations. CTS defines refueling operations as any operation involving movement of core components (those that affect the reactivity of the core) within the containment when the vessel head is removed. ITS 3.9.2 has Applicability in MODE 6. In MODE 6, the source range neutron flux monitors must be operable to determine changes in core reactivity. There are no other continuously monitored qualitative means available to check core reactivity levels. This change is more restrictive, because MODE 6 covers a much broader operational condition.	
	CTS:	**ITS:**
	15.03.08.03	LCO 3.09.02
M.02	CTS 15.3.8.3 requires core subcritical neutron flux to be continuously monitored by at least two neutron monitors when core geometry is being changed. ITS 3.9.2 requires two source range monitors to be operable in Mode 6. However, proposed ITS 3.9.2 Action A requires the suspension of core alterations and positive reactivity additions when one source range monitor is inoperable. This implies both source range monitors are required to be operable during core alterations and additions of positive reactivity. This change imposes additional requirements on plant operation and is more restrictive, because CTS 15.3.8.3 doesn't require both source range monitors to be operable during positive reactivity additions (other than changes in core geometry).	
	CTS:	**ITS:**
	15.03.08.03	LCO 3.09.02

DOC Number	DOC Text
M.03	CTS 15.3.8.3 requires one Source Range monitor to provide audible indication whenever core geometry is being changed. Proposed ITS LCO 3.9.2 requires one Source Range audible count rate circuit be OPERABLE in MODE 6. The Source Range count rate function provides indication to the operators of inadvertent reactivity additions. In order to provide the indication assumed during a boron dilution event, the audible count rate function should be available throughout MODE 6. Expanding the applicability to MODE 6 places additional requirements on plant operation and is therefore more restrictive. **CTS:** 15.03.08.03 **ITS:** LCO 3.09.02
M.04	The CTS 15.3.8.3 requires one source range monitor to be inservice during refueling operations when core geometry is not being changed. Proposed ITS 3.9.2 requires two source range neutron flux monitors to be OPERABLE during MODE 6. However, if one source range monitor is inoperable, continued operation in MODE 6 is permitted, once Core Alterations and positive reactivity additions are suspended. CTS 15.3.8.3 allows continued operation with the addition of positive reactivity (other than changes in core geometry) with one source range monitor inoperable. Therefore, this change imposes additional requirements on plant operation and is more restrictive. **CTS:** 15.03.08.03 **ITS:** LCO 3.09.02
M.05	CTS 15.3.8.9 specifies that in the event the limiting condition for monitoring core subcritical neutron flux is not met, refueling of the reactor shall cease. Additionally, work shall be initiated to correct the violated condition so that the specified limit is met, and no operations which may increase the reactivity of the core shall be made. In the event one source range monitor is inoperable and refueling operations are ceased, the additional actions of CTS 15.3.8.9 are no longer required, since the requirements of CTS 15.3.8.3 have now been met. Proposed ITS 3.9.2, Condition A, Required Actions A.1 and A.2 require the immediate suspension of Core Alterations and positive reactivity additions when one source range neutron monitor is inoperable. This change imposes more restrictive operational requirements, since CTS 15.3.8.3 allows the continuation of operations that may add positive reactivity with one inoperable source range monitor, as long as core geometry is not changed. **CTS:** 15.03.08.09 **ITS:** LCO 3.09.02 COND A LCO 3.09.02 COND A RA A.1 LCO 3.09.02 COND A RA A.2

DOC Number	DOC Text
M.06	CTS 15.3.8.9 is revised to provide additional actions if both source range monitors are inoperable (Condition B). ITS 3.9.2, Required Action B.1, specifies when both neutron monitors are inoperable, immediately initiate action to restore one source range neutron flux monitor to operable status. Additionally, ITS 3.9.2, Required Action B.2, requires verifying the boron concentration is within the limit specified in the COLR once per 12 hours. This will provide assurance that any changes in boron concentration will be detected, since both neutron monitors are inoperable and there is no direct method available to detect core reactivity. However, since CORE ALTERATIONS and positive reactivity additions are not to be made, the core reactivity condition is stabilized until the source range monitors are operable. This stabilized condition is determined by performing SR 3.9.1.1 to ensure the required boron exists. The completion time of 12 hours is sufficient to obtain and analyze a reactor coolant sample for boron and ensures that unplanned changes in boron concentration would be identified. This is reasonable considering the low probability of a change in core reactivity during this time period. **CTS:** 15.03.08.09 **ITS:** LCO 3.09.02 COND B LCO 3.09.02 COND B RA B.1 LCO 3.09.02 COND B RA B.2
M.07	ITS 3.9.2 Condition C is added to the CTS to address the loss of the audible count rate and requires action to be initiated immediately to isolate all unborated water sources and to immediately suspend CORE ALTERATIONS. The addition of this Condition and Required Actions is necessary to address the boron dilution event analyzed in the FSAR, which assumes a maximum unborated water flow and determines there is adequate time for operator action to mitigate the event. When Condition C is entered there is no assurance that prompt identification will occur, so Required Action C.1 dictates the closure of all unborated water source isolation valves to the RCS to preclude a boron dilution event. Additionally, with no audible count rate, prompt and definite indication of an inadvertent increase in core reactivity due to events such as an improperly loading fuel assembly, is lost. Therefore CORE ALTERATIONS are immediately suspended. This action will not prevent the completion of movement of a component to a safe position. **CTS:** NEW **ITS:** LCO 3.09.02 COND C LCO 3.09.02 COND C RA C.1

15.3.8 REFUELING

Applicability:

Applies to operating limitations during refueling operations.

Objective:

To ensure that no incident could occur during refueling operations that would affect public health and safety.

Specifications :

MODE 6 M.1

During refueling operations:

< See 3.9.4 >

1. The equipment hatch shall be closed and the personnel locks shall be capable of being closed. A temporary third door on the outside of the personnel lock shall be in place whenever both doors in a personnel lock are open (except for initial core loading).

< See 3.9.1 >

2. Radiation levels in fuel handling areas, the containment and spent fuel storage pool shall be monitored continuously.

Replace with Insert 3.9.3-1, LCO 3.9.2 A.2 M.2 M.3 M.4

3. Core subcritical neutron flux shall be continuously monitored by at least two neutron monitors, each with continuous visual indication in the control room and one with audible indication in the containment available whenever core geometry is being changed. When core geometry is not being changed, at least one neutron flux monitor shall be in service.

4. At least one residual heat removal loop shall be in operation. However, if refueling operations are affected by the residual heat removal loop flow, the operating residual heat removal loop may be removed from operation for up to one hour per eight hour period.

< See 3.9.5 >

5. During reactor vessel head removal and while loading and unloading fuel from the reactor, a minimum boron concentration of 2100 ppm* shall be maintained in the primary coolant system.

* This boron concentration value is in effect following U1R25 for Unit 1 and following U2R23 for Unit 2; and takes effect prior to loading fuel for those outages. Prior to U1R25, the Unit 1 boron concentration value of this specification is 1800 ppm. Prior to U2R23, the Unit 2 boron concentration value of this specification is 1800 ppm.

< See 3.9.1 >

A.1

< See 3.9.1 >

6. Direct communication between the control room and the operating floor of the containment shall be available whenever changes in core geometry are taking place.

< See 3.9.4 >

7. The Containment Purge and Vent System shall be operable. The Containment Purge and Vent System shall be demonstrated operable within 4 days prior to the start of and at least once per 7 days during refueling operations by verifying that Containment Purge and Vent isolation occurs on manual initiation and on high radiation test signal.

8. With the Containment Purge and Vent System inoperable, close the Purge and Vent containment penetrations.

Replace with Insert 3.9.3-2, Conditions A and B

M.5

M.6

< See 3.9.4 >
< See 3.9.1 >
< See 3.9.5 >
< See 3.9.1 >

9. If any of the specified limiting conditions in sections 1, 2, 3, 4, 5, and 6 are not met, refueling of the reactor shall cease. Work shall be initiated to correct the violated conditions so that the specified limits are met, and no operations which may increase the reactivity of the core shall be made.

Insert 3.9.3-3, New Condition C

M.7

Basis

The equipment and general procedures to be utilized during refueling are discussed in the Final Safety Analysis Report. Detailed instructions, the above specified precautions, and the design of the fuel handling equipment incorporating built-in interlocks and safety features, provide assurance that no incident could occur during the refueling operations that would result in a hazard to public health and safety.[1]

Whenever changes are not being made in core geometry, one flux monitor is sufficient. This permits maintenance of the instrumentation. Continuous monitoring of radiation levels (2. above) and neutron flux provides immediate indication of an unsafe condition. The residual heat pump is used to maintain a uniform boron concentration.

The shutdown margin indicated in Part 5 will keep the core subcritical, even if all control rods were withdrawn from the core. During refueling, the reactor refueling cavity is filled with approximately 275,000 gallons

< See 3.9.1 >

A.1

TABLE 15.4.1-1

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

NO.	CHANNEL DESCRIPTION	CHECK	CALIBRATE	TEST	PLANT CONDITIONS WHEN REQUIRED
1.	Nuclear Power Range	-	R	-	ALL
	-Heat Balance	S(1)	D(1,19)	-	ALL
	-Signal to delta T; bistable actions(rod stops, trips)	-	-	Q(1,2)	ALL
	-Compare results of the incore detector measurements to NIS axial flux difference	M(4,5,20)	-	-	PWR ← See 3.3 >
2.	Nuclear Intermediate Range	-	R	-	ALL
	-when not blocked	S(1)	-	-	ALL
	-logarithmic level;bistable action (rod stop, trips)	-	-	P	ALL
	Insert 3.9.3-4, New SR 3.9.2.2 Note ← L.1				
3.	Nuclear Source Range	-	R	-	ALL ← MODE 6 ← A.3
	-when not blocked	L.2 → 12 Hours → S	-	-	ALL ← MODE 6
	-Bistable action (alarm and trips)	-	-	P	ALL ← See 3.3 >
4.	Reactor Coolant Temperature	S	R	-	PWR,HOT S/D,COLD S/D
	-Overtemperature delta T	-	-	Q(1,2)	ALL
	-Overpower delta T	-	-	Q(1,2)	ALL
5.	Reactor Coolant Flow	S(1)	R	-	ALL
	-Analog and single loop loss-of-flow logic testing	-	-	Q(1,2)	ALL
	-Logic channel testing for reactor trip on loss of reactor coolant flow in both loops	-	-	R	ALL ← See 3.3 >
6.	Pressurizer Water Level	S(1)	R	Q(1,2)	ALL
7.	Pressurizer Pressure	S(1)	R	Q(1,2)	ALL
8.	Steam Generator Level	S(1)	R	Q(1,17)	ALL

NOTATION USED IN TABLE 15.4.1-1

A- Annually (12 months)
S- Each shift ← 12 hours ← L.2
D- Daily
W- Weekly
Q- Quarterly
M- Monthly
P- Prior to reactor criticality if not performed during the previous week.
R- Each refueling interval (18 months)
PWR- Power and Low Power Operation, as defined in Specifications 15.1.h. and 15.1.m.
HOT S/D- Hot Shutdown, as defined in Specification 15.1.g.1.
COLD S/D- Cold Shutdown, as defined in Specification 15.1.g.2.
REF S/D- Refueling Shutdown, as defined in Specification 15.1.g.3.
ALL- All conditions of operation, as defined in Specifications 15.1.g, h and m.

← See 3.3 >

NOTES USED IN TABLE 15.4.1-1

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

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

(3) Perform test of the isolation valve signal.

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

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

← See 3.3 >

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

(7) Radioactive Effluent Monitoring Instrumentation Surveillance Requirements are specified in Section 15.7.4. ← See 3.3 >

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

← See 3.1 >

(9) Test of Narrow Range Pressure, 3.0 psig, -3.0 psig excluded. ← See 3.3 >

Insert 3.9.3-1

LCO 3.9.2 Two source range neutron flux monitors shall be OPERABLE. M.4

AND

One source range audible count rate circuit shall be OPERABLE. A.2

APPLICABILITY: MODE 6. M.2 M.3

Insert 3.9.3-2

A. One required source range neutron flux monitor inoperable.	A.1	Suspend CORE ALTERATIONS.	Immediately
	AND		
	A.2	Suspend positive reactivity additions.	Immediately
B. Two required source range neutron flux monitors inoperable.	B.1	Initiate action to restore one source range neutron flux monitor to OPERABLE status.	Immediately
	AND		
	B.2	Perform SR 3.9.1.1.	Once per 12 hours

Insert 3.9.3-3

C. Required source range audible count rate circuit inoperable.	C.1	Initiate action to isolate unborated water sources.	Immediately

Insert 3.9.3-4

----------NOTE----------
Neutron detectors are excluded from CHANNEL CALIBRATION.

Justification For Deviations - NUREG-1431 Section 3.09.03

13-Nov-99

JFD Number	JFD Text
01	NUREG 1431, LCO 3.9.2, "Unborated Water Source Isolation Valves," was not adopted, based on the Point Beach design. Accordingly, the LCO and Bases have been renumbered.

ITS:	NUREG:
B 3.09.02	B 3.09.03
LCO 3.09.02	LCO 3.09.03
SR 3.09.02.01	SR 3.09.03.01
SR 3.09.02.02	SR 3.09.03.02

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

ITS:	NUREG:
B 3.09.02	B 3.09.03
LCO 3.09.02 COND A	LCO 3.09.03 COND A
LCO 3.09.02 COND B	LCO 3.09.03 COND B
SR 3.09.02.02	SR 3.09.03.02

JFD Number	JFD Text
03	NUREG 1431, LCO 3.9.3 Bases description of the Source Range Neutron Flux Monitors has been modified to reflect the wide-range Gamma-Metrics fission chamber detector installed to meet RG 1.97. Point Beach current licensing basis allows the use of this detector to meet the requirements of CTS 15.3.8.3, and therefore can be used to satisfy one of the two source range neutron flux monitors required by ITS LCO 3.9.2 in MODE 6. Although the fission chamber detector can adequately provide visual indication of core reactivity in the control room, one BF3 detector will still be required to provide an input signal to the audible count rate circuit.

ITS:	NUREG:
B 3.09.02	B 3.09.03

JFD Number	JFD Text
04	Reference to the General Design Criteria (GDC) of 10 CFR 50 Appendix A has been deleted from the Bases of the Technical Specifications, substituting reference to the appropriate section of the FSAR which specifies the Point Beach design criteria. Point Beach was constructed and licensed prior to the GDC being issued. The Point Beach construction permit was issued prior to the GDCs being issued in 1971. Point Beach was designed and constructed utilizing the 1967 proposed GDCs. Accordingly, reference has been provided to the appropriate criteria and section of the Point Beach FSAR which provides explanation of Point Beach's design basis.

ITS:	NUREG:
B 3.09.02	B 3.09.03

3.9 REFUELING OPERATIONS

3.9.3 Nuclear Instrumentation

LCO 3.9.3 Two source range neutron flux monitors shall be OPERABLE.

AND

One source range audible count rate circuit shall be OPERABLE

Approved TSTF-23

APPLICABILITY: MODE 6.

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. One required source range neutron flux monitor inoperable.	A.1	Suspend CORE ALTERATIONS.	Immediately
	AND		
	A.2	Suspend positive reactivity additions.	Immediately
B. Two required source range neutron flux monitors inoperable.	B.1	Initiate action to restore one source range neutron flux monitor to OPERABLE status.	Immediately
	AND		
	B.2	Perform SR 3.9.1.1.	~~4 hours~~ AND Once per 12 hours ~~thereafter~~
C. Required source range audible count rate circuit inoperable.	C.1	Initiate action to isolate unborated water sources.	Immediately

Approved TSTF-23

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.9.3.1	Perform CHANNEL CHECK.	12 hours
SR 3.9.3.2	----------NOTE---------- Neutron detectors are excluded from CHANNEL CALIBRATION. ---------- Perform CHANNEL CALIBRATION.	[18] months

B 3.9 REFUELING OPERATIONS

B 3.9.3 Nuclear Instrumentation

BASES

BACKGROUND

The source range neutron flux monitors are used during refueling operations to monitor the core reactivity condition. The installed source range neutron flux monitors are part of the Nuclear Instrumentation System (NIS). These detectors are located external to the reactor vessel and detect neutrons leaking from the core.

Replace with Insert B3.9.2-1.

The installed source range neutron flux monitors are BF3 detectors operating in the proportional region of the gas filled detector characteristic curve. The detectors monitor the neutron flux in counts per second. The instrument range covers six decades of neutron flux (1E+6 cps) with a [5]% instrument accuracy. The detectors also provide continuous visual indication in the control room and an audible alarm to alert operators to a possible dilution accident. The NIS is designed in accordance with the criteria presented in Reference 1.

count rate

Approved TSTF-23

APPLICABLE SAFETY ANALYSES

Two OPERABLE source range neutron flux monitors are required to provide a signal to alert the operator to unexpected changes in core reactivity such as with a boron dilution accident (Ref. 2) or an improperly loaded fuel assembly.

~~The need for a safety analysis for an uncontrolled boron dilution accident is eliminated by isolating all unborated water sources as required by LCO 3.9.2, "Unborated Water Source Isolation Valves."~~

Insert B3.9.2-2

The source range neutron flux monitors satisfy Criterion 3 of the NRC Policy Statement.

Approved TSTF-23

LCO

This LCO requires that two source range neutron flux monitors be OPERABLE to ensure that redundant monitoring capability is available to detect changes in core reactivity.

Insert B3.9.2-3

(continued)

BASES

1 → 2

APPLICABILITY

In MODE 6, the source range neutron flux monitors must be OPERABLE to determine changes in core reactivity. There are no other direct means available to check core reactivity levels. In MODES 2, 3, 4, and 5, these same installed source range detectors and circuitry are also required to be OPERABLE by LCO 3.3.1, "Reactor Trip System (RTS) Instrumentation."

the installed BF3

3

ACTIONS

A.1 and A.2

With only one source range neutron flux monitor OPERABLE, redundancy has been lost. Since these instruments are the only direct means of monitoring core reactivity conditions, CORE ALTERATIONS and positive reactivity additions must be suspended immediately. Performance of Required Action A.1 shall not preclude completion of movement of a component to a safe position.

B.1

With no source range neutron flux monitor OPERABLE, action to restore a monitor to OPERABLE status shall be initiated immediately. Once initiated, action shall be continued until a source range neutron flux monitor is restored to OPERABLE status.

B.2

With no source range neutron flux monitor OPERABLE, there are no direct means of detecting changes in core reactivity. However, since CORE ALTERATIONS and positive reactivity additions are not to be made, the core reactivity condition is stabilized until the source range neutron flux monitors are OPERABLE. This stabilized condition is determined by performing SR 3.9.1.1 to ensure that the required boron concentration exists.

Approved TSTF-96

once per 12

The Completion Time of 4 hours is sufficient to obtain and analyze a reactor coolant sample for boron concentration/ ~~The Frequency of once per 12 hours~~ ensures that unplanned changes in boron concentration would be identified. The 12 hour Frequency is reasonable, considering the low

Approved TSTF-96

and

(continued)

BASES

ACTIONS

B.2 (continued)

probability of a change in core reactivity during this time period.

Insert B3.9.2-4

Approved TSTF-23

SURVEILLANCE REQUIREMENTS

SR 3.9.3.1

SR 3.9.3.1 is the performance of a CHANNEL CHECK, wh ich is a comparison of the parameter indicated on one channel to a similar parameter on other channels. It is based on the assumption that the two indication channels should be consistent with core conditions. Changes in fuel loading and core geometry can result in significant differences between source range channels, but each channel should be consistent with its local conditions.

The Frequency of 12 hours is consistent with the CHANNEL CHECK Frequency specified similarly for the same instruments in LCO 3.3.1.

SR 3.9.3.2

SR 3.9.3.2 is the performance of a CHANNEL CALIBRATION every 18 months. This SR is modified by a Note stating that neutron detectors are excluded from the CHANNEL CALIBRATION. The CHANNEL CALIBRATION for the source range neutron flux monitors consists of obtaining the detector plateau or preamp discriminator curves, evaluating those curves, and comparing the curves to the manufacturer's data. The 18 month Frequency is bas ed on the need to perform this Surveillance under the conditions that apply during a plant outage. Operating experience has shown these components usually pass the Surveillance when performed at the 18 month Frequency.

The CHANNEL CALIBRATION also includes verification of the audible count rate

Approved TSTF-23

REFERENCES

1. 10 CFR 50, Appendix A, GDC 13, GDC 26, GDC 28, and GDC 29.

FSAR Sections 1.3.5, 3.1, 7.1 and 9.3

2. FSAR, Section [15.2.4].

14.1.4

NUREG Section 3.9.3 Markup Inserts

INSERT B3.9.2-1

There are three installed source range neutron flux monitors. Two are BF3 detectors operating in the proportional region of the gas filled detector characteristic curve and one is a fission chamber detector. The detectors monitor the neutron flux in counts per second. The instrument range covers six decades of neutron flux (1 to 1E+6 cps for the BF3 detector, and 0.1 to 1E+5 cps for the fission chamber detector). All three detectors also provide continuous visual indication in the control room. The BF3 detectors provide an audible count rate to alert operators to a possible dilution accident. The NIS is designed in accordance with the criteria presented in Reference 1.

INSERT B3.9.2-2

The audible count rate from the source range neutron flux monitors provides prompt and definite indication of a boron dilution event. The count rate increase is proportional to the subcritical multiplication factor and allows operators to promptly recognize the initiation of a boron dilution event. Prompt recognition of the initiation of the boron dilution event is consistent with the assumption of the safety analysis and is necessary to assure sufficient time is available for isolation of the primary water makeup source before SHUTDOWN MARGIN is lost (Ref. 2).

INSERT B3.9.2-3

To be OPERABLE, each monitor must provide visual indication in the control room. In addition, at least one of the two monitors must provide an OPERABLE audible count rate function in the control room, to alert operators to the initiation of a boron dilution event.

INSERT B3.9.2-4

C.1 and C.2

With no audible count rate available, prompt and definite indication of a boron dilution event, consistent with the assumptions of the safety analysis is lost. In this situation the boron dilution event may not be detected quickly enough to assure sufficient time is available for operations to manually isolate the unborated water sources and stop the dilution prior to the loss of SHUTDOWN MARGIN. Therefore, action must be taken to prevent an inadvertent boron dilution event from occurring. This is accomplished by isolating all of the unborated water flow paths to the reactor coolant system. Isolating these flow paths ensures an inadvertent dilution of the reactor coolant boron concentration is prevented. The Completion Time of "Immediately" assures a prompt response by operations and requires an operator to initiate actions to isolate an affected flow path immediately. Once actions are initiated they must be continued until all the necessary flow paths are isolated or the circuit is restored to OPERABLE status.

No Significant Hazards Considerations - NUREG-1431 Section 3.09.03

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.

No Significant Hazards Considerations - NUREG-1431 Section 3.09.03

13-Nov-99

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. The proposed change excludes neutron detectors from the calibration requirement. This is acceptable since the neutron detectors are passive devices with minimal drift, and because of the difficulty of simulating a meaningful signal. Therefore, this change does not involve an increase in the probability or consequences of an accident previously evaluated. 2. Does the change create the possibility of a new or different kind of accident from any accident previously evaluated? The proposed change does not involve any physical alteration of plant systems, structures or components, nor does it alter parameters governing normal plant operation. The proposed change does not introduce a new mode of operation or alter the method of normal plant operation. Therefore, the possibility of a new or different kind of accident from any accident previously evaluated is not created. 3. Does this change involve a significant reduction in a margin of safety? There are no margins of safety related to safety analyses that are dependent upon the proposed change. The requirements will continue to assure that limiting conditions for refueling are properly maintained. Therefore, this change does not involve a reduction in a margin of safety.

NSHC Number	NSHC Text
L.02	In accordance with the criteria set forth in 10 CFR 50.92, PBNP has evaluated this proposed Technical Specifications change and determined it does not represent a significant hazards consideration. The following is provided in support of this conclusion. 1. Does the change involve a significant increase in the probability or consequences of an accident previously evaluated? The proposed change does not involve any physical alteration of plant systems, structures or components, changes in parameters governing normal plant operation, or methods of operation. The proposed change extends the surveillance frequency for CHANNEL CHECKS from "each shift" (nominally 8 hours) to 12 hours. This is acceptable because the CHANNEL CHECK supplements less formal, but more frequent, checks of channels during normal operational use of the displays associated with the LCO required channels and because of the unlikelihood of a channel failure during this interval. Therefore, this change does not involve a significant increase in the probability or consequences of an accident previously evaluated. 2. Does the change create the possibility of a new or different kind of accident from any accident previously evaluated? The proposed change does not involve any physical alteration of plant systems, structures or components, nor does it alter parameters governing normal plant operation. The proposed change does not introduce a new mode of operation or alter the method of normal plant operation. Therefore, the possibility of a new or different kind of accident from any accident previously evaluated is not created. 3. Does this change involve a significant reduction in a margin of safety? There are no margins of safety related to safety analyses that are dependent upon the proposed change. The requirements will continue to assure that limiting conditions for the Nuclear Instrumentation are properly maintained. Therefore, this change does not involve a significant reduction in a margin of safety.

No Significant Hazards Considerations - NUREG-1431 Section 3.09.03

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.

3.9 REFUELING OPERATIONS

3.9.2 Nuclear Instrumentation

LCO 3.9.2 Two source range neutron flux monitors shall be OPERABLE.

AND

One source range audible count rate circuit shall be OPERABLE.

APPLICABILITY: MODE 6.

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. One required source range neutron flux monitor inoperable.	A.1	Suspend CORE ALTERATIONS.	Immediately
	AND		
	A.2	Suspend positive reactivity additions.	Immediately
B. Two required source range neutron flux monitors inoperable.	B.1	Initiate action to restore one source range neutron flux monitor to OPERABLE status.	Immediately
	AND		
	B.2	Perform SR 3.9.1.1.	Once per 12 hours
C. Required source range audible count rate circuit inoperable.	C.1	Initiate action to isolate unborated water sources.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.9.2.1	Perform CHANNEL CHECK.	12 hours
SR 3.9.2.2	----------NOTE---------- Neutron detectors are excluded from CHANNEL CALIBRATION. ---------- Perform CHANNEL CALIBRATION.	18 months

B 3.9 REFUELING OPERATIONS

B 3.9.2 Nuclear Instrumentation

BASES

BACKGROUND

The source range neutron flux monitors are used during refueling operations to monitor the core reactivity condition. The installed source range neutron flux monitors are part of the Nuclear Instrumentation System (NIS). These detectors are located external to the reactor vessel and detect neutrons leaking from the core.

There are three installed source range neutron flux monitors. Two are BF3 detectors operating in the proportional region of the gas filled detector characteristic curve, and one is a fission chamber detector. The detectors monitor the neutron flux in counts per second. The instrument range covers six decades of neutron flux (1 to 1E+6 cps for the BF3 detectors, and 0.1 to 1E+5 cps for the fission chamber detector). All three detectors also provide continuous visual indication in the control room. The BF3 detectors provide an audible count rate to alert operators to a possible dilution accident. The NIS is designed in accordance with the criteria presented in Reference 1.

APPLICABLE SAFETY ANALYSES

Two OPERABLE source range neutron flux monit ors are required to provide a signal to alert the operator to unexpected changes in core reactivity such as with a boron dilution accident (Ref. 2) or an improperly loaded fuel assembly.

The audible count rate from the source range neutro n flux monitors provides prompt and definite indication of a boron dilution event. The count rate increase is proportional to the subcritical multiplication factor and allows operators to promptly recognize the initiation of a boron dilution event. Prompt recognition of the initiation of the boron dilution event is consistent with the assumption of the safety analysis and is necessary to assure sufficient time is available for isolation of the primary water makeup source before SHUTDOWN MARGIN is lost (Ref. 2).

The source range neutron flux monitors satisfy Criterion 3 of the NRC Policy Statement.

BASES

LCO

This LCO requires that two source range neutron flux monitors be OPERABLE to ensure that redundant monitoring capability is available to detect changes in core reactivity.

To be OPERABLE, each monitor must provide visual indication in the control room. In addition, at least one of the two monitors must provide an OPERABLE audible count rate function in the control room to alert operators to the initiation of a boron dilution event.

APPLICABILITY

In MODE 6, the source range neutron flux monitors must be OPERABLE to determine changes in core reactivity. There are no other direct means available to check core reactivity levels. In MODES 2, 3, 4, and 5, the installed BF3 source range detectors and circuitry are also required to be OPERABLE by LCO 3.3.1, "Reactor Trip System (RTS) Instrumentation."

ACTIONS

A.1 and A.2

With only one source range neutron flux monitor OPERABLE, redundancy has been lost. Since these instruments are the only direct means of monitoring core reactivity conditions, CORE ALTERATIONS and positive reactivity additions must be suspended immediately. Performance of Required Action A.1 shall not preclude completion of movement of a component to a safe position.

B.1

With no source range neutron flux monitor OPERABLE, action to restore a monitor to OPERABLE status shall be initiated immediately. Once initiated, action shall be continued until a source range neutron flux monitor is restored to OPERABLE status.

B.2

With no source range neutron flux monitor OPERABLE, there are no direct means of detecting changes in core reactivity.

BASES

ACTIONS (continued)

However, since CORE ALTERATIONS and positive reactivity additions are not to be made, the core reactivity condition is stabilized until the source range neutron flux monitors are OPERABLE. This stabilized condition is determined by performing SR 3.9.1.1 to ensure that the required boron concentration exists.

The Completion Time of once per 12 hours is sufficient to obtain and analyze a reactor coolant sample for boron concentration and ensures that unplanned changes in boron concentration would be identified. The 12 hour Frequency is reasonable, considering the low probability of a change in core reactivity during this time period.

C.1 and C.2

With no audible count rate available, prompt and definite indication of a boron dilution event, consistent with the assumptions of the safety analysis is lost. In this situation the boron dilution event may not be detected quickly enough to assure sufficient time is available for operations to manually isolate the unborated water sources and stop the dilution prior to the loss of SHUTDOWN MARGIN. Therefore, action must be taken to prevent an inadvertent boron dilution event from occurring. This is accomplished by isolating all of the unborated water flow paths to the reactor coolant system. Isolating these flow paths ensures an inadvertent dilution of the reactor coolant boron concentration is prevented. The Completion Time of "Immediately" assures a prompt response by operations and requires an operator to initiate actions to isolate an affected flow path immediately. Once actions are initiated they must be continued until all the necessary flow paths are isolated or the circuit is restored to OPERABLE status.

SURVEILLANCE REQUIREMENTS

SR 3.9.2.1

SR 3.9.2.1 is the performance of a CHANNEL CHECK, which is a comparison of the parameter indicated on one channel to a similar parameter on other channels. It is based on the assumption that the two indication channels should be consistent with core conditions. Changes in fuel loading and core geometry can result in significant differences

BASES

SURVEILLANCE REQUIREMENTS (continued)

between source range channels, but each channel should be consistent with its local conditions.

The Frequency of 12 hours is consistent with the CHANNEL CHECK Frequency specified similarly for the same instruments in LCO 3.3.1.

SR 3.9.2.2

SR 3.9.2.2 is the performance of a CHANNEL CALIBRATION every 18 months. This SR is modified by a Note stating that neutron detectors are excluded from the CHANNEL CALIBRATION. The CHANNEL CALIBRATION for the source range neutron flux monitors consists of obtaining the detector plateau or preamp discriminator curves, evaluating those curves, and comparing the curves to the manufacturer's data. The CHANNEL CALIBRATION also includes verification of the audible count rate function. The 18 month Frequency is based on the need to perform this Surveillance under the conditions that apply during a plant outage. Operating experience has shown these components usually pass the Surveillance when performed at the 18 month Frequency.

REFERENCES

1. FSAR Sections 1.3.5, 3.1, 7.1 and 9.3.

2. FSAR, Section 14.1.4.

Cross-Reference Report - NUREG-1431 Section 3.09.04

ITS to CTS

13-Nov-99

ITS	CTS	DOC
LCO 3.09.03	15.03.08.01	A.02
	15.03.08.01	A.01
LCO 3.09.03 A	15.03.08.01	M.03
LCO 3.09.03 B	15.03.08.01	L.03
LCO 3.09.03 C	15.03.08.07	A.03
LCO 3.09.03 C.1	15.03.08.07	A.03
LCO 3.09.03 C.2	15.03.08.07	A.03
LCO 3.09.03 COND A	15.03.08.08	M.02
	15.03.08.09	L.02
LCO 3.09.03 COND A RA A.1	15.03.08.08	M.02
	15.03.08.09	L.02
LCO 3.09.03 COND A RA A.2	15.03.08.08	M.02
	15.03.08.09	L.02
SR 3.09.03.01	15.03.08.07 SR.01	M.01
SR 3.09.03.02	15.03.08.07 SR.02	L.01

Cross-Reference Report - NUREG-1431 Section 3.09.04

CTS to ITS

13-Nov-99

CTS	ITS	DOC
15.03.08.01	LCO 3.09.03	A.02
	LCO 3.09.03	A.01
	LCO 3.09.03 A	M.03
	LCO 3.09.03 B	L.03
	N/A	L.03
15.03.08.07	LCO 3.09.03 C	A.03
	LCO 3.09.03 C.1	A.03
	LCO 3.09.03 C.2	A.03
15.03.08.07 SR.01	SR 3.09.03.01	M.01
15.03.08.07 SR.02	SR 3.09.03.02	L.01
15.03.08.08	LCO 3.09.03 COND A	M.02
	LCO 3.09.03 COND A RA A.1	M.02
	LCO 3.09.03 COND A RA A.2	M.02
15.03.08.09	LCO 3.09.03 COND A	L.02
	LCO 3.09.03 COND A RA A.1	L.02
	LCO 3.09.03 COND A RA A.2	L.02

Description of Changes - NUREG-1431 Section 3.09.04

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:** 15.03.08.01 **ITS:** LCO 3.09.03
A.02	CTS 15.3.8.1 and 15.3.8.7 are applicable during refueling operations. Proposed ITS LCO 3.9.3 is applicable during Core Alterations and during movement of irradiated fuel assemblies within containment. As discussed in ITS Section 1.0, the CTS definition of refueling operations is being replaced with the ITS definition of Core Alterations. Since proposed ITS LCO 3.9.3 applicability also includes the movement of irradiated fuel inside the containment, the combination of the defined term and specified applicability is equivalent to the CTS 15.3.8.1 and 15.3.8.7 applicabilities, with the exception of the movement of components other than irradiated fuel within containment. As discussed in ITS Section 1.0, the movement of components other than irradiated fuel within containment has been deleted from the Technical Specifications and moved to licensee controlled documents, without an impact on safety. **CTS:** 15.03.08.01 **ITS:** LCO 3.09.03
A.03	CTS 15.3.8.7 requires the Containment Purge and Vent System be operable. Proposed ITS 3.9.3.c requires Containment Purge and Exhaust System penetrations providing direct access from the containment atmosphere to the outside atmosphere be either: 1. closed by a manual or automatic isolation valve, blind flange, or equivalent, or 2. capable of being closed by an OPERABLE Containment Purge and Exhaust Isolation System. Proposed ITS LCO 3.9.3.c requires Containment Purge and Exhaust penetrations that are not capable of being closed by an OPERABLE isolation system to be isolated. This is consistent with CTS 15.3.8.8, which requires the closure of the Containment Purge and Exhaust System penetrations, if the Containment Purge and Exhaust System is inoperable. Therefore, CTS 15.3.8.7 and ITS 3.9.3.c both allow continued refueling operations with the isolation of any required penetrations that are inoperable. **CTS:** 15.03.08.07 **ITS:** LCO 3.09.03 C LCO 3.09.03 C.1 LCO 3.09.03 C.2

Description of Changes - NUREG-1431 Section 3.09.04

13-Nov-99

DOC Number	DOC Text
L.01	CTS 15.3.8.7 requires the Containment Purge and Vent System be demonstrated operable within 4 days prior to the start of and at least once per 7 days during refueling operations by verifying that Containment Purge and Vent isolation occurs on manual initiation and on high radiation test signal. Proposed ITS SR 3.9.3.2 requires verification of each containment purge and exhaust valve actuates to the isolation position on an actual or simulated actuation signal once per 18 months. Adopting a less restrictive frequency for verification that each containment purge and exhaust valve actuates to the isolation position on an actual or simulated actuation signal is acceptable. The frequency of 18 months for SR 3.9.3.2 is consistent with other similar valve actuation tests. **CTS:** 15.03.08.07 SR.02 **ITS:** SR 3.09.03.02
L.02	CTS 15.3.8.9 specifies that in the event the limiting condition for the equipment hatch and personnel locks is not met, refueling of the reactor shall cease. Additionally, work shall be initiated to correct the violated condition so that the specified limit is met, and no operations which may increase the reactivity of the core shall be made. Proposed ITS 3.9.3, Condition A, Required Actions A.1 and A.2 specify to immediately suspend Core Alterations and the movement of irradiated fuel assemblies within containment. This is a relaxation of requirements and is less restrictive. However, this change is acceptable since performing ITS 3.9.3 Required Actions A.1 and A.2 places the plant in a condition whereby LCO 3.9.3 is no longer applicable. **CTS:** 15.03.08.09 **ITS:** LCO 3.09.03 COND A LCO 3.09.03 COND A RA A.1 LCO 3.09.03 COND A RA A.2

DOC Number	DOC Text
L.03	CTS 15.3.8.1 requires the personnel locks be capable of being closed during refueling operations. CTS 15.3.8.1 also requires a temporary third door on the outside of the personnel lock to be in place whenever both doors in a personnel lock are open (except for initial core loading.) ITS LCO 3.9.3.b requires one door in each airlock to be capable of being closed, during CORE ALTERATIONS, and during the movement of irradiated fuel assemblies within containment. This is consistent with NUREG 1431, LCO 3.9.4. The allowance to have containment personnel airlocks open during fuel movements and CORE ALTERATIONS is based on the Point Beach confirmatory dose calculation of a fuel handling accident. This calculation assumes a ground level release with acceptable radiological consequences. The personnel airlocks are not assumed to be closed during the fuel handling accident, nor are the airlocks assumed to be closed within any amount of time following the fuel handling accident. Although this change results in a relaxation of the current requirements, it is acceptable. Adopting the requirements of NUREG 1431 does not result in a significant reduction in the margin of safety, because the closure of the personnel airlock doors is not assumed to mitigate the radiological consequences of the fuel handling accident. **CTS:** 15.03.08.01 **ITS:** LCO 3.09.03 B N/A
M.01	CTS 15.3.8.7 is revised to adopt ITS SR 3.9.3.1, which requires a weekly verification that each required containment penetration is in the required status. This surveillance demonstrates that each of the containment purge and exhaust system penetrations that are not capable of being closed by an OPERABLE Containment Purge and Exhaust Isolation System is isolated. Since this change imposes new requirements, it is more restrictive and has no adverse impact on safety. **CTS:** 15.03.08.07 SR.01 **ITS:** SR 3.09.03.01

DOC Number	DOC Text
M.02	CTS 15.3.8.8 requires that if the Containment Purge and Vent System is inoperable, the Purge and Vent containment penetrations shall be closed, with no further actions specified if this condition can not be met. This allows the continuation of refueling operations, with or without the isolation of the containment penetrations. Proposed ITS LCO 3.9.3 is met if the Containment Purge and Vent penetrations are isolated or capable of being isolated. ITS 3.9.3, Condition A, is entered when one or more containment penetrations are not in the required status. ITS 3.9.3, Action A.1, requires Core Alterations to be suspended and Action A.2 requires the movement of irradiated fuel assemblies to be suspended. This change is more restrictive in that it introduces additional requirements on plant operation. **CTS:** 15.03.08.08 **ITS:** LCO 3.09.03 COND A LCO 3.09.03 COND A RA A.1 LCO 3.09.03 COND A RA A.2
M.03	CTS 15.3.8.1 states, "The equipment hatch shall be closed." ITS 3.9.3.a specifies the equipment hatch closure with "...held in place with all bolts." Specifying the equipment hatch be held in place with all bolts places additional requirements on unit operation and is therefore more restrictive. This change is necessary to ensure the equipment hatch will be sufficiently secured in place to minimize the escape of fission product radioactivity to the environment that may be released from the reactor core following an accident. **CTS:** 15.03.08.01 **ITS:** LCO 3.09.03 A

15.3.8 REFUELING

Applicability:

Applies to operating limitations during refueling operations.

Objective:

To ensure that no incident could occur during refueling operations that would affect public health and safety.

Specifications :

L.3
A.2
M.3

Replace with Insert 3.9.4-1, LCO 3.9.3

During refueling operations :

1. The equipment hatch shall be closed and the personnel locks shall be capable of being closed. ~~A temporary third door on the outside of the personnel lock shall be in place whenever both doors in a personnel lock are open (except for initial core loading).~~ L.3

2. Radiation levels in fuel handling areas, the containment and spent fuel storage pool shall be monitored continuously. < See 3.9.1 >

< See 3.9.3 >

3. Core subcritical neutron flux shall be continuously monitored by at least two neutron monitors, each with continuous visual indication in the control room and one with audible indication in the containment available whenever core geometry is being changed. When core geometry is not being changed, at least one neutron flux monitor shall be in service.

< See 3.9.5 >

4. At least one residual heat removal loop shall be in operation. However, if refueling operations are affected by the residual heat removal loop flow, the operating residual heat removal loop may be removed from operation for up to one hour per eight hour period.

5. During reactor vessel head removal and while loading and unloading fuel from the reactor, a minimum boron concentration of 2100 ppm* shall be maintained in the primary coolant system.

* This boron concentration value is in effect following U1R25 for Unit 1 and following U2R23 for Unit 2; and takes effect prior to loading fuel for those outages. Prior to U1R25, the Unit 1 boron concentration value of this specification is 1800 ppm. Prior to U2R23, the Unit 2 boron concentration value of this specification is 1800 ppm. < See 3.9.1 >

A.1

< See 3.9.1 >

6. Direct communication between the control room and the operating floor of the containment shall be available whenever changes in core geometry are taking place.

Replace with Insert 3.9.4-1, LCO 3.9.3.c

A.3

7. The Containment Purge and Vent System shall be operable. The Containment Purge and Vent System shall be demonstrated operable within 4 days prior to the start of and at least once per 7 days during refueling operations by verifying that Containment Purge and Vent isolation occurs on manual initiation and on high radiation test signal.

Replace with Insert 3.9.4-2, SR 3.9.3.2

L.1

8. With the Containment Purge and Vent System inoperable, close the Purge and Vent containment penetrations.

M.2

Replace with Insert 3.9.4-3, Condition A

< See 3.9.3 >

< See 3.9.1 >

< See 3.9.5 >

< See 3.9.1 >

9. If any of the specified limiting conditions in sections 1, 2, 3, 4, 5, and 6 are not met, refueling of the reactor shall cease. Work shall be initiated to correct the violated conditions so that the specified limits are met, and no operations which may increase the reactivity of the core shall be made.

Replace with Insert 3.9.4-3, Condition A

Insert 3.9.4-4, New SR 3.9.3.1

M.1

L.2

Basis

< See 3.9.1 >

The equipment and general procedures to be utilized during refueling are discussed in the Final Safety Analysis Report. Detailed instructions, the above specified precautions, and the design of the fuel handling equipment incorporating built-in interlocks and safety features, provide assurance that no incident could occur during the refueling operations that would result in a hazard to public health and safety.[1]

Whenever changes are not being made in core geometry, one flux monitor is sufficient. This permits maintenance of the instrumentation. Continuous monitoring of radiation levels (2. above) and neutron flux provides immediate indication of an unsafe condition. The residual heat pump is used to maintain a uniform boron concentration.

The shutdown margin indicated in Part 5 will keep the core subcritical, even if all control rods were withdrawn from the core. During refueling, the reactor refueling cavity is filled with approximately 275,000 gallons

A.1

Insert 3.9.4-1

M.3

LCO 3.9.3 The containment penetrations shall be in the following status:

a. The equipment hatch closed and held in place with all bolts;

L.3 b. One door in each air lock is capable of being closed;

c. Each Containment Purge and Exhaust System pe netration either:

1. closed by a manual or automatic isolation valve, blind flange, or equivalent, or

A.3 2. capable of being closed by an OPERABLE Containment Purge and Exhaust Isolation System.

APPLICABILITY: During CORE ALTERATIONS,
During movement of irradiated fuel assemblies within containment.

A.2

Insert 3.9.4-2

SR 3.9.3.2	----------NOTE---------- Not applicable to containment purge and exhaust valve(s) in penetrations closed to comply with LCO 3.9.3.c.1. ---------- Verify each required containment purge and exhaust valve actuates to the isolation position on an actual or simulated actuation signal.	18 months

L.1

Insert 3.9.4-3

A.	One or more containment penetrations not in required status.	A.1	Suspend CORE ALTERATIONS.	Immediately
		AND		
		A.2	Suspend movement of irradiated fuel assemblies within containment.	Immediately

Insert 3.9.4-4

SR	3.9.3.1	Verify each required containment penetration is in the required status.	7 days

M.1

Justification For Deviations - NUREG-1431 Section 3.09.04

13-Nov-99

JFD Number	JFD Text
01	LCO 3.9.2 "Unborated Water Source Isolation Valves" was not adopted, based on the Point Beach design. Accordingly, the LCO and Bases have been renumbered, and the reference to LCO 3.9.7 within the Bases has been revised to LCO 3.9.6, to reflect this exclusion to Section 3.9 of the ITS. **ITS:** / **NUREG:** B 3.09.03 / B 3.09.04 LCO 3.09.03 / LCO 3.09.04 SR 3.09.03.01 / SR 3.09.04.01 SR 3.09.03.02 / SR 3.09.04.02
02	NUREG 1431, LCO 3.3.6, Containment Purge and Exhaust Isolation Instrumentation, has not been adopted in ITS. The containment radiation detectors that initiate Containment Ventilation Isolation on high gaseous radioactivity are not classified as safety related. No FSAR chapter 14 accident relies on these detectors to function for Containment Ventilation Isolation. The offsite dose analysis for a fuel handling accident does not credit Containment Ventilation Isolation and conservatively assumes all radioactivity released during the accident is vented from containment (FSAR 7.3.3.a). **ITS:** / **NUREG:** B 3.09.03 / B 3.09.04
03	The closure or closure capability requirement for penetrations providing direct access from the containment atmosphere to the outside atmosphere has been limited to those penetrations associated with the Containment Purge and Exhaust System. Other containment penetrations are not included in the current licensing basis for Point Beach during CORE ALTERATIONS and movement of irradiated fuel assemblies within containment. The deletion of information pertaining to other penetrations has also resulted in the deletion of Reference 1 and the renumbering of subsequent references. Additionally, all Containment Purge and Exhaust System penetrations provide direct access from the containment to the outside atmosphere, therefore it is unnecessary to make this distinction. TSTF 312, adds a Note to NUREG 3.9.4, allowing penetration flowpaths with direct access from the containment atmosphere to the outside atmosphere to be unisolated under administrative controls. As a result of this deviation, TSTF 312 has not been incorporated. **ITS:** / **NUREG:** B 3.09.03 / B 3.09.04 LCO 3.09.03 C / LCO 3.09.04 C

Justification For Deviations - NUREG-1431 Section 3.09.04

13-Nov-99

JFD Number	JFD Text
04	Proposed ITS SR 3.9.3.2 is modified by a Note which clarifies that only unisolated containment penetrations are required to be tested. Isolated penetrations are already in their post accident position and therefore do not require testing. With this change, containment purge and exhaust valves that are in the closed position, as allowed by ITS LCO 3.9.3.c.1, will not result in a failure to meet the surveillance. **ITS:** B 3.09.03 **NUREG:** B 3.09.04 **ITS:** SR 3.09.03.02 NOTE **NUREG:** N/A
05	LCO 3.9.3 Bases description of the Containment Purge and Exhaust System has been modified to reflect Point Beach design. Point Beach Containment Purge and Exhaust System includes 36 inch purge and exhaust penetrations. During normal reactor operation at power, the containment may be continuously vented by use of the 1 inch containment gaseous and particulate sampling and monitoring penetrations. If containment pressure reaches 2 psig, the radiation monitoring forced ventilation pump is placed in service, which discharges to the purge exhaust filter units. The two valves in the purge penetration and the valve in the exhaust penetration used for the continuous ventilation of the containment will not close on a high radiation signal. These penetrations are too small to be within the scope of NUREG 737, item II.E.4.2(7), which requires automatic closure of containment purge and ventilation isolation valves over 3 in. nom. dia. on a high radiation signal. These 1 inch containment ventilation isolation valves do receive a close signal from the Containment Isolation Actuation System and are required to be OPERABLE in accordance with LCO 3.6.3. **ITS:** B 3.09.03 **NUREG:** B 3.09.04
06	Response Time Testing is not part of Point Beach's current licensing basis, and as such the reference to these surveillances has been deleted from the proposed ITS LCO 3.9.3 Bases. **ITS:** B 3.09.03 **NUREG:** B 3.09.04
07	The brackets have been removed and the proper plant specific information has been provided. **ITS:** B 3.09.03 **NUREG:** B 3.09.04 **ITS:** LCO 3.09.03 B **NUREG:** LCO 3.09.04 B **ITS:** SR 3.09.03.02 **NUREG:** SR 3.09.04.02

Justification For Deviations - NUREG-1431 Section 3.09.04

13-Nov-99

JFD Number	JFD Text
08	The containment equipment hatch at Point Beach is required to be held in place with all bolts in order to effect an adequate seal. As a result, "good engineering practice" to ensure the in-place bolts are equally spaced, is not an issue and can be deleted from the bases.

ITS:	NUREG:
B 3.09.03	B 3.09.04
LCO 3.09.03 A	LCO 3.09.04 A

1 → 3

3.9 REFUELING OPERATIONS

3.9.4 Containment Penetrations

1 → 3

8 → with all

LCO 3.9.4 The containment penetrations shall be in the following status:

a. The equipment hatch closed and held in place by [four] bolts;

7 → is capable of being → [is capable of being] ← Approved TSTF 68

3 → Containment Purge and Exhaust System

b. One door in each air lock closed; and

c. Each penetration ~~providing direct access from the containment atmosphere to the outside atmosphere~~ either:

3

1. closed by a manual or automatic isolation valve, blind flange, or equivalent, or

2. capable of being closed by an OPERABLE Containment Purge and Exhaust Isolation System.

APPLICABILITY: During CORE ALTERATIONS,
During movement of irradiated fuel assemblies within containment.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more containment penetrations not in required status.	A.1 Suspend CORE ALTERATIONS.	Immediately
	AND	
	A.2 Suspend movement of irradiated fuel assemblies within containment.	Immediately

1 → 3

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.9.4.1 (1 → 3)	Verify each required containment penetration is in the required status.	7 days
SR 3.9.4.2 (1 → 3)	Verify each required containment purge and exhaust valve actuates to the isolation position on an actual or simulated actuation signal.	[18] months (18 ← 7)

----------NOTE----------
Not applicable to containment purge and exhaust valve(s) in penetrations closed to comply with LCO 3.9.3.c.1.

4

1 → 3

B 3.9 REFUELING OPERATIONS

B 3.9.4 Containment Penetrations

3 ← 1

BASES

BACKGROUND

During CORE ALTERATIONS or movement of irradiated fuel assemblies within containment, a release of fission product radioactivity within containment will be restricted from escaping to the environment when the LCO requirements are met. In MODES 1, 2, 3, and 4, this is accomplished by maintaining containment OPERABLE as described in LCO 3.6.1, "Containment." In MODE 6, the potential for containment pressurization as a result of an accident is not likely; therefore, requirements to isolate the containment from the outside atmosphere can be less stringent. ~~The LCO requirements are referred to as "containment closure" rather than "containment OPERABILITY." Containment closure means that all potential escape paths are closed or capable of being closed.~~ Since there is no potential for containment pressurization, the Appendix J leakage criteria and tests are not required.

3

minimize the escape of ← 3

to the environment

The containment serves to contain fission product radioactivity that may be released from the reactor core following an accident, such that offsite radiation exposures are maintained well within the requirements of 10 CFR 100. Additionally, the containment provides radiation shielding from the fission products that may be present in the containment atmosphere following accident conditions.

The containment equipment hatch, which is part of the containment pressure boundary, provides a means for moving large equipment and components into and out of containment. During CORE ALTERATIONS or movement of irradiated fuel assemblies within containment, the equipment hatch must be held in place with all by at least four bolts. ~~Good engineering practice dictates that the bolts required by this LCO be approximately equally spaced.~~

with all

8

The containment air locks, which are also part of the containment pressure boundary, provide a means for personnel access during MODES 1, 2, 3, and 4 unit operation in accordance with LCO 3.6.2, "Containment Air Locks." Each air lock has a door at both ends. The doors are normally interlocked to prevent simultaneous opening when containment OPERABILITY is required. During periods of unit shutdown

1 → 3

BACKGROUND (Continued)

when containment closure is not required, the door interlock mechanism may be disabled, allowing both doors of an air lock to remain open for extended periods when frequent containment entry is necessary. During CORE ALTERATIONS or movement of irradiated fuel assemblies within containment, ~~containment closure is required; therefore, the door interlock mechanism may remain disabled, but~~ one air lock door must always remain closed.

The requirements for containment penetration closure ensure that a release of fission product radioactivity within containment will be restricted from escaping to the environment. The closure restrictions are sufficient to restrict fission product radioactivity release from containment due to a fuel handling accident during refueling.

The Containment Purge and Exhaust System ~~includes two subsystems. The normal subsystem~~ includes a 42 inch purge penetration and a 42 inch exhaust penetration. ~~The second subsystem, a minipurge system, includes an 8 inch purge penetration and an 8 inch exhaust penetration.~~ During MODES 1, 2, 3, and 4, the two valves in each of the ~~normal~~ purge and exhaust penetrations are secured in the closed position. ~~The two valves in each of the two minipurge penetrations can be opened intermittently, but are closed automatically by the Engineered Safety Features Actuation System (ESFAS).~~ Neither of the subsystems is subject to a Specification in MODE 5.

In MODE 6, large air exchangers are necessary to conduct refueling operations. The normal 42 inch purge system is used for this purpose, and all four valves are closed by the ESFAS in accordance with LCO 3.3.2, "Engineered Safety Feature Actuation System (ESFAS) Instrumentation."

~~The minipurge system remains operational in MODE 6, and all four valves are also closed by the ESFAS.~~

~~or~~

~~The minipurge system is not used in MODE 6. All four 8 inch valves are secured in the closed position.~~

~~The other containment penetrations that provide direct access from containment atmosphere to outside atmosphere~~

BACKGROUND (Continued)

~~must be isolated on at least one side. Isolation may be achieved by an OPERABLE automatic isolation valve, or by a manual isolation valve, blind flange, or equivalent. Equivalent isolation methods must be approved and may include use of a material that can provide a temporary, atmospheric pressure, ventilation barrier for the other containment penetrations during fuel movements (Ref. 1).~~

APPLICABLE SAFETY ANALYSES

During CORE ALTERATIONS or movement of irradiated fuel assemblies within containment, the most severe radiological consequences result from a fuel handling accident. The fuel handling accident is a postulated event that involves damage ~~to irradiated fuel (Ref. 2)~~. Fuel handling accidents, analyzed in Reference ~~3~~ include dropping a single ~~irradiated fuel assembly~~ and handling tool or a heavy object onto other irradiated fuel assemblies. The requirements of ~~LCO 3.9.7~~, "Refueling Cavity Water Level," and the minimum decay time of 100 hours prior to CORE ALTERATIONS ensure that the release of fission product radioactivity, subsequent to a fuel handling accident, results in doses that are well within the guideline values specified in 10 CFR 100. Standard Review Plan, Section 15.7.4, Rev. 1 ~~(Ref. 3)~~, defines "well within" 10 CFR 100 to be 25% or less of the 10 CFR 100 values. The acceptance limits for offsite radiation exposure will be 25% of 10 CFR 100 values or the NRC staff approved licensing basis (e.g., a specified fraction of 10 CFR 100 limits).

Containment penetrations satisfy Criterion 3 of the NRC Policy Statement.

LCO

Containment Purge and Exhaust System

This LCO limits the consequences of a fuel handling accident in containment by limiting the potential escape paths for fission product radioactivity released within containment. The LCO requires any penetration ~~providing direct access from the containment atmosphere to the outside atmosphere~~ to be closed except for the OPERABLE containment purge and exhaust penetrations. For the OPERABLE containment purge and exhaust penetrations, this LCO ensures that these penetrations are isolable by the Containment Purge and Exhaust Isolation System. The OPERABILITY requirements for this LCO ensure that the automatic purge and exhaust valve

~~[and the containment personnel airlocks.]~~

Approved TSTF 68

LCO (continued)

Approved TSTF 68

Insert B3.9.3-1.

closure ~~times~~ specified in the FSAR can be achieved ~~and, therefore, meet the assumptions used in the safety analysis to ensure that releases through the valves are terminated, such that radiological doses are within the acceptance limit~~.

APPLICABILITY

The containment penetration requirements are applicable during CORE ALTERATIONS or movement of irradiated fuel assemblies within containment because this is when there is a potential for a fuel handling accident. In MODES 1, 2, 3, and 4, containment penetration requirements are addressed by LCO 3.6.1. In MODES 5 and 6, when CORE ALTERATIONS or movement of irradiated fuel assemblies within containment are not being conducted, the potential for a fuel handling accident does not exist. Therefore, under these conditions no requirements are placed on containment penetration status.

ACTIONS

A.1 and A.2

Purge and Exhaust System

If the containment equipment hatch, air locks, or any containment penetration ~~that provides direct access from the containment atmosphere to the outside atmosphere~~ is not in the required status, including the Containment Purge and Exhaust Isolation System not capable of automatic actuation when the purge and exhaust valves are open, the unit must be placed in a condition where the isolation function is not needed. This is accomplished by immediately suspending CORE ALTERATIONS and movement of irradiated fuel assemblies within containment. Performance of these actions shall not preclude completion of movement of a component to a safe position.

SURVEILLANCE REQUIREMENTS

SR 3.9.4.1

This Surveillance demonstrates that each of the containment penetrations required to be in its closed position is in that position. The Surveillance on the open purge and exhaust valves will demonstrate that the valves are not blocked from closing. Also the Surveillance will

SURVEILLANCE REQUIREMENTS (continued)

demonstrate that each valve operator has motive power, which will ensure that each valve is capable of being closed by an OPERABLE automatic containment purge and exhaust isolation signal.

The Surveillance is performed every 7 days during CORE ALTERATIONS or movement of irradiated fuel assemblies within containment. The Surveillance interval is selected to be commensurate with the normal duration of time to complete fuel handling operations. A surveillance before the start of refueling operations will provide two or three surveillance verifications during the applicable period for this LCO. ~~As such, this Surveillance ensures that a postulated fuel handling accident that releases fission product radioactivity within the containment will not result in a release of fission product radioactivity to the environment.~~

~~in excess of those recommended by Standard Review Plan Section 15.7.4 (Reference 3.)~~

Approved TSTF 68

SR 3.9.4.2

This Surveillance demonstrates that each containment purge and exhaust valve actuates to its isolation position on manual initiation or on an actual or simulated high radiation signal. The 18 month Frequency maintains consistency with other similar ESFAS instrumentation and valve testing requirements. ~~In LCO 3.3.6, the Containment Purge and Exhaust Isolation instrumentation requires a CHANNEL CHECK every 12 hours and a COT every 92 days to ensure the channel OPERABILITY during refueling operations. Every 18 months a CHANNEL CALIBRATION is performed. The system actuation response time is demonstrated every 18 months, during refueling, on a STAGGERED TEST BASIS.~~ SR 3.6.3.5 demonstrates that the isolation time of each valve is in accordance with the Inservice Testing Program requirements. These Surveillances performed during MODE 6 will ensure that the valves are capable of closing after a postulated fuel handling accident to limit a release of fission product radioactivity from the containment.

The SR is modified by a Note stating that this demonstration is not applicable to valves in isolated penetrations. LCO 3.9.3.c.1 provides the option to close penetrations in lieu of requiring automatic isolation capability.

REFERENCES

~~1. GPU Nuclear Safety Evaluation SE-0002000-001, Rev. 0, May 20, 1988.~~

FSAR, Section [15.4.5].

NUREG-0800, Section 15.7.4, Rev. 1, July 1981.

Insert B3.9.3-1:

The containment personnel airlock doors may be open during movement of irradiated fuel in the containment and during CORE ALTERATIONS provided that one door is capable of being closed in the event of a fuel handling accident. Should a fuel handling accident occur inside containment, one personnel airlock door will be closed following an evacuation of containment.

No Significant Hazards Considerations - NUREG-1431 Section 3.09.04

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.

No Significant Hazards Considerations - NUREG-1431 Section 3.09.04

13-Nov-99

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. The proposed change adopts a less restrictive frequency for verifying containment purge and exhaust valves actuate to the isolation position on an actual or simulated actuation signal. The logic associated with this function is adequately tested per LCO 3.3.6, and the proposed frequency of 18 months is consistent with other similar valve actuation tests. Therefore, this change does not involve a significant increase in the probability or consequences of an accident previously evaluated. 2. Does the change create the possibility of a new or different kind of accident from any accident previously evaluated? The proposed change does not involve any physical alteration of plant systems, structures or components, nor does it alter parameters governing normal plant operation. The proposed change does not introduce a new mode of operation or alter the method of normal plant operation. Therefore, the possibility of a new or different kind of accident from any accident previously evaluated is not created. 3. Does this change involve a significant reduction in a margin of safety? There are no margins of safety related to safety analyses that are dependent upon the proposed change. The requirements will continue to assure that limiting conditions for refueling are properly maintained. Therefore, this change does not involve a significant reduction in a margin of safety.

NSHC Number	NSHC Text
L.02	In accordance with the criteria set forth in 10 CFR 50.92, PBNP has evaluated this proposed Technical Specifications change and determined it does not represent a significant hazards consideration. The following is provided in support of this conclusion. 1. Does the change involve a significant increase in the probability or consequences of an accident previously evaluated? The proposed change does not involve any physical alteration of plant systems, structures or components, changes in parameters governing normal plant operation, or methods of operation. This change limits the actions required in the event the equipment hatch and/or personnel locks are not in the required status, as specified in ITS LCO 3.9.3. However, this change is acceptable since performing ITS 3.9.3 Required Actions A.1 and A.2 places the plant in a condition whereby LCO 3.9.3 is no longer applicable. Therefore, this change does not involve an increase in the probability or consequences of an accident previously evaluated. 2. Does the change create the possibility of a new or different kind of accident from any accident previously evaluated? The proposed change does not involve any physical alteration of plant systems, structures or components, nor does it alter parameters governing normal plant operation. The proposed change does not introduce a new mode of operation or alter the method of normal plant operation. Therefore, the possibility of a new or different kind of accident from any accident previously evaluated is not created. 3. Does this change involve a significant reduction in a margin of safety? There are no margins of safety related to safety analyses that are dependent upon the proposed change. The requirements will continue to assure that limiting conditions for refueling are properly maintained. Therefore, this change does not involve a reduction in a margin of safety.

NSHC Number	NSHC Text
L.03	In accordance with the criteria set forth in 10 CFR 50.92, PBNP has evaluated this proposed Technical Specifications change and determined it does not represent a significant hazards consideration. The following is provided in support of this conclusion. CTS 15.3.8.1 requires the personnel locks be capable of being closed during refueling operations. CTS 15.3.8.1 also requires a temporary third door on the outside of the personnel lock to be in place whenever both doors in a personnel lock are open (except for initial core loading.) Proposed ITS LCO 3.9.3.b requires one door in each airlock to be capable of being closed, during CORE ALTERATIONS, and during the movement of irradiated fuel assemblies within containment, consistent with NUREG 1431, LCO 3.9.4 1. Does the change involve a significant increase in the probability or consequences of an accident previously evaluated? The proposed change does not involve any physical alteration of plant systems, structures or components, changes in parameters governing normal plant operation, or methods of operation. This change relaxes the requirement of personnel airlock doors during refueling operations. This change is acceptable, because the allowance to have containment personnel airlocks open during fuel movements and CORE ALTERATIONS is based on the Point Beach confirmatory dose calculation of a fuel handling accident. This calculation assumes a ground level release with acceptable radiological consequences. The personnel airlocks are not assumed to be closed during the fuel handling accident, nor are the airlocks assumed to be closed within any amount of time following the fuel handling accident. Therefore, this change does not involve a significant increase in the probability or consequences of an accident previously evaluated. 2. Does the change create the possibility of a new or different kind of accident from any accident previously evaluated? The proposed change does not involve any physical alteration of plant systems, structures or components, nor does it alter parameters governing normal plant operation. The proposed change does not introduce a new mode of operation or alter the method of normal plant operation. Therefore, the possibility of a new or different kind of accident from any accident previously evaluated is not created. 3. Does this change involve a significant reduction in a margin of safety? There are no margins of safety related to safety analyses that are dependent upon the proposed change, because the allowance to have containment personnel airlocks open during fuel movements and CORE ALTERATIONS is based on the Point Beach confirmatory dose calculation of a fuel handling accident. The requirements will continue to assure that limiting conditions for refueling are properly maintained. Therefore, this change does not involve a significant reduction in a margin of safety.

No Significant Hazards Considerations - NUREG-1431 Section 3.09.04

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.

3.9 REFUELING OPERATIONS

3.9.3 Containment Penetrations

LCO 3.9.3 The containment penetrations shall be in the following status:

a. The equipment hatch closed and held in place with all bolts;

b. One door in each air lock is capable of being closed; and

c. Each Containment Purge and Exhaust System penetration either:

1. closed by a manual or automatic isolation valve, blind flange, or equivalent, or

2. capable of being closed by an OPERABLE Containment Purge and Exhaust Isolation System.

APPLICABILITY: During CORE ALTERATIONS.
During movement of irradiated fuel assemblies within containment.

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. One or more containment penetrations not in required status.	A.1	Suspend CORE ALTERATIONS.	Immediately
	AND		
	A.2	Suspend movement of irradiated fuel assemblies within containment.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.9.3.1	Verify each required containment penetration is in the required status.	7 days
SR 3.9.3.2	----------NOTE---------- Not applicable to containment purge and exhaust valve(s) in penetrations closed to comply with LCO 3.9.3.c.1. ---------- Verify each required containment purge and exhaust valve actuates to the isolation position on an actual or simulated actuation signal.	18 months

B 3.9 REFUELING OPERATIONS

B 3.9.3 Containment Penetrations

BASES

BACKGROUND

During CORE ALTERATIONS or movement of irradiated fuel assemblies within containment, a release of fission product radioactivity within containment will be restricted from escaping to the environment when the LCO requirements are met. In MODES 1, 2, 3, and 4, this is accomplished by maintaining containment OPERABLE as described in LCO 3.6.1, "Containment." In MODE 6, the potential for containment pressurization as a result of an accident is not likely; therefore, requirements to isolate the containment from the outside atmosphere can be less stringent. Since there is no potential for containment pressurization, the Appendix J leakage criteria and tests are not required.

The containment serves to minimize the escape of fission product radioactivity to the environment that may be released from the reactor core following an accident, such that offsite radiation exposures are maintained well within the requirements of 10 CFR 100. Additionally, the containment provides radiation shielding from the fission products that may be present in the containment atmosphere following accident conditions.

The containment equipment hatch, which is part of the containment pressure boundary, provides a means for moving large equipment and components into and out of containment. During CORE ALTERATIONS or movement of irradiated fuel assemblies within containment, the equipment hatch must be held in place with all bolts.

The containment air locks, which are also part of the containment pressure boundary, provide a means for personnel access during MODES 1, 2, 3, and 4 unit operation in accordance with LCO 3.6.2, "Containment Air Locks." Each air lock has a door at both ends. The doors are normally interlocked to prevent simultaneous opening when containment OPERABILITY is required. During periods of unit shutdown when containment closure is not required, the door interlock mechanism may be disabled, allowing both doors of an air lock to remain open for extended periods when frequent containment entry is necessary. During CORE ALTERATIONS or movement of irradiated fuel assemblies within containment, one airlock door must always remain capable of being closed.

BASES

BACKGROUND (continued)

The requirements for containment purge and exhaust system penetration closure ensure that a release of fission product radioactivity within containment will be restricted to within regulatory limits.

The Containment Purge and Exhaust System includes a 36 inch purge penetration and a 36 inch exhaust penetration. During MODES 1, 2, 3, and 4, the two valves in each of the purge and exhaust penetrations are secured in the closed position. The Containment Purge and Exhaust System is not subject to a Specification in MODE 5.

In MODE 6, large air exchanges are necessary to conduct refueling operations. The 36 inch purge system is used for this purpose, and all four valves are closed by the Containment Purge and Exhaust Isolation Instrumentation.

APPLICABLE SAFETY ANALYSES

During CORE ALTERATIONS or movement of irradiated fuel assemblies within containment, the most severe radiological consequences result from a fuel handling accident. The fuel handling accident is a postulated event that involves damage to irradiated fuel (Ref. 1). Fuel handling accidents, analyzed in Reference 2, include dropping a single irradiated fuel assembly and handling tool or a heavy object onto other irradiated fuel assemblies. The requirements of LCO 3.9.6, "Refueling Cavity Water Level," and the minimum decay time of 100 hours prior to CORE ALTERATIONS ensure that the release of fission product radioactivity subsequent to a fuel handling accident, results in doses that are well within the guideline values specified in 10 CFR 100. Standard Review Plan, Section 15.7.4, Rev. 1 (Ref. 2), defines "well within" 10 CFR 100 to be 25% or less of the 10 CFR 100 values. The acceptance limits for offsite radiation exposure will be 25% of 10 CFR 100 values or the NRC staff approved licensing basis (e.g., a specified fraction of 10 CFR 100 limits).

Containment penetrations satisfy Criter ion 3 of the NRC Policy Statement.

BASES

LCO

This LCO limits the consequences of a fuel handling accident in containment by limiting the potential escape paths for fission product radioactivity released within containment. The LCO requires any Containment Purge and Exhaust System penetration to be closed except for the OPERABLE containment purge and exhaust penetrations. For the OPERABLE containment purge and exhaust penetrations, this LCO ensures that these penetrations are isolable by the Containment Purge and Exhaust Isolation System. The OPERABILITY requirements for this LCO ensure that the automatic purge and exhaust valve closure specified in the FSAR can be achieved.

The containment personnel airlock d oors may be open during movement of irradiated fuel in the containment and during CORE ALTERATIONS provided that one door is capable of being closed in the event of a fuel handling accident. Should a fuel handling accident occur inside containment, one personnel airlock door will be closed following an evacuation of containment.

APPLICABILITY

The containment penetration requirements are applicable during CORE ALTERATIONS or movement of irradiated fuel assemblies within containment because this is when there is a potential for a fuel handling accident. In MODES 1, 2, 3, and 4, containment penetration requirements are addressed by LCO 3.6.1. In MODES 5 and 6, when CORE ALTERATIONS or movement of irradiated fuel assemblies within containment are not being conducted, the potential for a fuel handling accident does not exist. Therefore, under these conditions no requirements are placed on containment penetration status.

ACTIONS

A.1 and A.2

If the containment equipment hatch, air locks, or any containment Purge and Exhaust System penetration is not in the required status, including the Containment Purge and Exhaust Isolation System not capable of automatic actuation when the purge and exhaust valves are open, the unit must be

BASES

ACTIONS (continued)

placed in a condition where the isolation function is not needed. This is accomplished by immediately suspending CORE ALTERATIONS and movement of irradiated fuel assemblies within containment. Performance of these actions shall not preclude completion of movement of a component to a safe position.

SURVEILLANCE REQUIREMENTS

SR 3.9.3.1

This Surveillance demonstrates that each of the containment penetrations required to be in its closed position is in that position. The Surveillance on the open purge and exhaust valves will demonstrate that the valves are not blocked from closing. Also the Surveillance will demonstrate that each valve operator has motive power, which will ensure that each valve is capable of being closed by an OPERABLE automatic containment purge and exhaust isolation signal.

The Surveillance is performed every 7 days during CORE ALTERATIONS or movement of irradiated fuel assemblies within containment. The Surveillance interval is selected to be commensurate with the normal duration of time to complete fuel handling operations. A surveillance before the start of refueling operations will provide two or three surveillance verifications during the applicable period for this LCO.

SR 3.9.3.2

This Surveillance demonstrates that each containment purge and exhaust valve actuates to its isolation position on manual initiation or on an actual or simulated high radiation signal. The 18 month Frequency maintains consistency with other similar ESFAS instrumentation and valve testing requirements. SR 3.6.3.5 demonstrates that the isolation time of each valve is in accordance with the Inservice Testing Program requirements. These Surveillances performed during MODE 6 will ensure that the valves are capable of closing after a postulated fuel handling accident to limit a release of fission product radioactivity from the containment.

BASES

SURVEILLANCE REQUIREMENTS (continued)

The SR is modified by a Note stating that this demonstration is not applicable to valves in isolated penetrations. LCO 3.9.3.c.1 provides the option to close penetrations in lieu of requiring automatic isolation capability.

REFERENCES

1. FSAR, Section 14.2.1.

2. NUREG-0800, Section 15.7.4, Rev. 1, July 1981.

Cross-Reference Report - NUREG-1431 Section 3.09.05

ITS to CTS

13-Nov-99

ITS	CTS	DOC
LCO 3.09.04	15.03.01.A.03.B	M.01
	15.03.01.A.03.B.03	M.01
	15.03.08.04	M.01
	15.03.08.04	A.01
LCO 3.09.04 COND A	15.03.01.A.03.B.02	A.02
	15.03.08.09	A.02
LCO 3.09.04 COND A RA A.1	15.03.01.A.03.B.02	A.02
	15.03.08.09	A.02
LCO 3.09.04 COND A RA A.2	15.03.01.A.03.B.02	A.02
	15.03.08.09	A.02
LCO 3.09.04 COND A RA A.3	15.03.01.A.03.B.02	A.02
	15.03.08.09	A.02
LCO 3.09.04 NOTE	15.03.08.04	M.02
SR 3.09.04.01	NEW	M.03

Cross-Reference Report - NUREG-1431 Section 3.09.05

CTS to ITS

13-Nov-99

CTS	ITS	DOC
15.03.01.A.03.B	LCO 3.09.04	M.01
15.03.01.A.03.B.02	LCO 3.09.04 COND A	A.02
	LCO 3.09.04 COND A RA A.1	A.02
	LCO 3.09.04 COND A RA A.2	A.02
	LCO 3.09.04 COND A RA A.3	A.02
15.03.01.A.03.B.03	LCO 3.09.04	M.01
15.03.08.04	LCO 3.09.04	M.01
	LCO 3.09.04	A.01
	LCO 3.09.04 NOTE	M.02
15.03.08.09	LCO 3.09.04 COND A	A.02
	LCO 3.09.04 COND A RA A.1	A.02
	LCO 3.09.04 COND A RA A.2	A.02
	LCO 3.09.04 COND A RA A.3	A.02

Description of Changes - NUREG-1431 Section 3.09.05

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:
15.03.08.04	LCO 3.09.04

A.02

CTS 15.3.8.9 requires the following actions, if at least one RHR loop is not in operation during refueling operations; cease refueling of the reactor, initiate actions to restore at least one RHR loop to operation, and suspend operations which may increase the reactivity of the core. Additionally, CTS 15.3.1.A.3.b.2 requires the suspension of operations causing an increase in the reactor decay heat load or a reduction in the reactor coolant boron concentration, and the immediate initiation of actions to return a decay heat removal method to operation, if no RHR loop is in operation with the reactor coolant temperature < 140F.

Proposed ITS LCO 3.9.4 requires one RHR loop be OPERABLE and in operation in MODE 6, when water level is greater than or equal to 23 ft above the top of reactor vessel flange. In the event the RHR loop requirements are not met, Condition A is entered and the Required Actions consist of suspending operations causing a reduction in boron concentration, suspending the loading of irradiated fuel assemblies in the core and initiating actions to satisfy RHR loop requirements.

These actions are consistent with the requirements of CTS 15.3.1.A.3.b.2 and CTS 15.3.8.9, and this change is therefore administrative.

CTS:	ITS:
15.03.01.A.03.B.02	LCO 3.09.04 COND A
	LCO 3.09.04 COND A RA A.1
	LCO 3.09.04 COND A RA A.2
	LCO 3.09.04 COND A RA A.3
15.03.08.09	LCO 3.09.04 COND A
	LCO 3.09.04 COND A RA A.1
	LCO 3.09.04 COND A RA A.2
	LCO 3.09.04 COND A RA A.3

Description of Changes - NUREG-1431 Section 3.09.05

13-Nov-99

DOC Number	DOC Text
M.01	The requirements of CTS 15.3.1.A.3.b and CTS 15.3.8.4 are combined and revised into proposed ITS LCOs 3.4.7, 3.4.8, 3.9.4, and 3.9.5. The following is a description of the requirements contained in ITS 3.9.4. The requirements contained in ITS 3.4.7, 3.4.8 and 3.9.5 are discussed in the DOCs for those LCOs. CTS 15.3.8.4 requires at least one RHR loop in operation during refueling operations. CTS 15.3.1.A.3.b requires both RHR loops be operable with reactor coolant temperature <140F. However one RHR loop may be out of service when the reactor vessel head is removed and the refueling cavity is flooded. Proposed ITS 3.9.4 requires one RHR loop be OPERABLE and in operation in MODE 6, when water level is greater than or equal to 23 ft above the top of reactor vessel flange. Specifying the extent to which the refueling cavity is flooded, places an additional requirement on plant operation and is therefore more restrictive. With the introduction of ITS LCO 3.9.6, "Refueling Cavity Water Level", CORE ALTERATIONS and the movement of irradiated fuel assemblies within containment can only occur when water level is greater than or equal to 23 ft above the top of the reactor vessel flange. Therefore, since ITS 3.9.4 requires one OPERABLE RHR loop to be in operation under these conditions, it is consistent with CTS 15.3.8.4. The portion of CTS 15.3.1.A.3.b that applies when the refueling cavity is not flooded, is addressed in proposed ITS LCOs 3.4.7, 3.4.8 and 3.9.5. **CTS:** — **ITS:** 15.03.01.A.03.B — LCO 3.09.04 15.03.01.A.03.B.03 — LCO 3.09.04 15.03.08.04 — LCO 3.09.04
M.02	The CTS 15.3.8.4 allowance to remove an RHR loop from operation for up to one hour per eight hour period during refueling operations has been modified by the addition of the phrase "provided no operations are permitted that would cause reduction of the Reactor Coolant System boron concentration." This is an added restriction with respect to removing an RHR loop from operation. The restriction, which prohibits operations that would cause dilution of the RCS boron concentration, is necessary because forced circulation is required to ensure mixing of the borated coolant. Mixing of the coolant is an assumption in the "Dilution During Refueling" event described in FSAR Chapter 14. **CTS:** — **ITS:** 15.03.08.04 — LCO 3.09.04 NOTE

Description of Changes - NUREG-1431 Section 3.09.05

13-Nov-99

DOC Number	DOC Text
M.03	The addition of SR 3.9.4.1 is proposed to demonstrate that one RHR loop is in operation and circulating reactor coolant. Verification includes flow rate, temperature, or pump status monitoring, which help ensure that forced flow is providing decay heat removal capability and mixing of the borated coolant to prevent thermal and boron stratification in the core. The Frequency of 12 hours is sufficient, considering the indications available to the operator for monitoring the RHR System in the control room. The addition of this SR places an additional operating requirement on the plant and is, therefore, more restrictive. **CTS:** NEW **ITS:** SR 3.09.04.01

15.3.8 REFUELING

Applicability:

Applies to operating limitations during refueling operations.

Objective:

To ensure that no incident could occur during refueling operations that would affect public health and safety.

Specifications : MODE 6 with the water level ≥ 23 ft above the top of the reactor vessel flange M.1

During refueling operations:

< See 3.9.4 >

1. The equipment hatch shall be closed and the personnel locks shall be capable of being closed. A temporary third door on the outside of the personnel lock shall be in place whenever both doors in a personnel lock are open (except for initial core loading).

2. Radiation levels in fuel handling areas, the containment and spent fuel storage pool shall be monitored continuously. < See 3.9.1 >

< See 3.9.3 >

3. Core subcritical neutron flux shall be continuously monitored by at least two neutron monitors, each with continuous visual indication in the control room and one with audible indication in the containment available whenever core geometry is being changed. When core geometry is not being changed, at least one neutron flux monitor shall be in service.

4. At least one residual heat removal loop shall be in operation. However, if refueling operations are affected by the residual heat removal loop flow, the operating residual heat removal loop may be removed from operation for up to one hour per eight hour period.

5. During reactor vessel head removal and while loading and unloading fuel from the reactor, a minimum boron concentration of 2100 ppm* shall be maintained in the primary coolant system.

< See 3.9.1 >

M.2 M.1 Replace with Insert 3.9.5-1, LCO 3.9.4

* This boron concentration value is in effect following U1R25 for Unit 1 and following U2R23 for Unit 2; and takes effect prior to loading fuel for those outages. Prior to U1R25, the Unit 1 boron concentration value of this specification is 1800 ppm. Prior to U2R23, the Unit 2 boron concentration value of this specification is 1800 ppm.

< See 3.9.1 >

< See 3.9.1 >

6. Direct communication between the control room and the operating floor of the containment shall be available whenever changes in core geometry are taking place.

< See 3.9.4 >

7. The Containment Purge and Vent System shall be operable. The Containment Purge and Vent System shall be demonstrated operable within 4 days prior to the start of and at least once per 7 days during refueling operations by verifying that Containment Purge and Vent isolation occurs on manual initiation and on high radiation test signal.

8. With the Containment Purge and Vent System inoperable, close the Purge and Vent containment penetrations.

< See 3.9.1 > < See 3.9.3 >

< See 3.9.4 > < See 3.9.1 >

9. If any of the specified limiting conditions in sections 1, 2, 3, 4, 5, and 6 are not met, refueling of the reactor shall cease. Work shall be initiated to correct the violated conditions so that the specified limits are met, and no operations which may increase the reactivity of the core shall be made.

Replace with Insert 3.9.5-2, Condition A — A.2

Insert 3.9.5-3, New SR 3.9.4.1 — M.3

Basis

< See 3.9.1 >

The equipment and general procedures to be utilized during refueling are discussed in the Final Safety Analysis Report. Detailed instructions, the above specified precautions, and the design of the fuel handling equipment incorporating built-in interlocks and safety features, provide assurance that no incident could occur during the refueling operations that would result in a hazard to public health and safety.[1]

Whenever changes are not being made in core geometry, one flux monitor is sufficient. This permits maintenance of the instrumentation. Continuous monitoring of radiation levels (2. above) and neutron flux provides immediate indication of an unsafe condition. The residual heat pump is used to maintain a uniform boron concentration.

The shutdown margin indicated in Part 5 will keep the core subcritical, even if all control rods were withdrawn from the core. During refueling, the reactor refueling cavity is filled with approximately 275,000 gallons

A.1

(c) Residual Heat Removal Loop (A)*

(d) Residual Heat Removal Loop (B)*

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

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

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

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

< See Section 3.4 >

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

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

b. Reactor Coolant Temperature Less Than 140°F

< See 3.9.6 >

(1) Both residual heat removal loops shall be operable except as permitted in items (3) or (4) below < See Section 3.4 >

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

Replace with Insert 3.9.5-2, Condition A

A.2

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

Replace with Insert 3.9.5-1, LCO 3.9.4

M.1

(4) One of the two residual heat removal loops may be temporarily out of service to meet surveillance requirements. < See Section 3.4 >

4. Pressurizer Safety Valves

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

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

---

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

Insert 3.9.5-1

LCO 3.9.4 One RHR loop shall be OPERABLE and in operation. M.1

M.2

NOTES

The required RHR loop may be not in operation for ≤ 1 hour per 8 hour period, provided no operations are permitted that would cause reduction of the Reactor Coolant System boron concentration.

APPLICABILITY: MODE 6 with the water level ≥ 23 ft above the top of reactor vessel flange. M.1

Insert 3.9.5-2

A. RHR loop requirements not met.	A.1	Suspend operations involving a reduction in reactor coolant boron concentration.	Immediately
	AND		
	A.2	Suspend loading irradiated fuel assemblies in the core.	Immediately
	AND		
	A.3	Initiate action to satisfy RHR loop requirements.	Immediately

A.2

Insert 3.9.5-3

SR 3.9.4.1	Verify one RHR loop is in operation.	12 hours

M.3

Justification For Deviations - NUREG-1431 Section 3.09.05

13-Nov-99

JFD Number	JFD Text
01	LCO 3.9.2 "Unborated Water Source Isolation Valves" was not adopted, based on the Point Beach design. Accordingly, the LCO and Bases have been renumbered, and the references to LCOs 3.9.6 and 3.9.7 within the Bases have been revised, to reflect this exclusion to Section 3.9 of the ITS. Additionally, the Bases for Required Action A.1 have been modified to reflect the unborated water sources to the RCS are not necessarily isolated. **ITS:** B 3.09.04 — **NUREG:** B 3.09.05 LCO 3.09.04 — LCO 3.09.05
02	The wording of LCO 3.9.4 Note was changed from "...may be removed from operation..." 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 RHR loop operation. To prevent this misunderstanding, the wording has been changed to, "...may be not in operation..." **ITS:** B 3.09.04 — **NUREG:** B 3.09.05 LCO 3.09.04 NOTE — LCO 3.09.05 NOTE
03	The closure of containment penetrations providing direct access from containment atmosphere to outside atmosphere has been excluded from Required Actions for not meeting RHR loop requirements. The closure of containment penetrations in this mode of operation is not included in the Point Beach Current Licensing Basis. Accordingly, the changes to this Required Action and the associated Bases identified in TSTF-197, Rev. 2 were not adopted. **ITS:** B 3.09.04 — **NUREG:** B 3.09.05 N/A — LCO 3.09.05 COND A RA A.4
04	The flow rate for the RHR loop in operation specified in SR 3.9.4.1 has been deleted. For Point Beach, the boron dilution accident is the only accident postulated to occur in MODE 6 which assumes the RHR system is in operation. The analysis only assumes there is some mixing of the borated coolant as a result of a RHR pump being in operation and does not specify a given flow rate. Therefore, there is no analytical basis for the inclusion of a flow rate in SR 3.9.4.1. **ITS:** B 3.09.04 — **NUREG:** B 3.09.05 SR 3.09.04.01 — SR 3.09.05.01

Justification For Deviations - NUREG-1431 Section 3.09.05

13-Nov-99

JFD Number	JFD Text
05	Reference to the General Design Criteria (GDC) of 10 CFR 50 Appendix A has been deleted from the Bases of the Technical Specifications, substituting reference to the appropriate section of the FSAR which specifies the Point Beach design criteria. Point Beach was constructed and licensed prior to the GDC being issued. The Point Beach construction permit was issued prior to the GDCs being issued in 1971. Point Beach was designed and constructed utilizing the 1967 proposed GDCs. Accordingly, reference has been provided to the appropriate criteria and section of the Point Beach FSAR which provides explanation of Point Beach's design basis. **ITS:** B 3.09.04 **NUREG:** B 3.09.05
06	The brackets have been removed and the proper plant specific information has been provided. **ITS:** B 3.09.04 **NUREG:** B 3.09.05
07	The Bases description of the RHR flowpath has been modified to reflect Point Beach design. At Point Beach, RHR flow is returned to the RCS via one RCS cold leg. **ITS:** B 3.09.04 **NUREG:** B 3.09.05

1 → 4

3.9 REFUELING OPERATIONS

3.9.5 Residual Heat Removal (RHR) and Coolant Circulation -High Water Level

1 → 4

LCO 3.9.5 One RHR loop shall be OPERABLE and in operation.

Approved TSTF 153

2 → be not in → not be in

---NOTES---

The required RHR loop may be removed from operation for ≤ 1 hour per 8 hour period, provided no operations are permitted that would cause reduction of the Reactor Coolant System boron concentration.

APPLICABILITY: MODE 6 with the water level ≥ 23 ft above the top of reactor vessel flange.

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. RHR loop requirements not met.	A.1	Suspend operations involving a reduction in reactor coolant boron concentration.	Immediately
	AND		
	A.2	Suspend loading irradiated fuel assemblies in the core.	Immediately
	AND		
	A.3	Initiate action to satisfy RHR loop requirements.	Immediately
	~~AND~~		
	3 →		~~(continued)~~

3 1 4

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. (continued)	A.4 Close all containment penetrations providing direct access from containment atmosphere to outside atmosphere.	4 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.9.5.1	Verify one RHR loop is in operation ~~and circulating reactor coolant at a flow rate of ≥ [2800] gpm.~~	12 hours

1 4 4

1 → 4

B 3.9 REFUELING OPERATIONS

B 3.9.5 Residual Heat Removal (RHR) and Coolant Circulation -High Water Level

1 → 4

BASES

BACKGROUND

The purpose of the RHR System in MODE 6 is to remove decay heat and sensible heat from the Reactor Coolant System (RCS) ~~, as required by GDC 34,~~ to provide mixing of borated coolant and to prevent boron stratification (Ref. 1). Heat is removed from the RCS by circulating reactor coolant through the RHR heat exchanger(s), where the heat is transferred to the Component Cooling Water System. The coolant is then returned to the RCS via the RCS cold leg ~~(s)~~. Operation of the RHR System for normal cooldown or decay heat removal is manually accomplished from the control room. The heat removal rate is adjusted by controlling the flow of reactor coolant through the RHR heat exchanger(s) and the bypass. Mixing of the reactor coolant is maintained by this continuous circulation of reactor coolant through the RHR System.

5 → 7 →

APPLICABLE SAFETY ANALYSES

If the reactor coolant temperature is not maintained below 200°F, boiling of the reactor coolant could result. This could lead to a loss of coolant in the reactor vessel. Additionally, boiling of the reactor coolant could lead to a reduction in boron concentration in the coolant due to boron plating out on components near the areas of the boiling activity. The loss of reactor coolant and the reduction of boron concentration in the reactor coolant would eventually challenge the integrity of the fuel cladding, which is a fission product barrier. One train of the RHR System is required to be operational in MODE 6, with the water level ≥ 23 ft above the top of the reactor vessel flange, to prevent this challenge. The LCO does permit de-energizing the RHR pump for short durations, under the condition that the boron concentration is not diluted. This conditional de-energizing of the RHR pump does not result in a challenge to the fission product barrier.

Although the RHR System does not meet a specific criterion of the NRC Policy Statement, it was identified in the NRC Policy Statement as an important contributor to risk reduction. Therefore, the RHR System is retained as a Specification.

1 → 4

1 → 4

LCO

Only one RHR loop is required for decay heat removal in MODE 6, with the water level ≥ 23 ft above the top of the reactor vessel flange. Only one RHR loop is required to be OPERABLE, because the volume of water above the reactor vessel flange provides backup decay heat removal capability. At least one RHR loop must be OPERABLE and in operation to provide:

a. Removal of decay heat;

b. Mixing of borated coolant to minimize the possibility of criticality; and

c. Indication of reactor coolant temperature.

An OPERABLE RHR loop includes an RHR pump, a heat exchanger, valves, piping, instruments, and controls to ensure an OPERABLE flow path and to determine the low end temperature. The flow path starts in one of the RCS hot legs and is returned to the RCS cold leg~~s~~.

7 → not be in operation ← Approved TSTF 153

The LCO is modified by a Note that allows the required operating RHR loop to be removed from service for up to 1 hour per 8 hour period, provided no operations are permitted that would cause a reduction of the RCS boron concentration. Boron concentration reduction is prohibited because uniform concentration distribution cannot be ensured without forced circulation. This permits operations such as core mapping or alterations in the vicinity of the reactor vessel hot leg nozzles and RCS to RHR isolation valve testing. During this 1 hour period, decay heat is removed by natural convection to the large mass of water in the refueling cavity.

APPLICABILITY

One RHR loop must be OPERABLE and in operation in MODE 6, with the water level ≥ 23 ft above the top of the reactor vessel flange, to provide decay heat removal. The 23 ft water level was selected because it corresponds to the 23 ft

1 → 4

APPLICABILITY (continued)

requirement established for fuel movement in LCO 3.9.7, "Refueling Cavity Water Level." Requirements for the RHR System in other MODES are covered by LCOs in Section 3.4, Reactor Coolant System (RCS), and Section 3.5, Emergency Core Cooling Systems (ECCS). RHR loop requirements in MODE 6 with the water level < 23 ft are located in LCO 3.9.6, "Residual Heat Removal (RHR) and Coolant Circulation -Low Water Level."

ACTIONS

RHR loop requirements are met by having one RHR loop OPERABLE and in operation, except as permitted in the Note to the LCO.

A.1

If RHR loop requirements are not met, there will be no forced circulation to provide mixing to establish uniform boron concentrations. ~~Reduced boron concentrations cannot occur by the addition of water with a lower boron concentration than that contained in the RCS because all of unborated water sources are isolated.~~

A.2

If RHR loop requirements are not met, actions shall be taken immediately to suspend loading of irradiated fuel assemblies in the core. With no forced circulation cooling, decay heat removal from the core occurs by natural convection to the heat sink provided by the water above the core. A minimum refueling water level of 23 ft above the reactor vessel flange provides an adequate available heat sink. Suspending any operation that would increase decay heat load, such as loading a fuel assembly, is a prudent action under this condition.

A.3

If RHR loop requirements are not met, actions shall be initiated and continued in order to satisfy RHR loop requirements. With the unit in MODE 6 and the refueling

1 4

ACTIONS (continued)

water level ≥ 23 ft above the top of the reactor vessel flange, corrective actions shall be initiated immediately.

3

~~A.4~~

~~If RHR loop requirements are not met, all containment penetrations providing direct access from the containment atmosphere to the outside atmosphere must be closed within 4 hours. With the RHR loop requirements not met, the potential exists for the coolant to boil and release radioactive gas to the containment atmosphere. Closing containment penetrations that are open to the outside atmosphere ensures dose limits are not exceeded.~~

~~The Completion Time of 4 hours is reasonable, based on the low probability of the coolant boiling in that time.~~

SURVEILLANCE REQUIREMENTS

SR 3.9.5.1 4 1

This Surveillance demonstrates that the RHR loop is in operation and circulating reactor coolant. The flow rate is determined by the flow rate necessary to provide sufficient decay heat removal capability and to prevent thermal and boron stratification in the core. The Frequency of 12 hours is sufficient, considering the flow, temperature, pump control, and alarm indications available to the operator in the control room for monitoring the RHR System.

REFERENCES

1. FSAR, Section [5.5.7]. 9.2 and 14.1.4 6

Verification includes flow rate, temperature, or pump status monitoring, which help ensure that forced flow is providing decay heat removal capability and mixing of the borated coolant to prevent thermal and boron stratification in the core.

4

1 4

No Significant Hazards Considerations - NUREG-1431 Section 3.09.05

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.

No Significant Hazards Considerations - NUREG-1431 Section 3.09.05

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.

3.9 REFUELING OPERATIONS

3.9.4 Residual Heat Removal (RHR) and Coolant Circulation-High Water Level

LCO 3.9.4 One RHR loop shall be OPERABLE and in operation.

----------NOTES----------
The required RHR loop may be not in operation for ≤ 1 hour per 8 hour period, provided no operations are permitte d that would cause reduction of the Reactor Coolant System boron concentration.

APPLICABILITY: MODE 6 with the water level ≥ 23 ft above the top of reactor vessel flange.

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. RHR loop requirements not met.	A.1	Suspend operations involving a reduction in reactor coolant boron concentration.	Immediately
	AND		
	A.2	Suspend loading irradiated fuel assemblies in the core.	Immediately
	AND		
	A.3	Initiate action to satisfy RHR loop requirements.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.9.4.1	Verify one RHR loop is in operation.	12 hours

B 3.9 REFUELING OPERATIONS

B 3.9.4 Residual Heat Removal (RHR) and Coolant Circulation -High Water Level

BASES

BACKGROUND

The purpose of the RHR System in MODE 6 is to remove decay heat and sensible heat from the Reactor Coolant System (RCS) to provide mixing of borated coolant and to prevent boron stratification (Ref. 1). Heat is removed from the RCS by circulating reactor coolant through the RHR heat exchanger(s), where the heat is transferred to the Component Cooling Water System. The coolant is then returned to the RCS via the RCS cold leg. Operation of the RHR System for normal cooldown or decay heat removal is manually accomplished from the control room. The heat removal rate is adjusted by controlling the flow of reactor coolant through the RHR heat exchanger(s) and the bypass. Mixing of the reactor coolant is maintained by this continuous circulation of reactor coolant through the RHR System.

APPLICABLE SAFETY ANALYSES

If the reactor coolant temperature is not maintained below 200°F, boiling of the reactor coolant could result. This could lead to a loss of coolant in the reactor vessel. Additionally, boiling of the reactor coolant could lead to a reduction in boron concentration in the coolant due to boron plating out on components near the areas of the boiling activity. The loss of reactor coolant and the reduction of boron concentration in the reactor coolant would eventually challenge the integrity of the fuel cladding, which is a fission product barrier. One train of the RHR System is required to be operational in MODE 6, with the water level ≥ 23 ft above the top of the reactor vessel flange, to prevent this challenge. The LCO does permit de-energizing the RHR pump for short durations, under the condition that the boron concentration is not diluted. This conditional de-energizing of the RHR pump does not result in a challenge to the fission product barrier.

Although the RHR System does not meet a specific criterion of the NRC Policy Statement, it was identified in the NRC Policy Statement as an important contributor to risk reduction. Therefore, the RHR System is retained as a Specification.

BASES

LCO

Only one RHR loop is required for decay heat removal in MODE 6, with the water level ≥ 23 ft above the top of the reactor vessel flange. Only one RHR loop is required to be OPERABLE, because the volume of water above the reactor vessel flange provides backup decay heat removal capability. At least one RHR loop must be OPERABLE and in operation to provide:

a. Removal of decay heat;

b. Mixing of borated coolant to minimize the possibility of criticality; and

c. Indication of reactor coolant temperature.

An OPERABLE RHR loop includes an RHR pump, a heat exchanger, valves, piping, instruments, and controls to ensure an OPERABLE flow path and to determine the low end temperature. The flow path starts in one of the RCS hot legs and is returned to the RCS cold leg.

The LCO is modified by a Note that allows the required operating RHR loop to not be in operation for up to 1 hour per 8 hour period, provided no operations are permitted that would cause a reduction of the RCS boron concentration. Boron concentration reduction is prohibited because uniform concentration distribution cannot be ensured without forced circulation. This permits operations such as core mapping or alterations in the vicinity of the reactor vessel hot leg nozzles and RCS to RHR isolation valve testing. During this 1 hour period, decay heat is removed by natural convectio n to the large mass of water in the refueling cavity.

APPLICABILITY

One RHR loop must be OPERABLE and in operation in MODE 6, with the water level ≥ 23 ft above the top of the reactor vessel flange, to provide decay heat removal. The 23 ft water level was selected because it corresponds to the 23 ft requirement established for fuel movement in LCO 3.9.6, "Refueling Cavity Water Level." Requirements for the RHR System in other MODES are covered by LCOs in Section 3.4, Reactor Coolant System (RCS), and Section 3.5, Emergency Core Cooling Systems (ECCS). RHR loop requirements in MODE 6 with the water level < 23 ft are located in LCO 3.9.5, "Residual Heat Removal (RHR) and C oolant Circulation - Low Water Level."