15.4.5 EMERGENCY CORE COOLING SYSTEM AND CONTAINMENT COOLING SYSTEM TESTS

A.6

< See Section 3.6 >

Applicability:

Applies to testing of the Emergency Core Cooling System and the Containment Cooling System.

Objective:

~~To verify that the subject systems will respond promptly and perform their design functions, if required.~~

A.7

Specification:

I. System Tests and Surveillances

M.3

18 months

A. Safety Injection System

SR 3.5.3.1
SR 3.5.2.3
SR 3.5.2.4

1. System tests shall be performed during reactor shutdowns for major fuel reloading. ~~The test shall be performed in accordance with the following procedure:~~

   a. ~~With the Reactor Coolant System pressure less than or equal to 350 psig and temperature less than or equal to 350°F, a test safety injection signal will be applied to initiate operation of the system. The motor breakers for the safety injection and residual heat removal pumps may be placed in the "test" position or racked in and operable for this test.~~

2. The test will be considered satisfactory ~~if control board indication and visual observations indicate~~ that all components have received the safety injection signal ~~in the proper sequence and timing.~~

LA.3

SR 3.5.3.1 -
SR 3.5.2.3
SR 3.5.2.4

That is, the appropriate pump motor breakers shall have opened and closed, and all valves shall have completed their travel.

B. Containment Spray System

1. System tests shall be performed during reactor shutdowns for major fuel reloading. The test shall be performed with the isolation valves in the spray supply lines at the containment blocked closed. Operation of the system is initiated by tripping the normal actuation instrumentation. The motor breakers for the pumps shall be placed in the "test" position for this test.

2. The test will be considered satisfactory if visual observations indicate all components have operated satisfactorily.

3. The spray nozzles shall be checked to verify that they are not obstructed at intervals not exceeding five years.

C. Containment Fan Coolers

1. Each fan cooler unit shall be tested at each refueling to verify proper operation of the backdraft dampers and the service water bypass valves.

2. Containment fan cooler accident fans shall be tested monthly to verify operability. Acceptable performance shall be that the accident fan starts and running current is verified.

< See Section 3.6 >

II. Component Tests and Surveillances

A. Pumps

1. The safety injection pumps, residual heat removal pumps, and

SR 3.5.3.1 -
SR 3.5.2.2

A.4

A.1

< See Section 3.6 >

A.1

SR 3.5.3.1 -
SR 3.5.2.2

containment spray pumps shall be tested in accordance with the Inservice Test Program.

2. Acceptable levels of performance shall be that the pumps start, reach their required developed head at~~, and operate for at least fifteen minutes on the full-flow test lines~~.

LA.2

B. Other

18 months

M.2

SR 3.5.3.1 -
SR 3.5.2.5

1. ~~At least every refueling~~, verify by visual inspection each containment sump suction inlet is not restricted by debris and the debris strainers show no evidence of structural distress or abnormal corrosion.

A.1

SR 3.5.3.1 -
SR 3.5.2.1

2. Verify each manual, power operated, and automatic valve necessary to insure system operability in the emergency core cooling and containment spray systems that is not locked, sealed, or otherwise secured in position, is in the correct position at least once every 31 days.

< See Section 3.6 >

A.3

Basis

The Safety Injection System and the Containment Spray System are principal plant Safety Systems that are normally inoperative during reactor operation. Complete systems tests cannot be performed when the reactor is operating because a safety injection signal causes containment isolation and a Containment Spray System test requires the system to be temporarily disabled. The method of assuring operability of these systems is therefore to combine systems tests to be performed during refueling shutdowns, with more frequent component tests, which can be performed during reactor operation.

The systems tests demonstrate proper automatic operation of the Safety Injection and Containment Spray Systems. With the pumps blocked from starting, a test signal is applied to initiate automatic action, and verification is made that the components receive the safety injection signal in the proper sequence. The test demonstrates the operation of the valves, pump circuit breakers, and automatic circuitry.[1]

< See Section 3.6 >

During reactor operation, the instrumentation which is depended on to initiate safety injection and containment spray is generally checked weekly and the initiating circuits are tested monthly (in accordance with Specification 15.4.1). In addition, the active components (pumps and valves) are to be tested in accordance with ASME Section XI requirements, to check the operation of the starting circuits and to verify that the pumps are in satisfactory running order. More frequent testing would not significantly increase the reliability (i.e. the probability that the component would operate when required), yet more frequent testing would result in increased wear over a long period of time.

Other systems that are also important to the emergency cooling function are the accumulators, the Component Cooling System, the Service Water System and the containment fan coolers. The accumulators are a passive safeguard. In accordance with Specification 15.4.1, the water volume and pressure in the accumulators are checked periodically. The other systems mentioned operate when the reactor is in operation and by these means are continuously monitored for satisfactory performance.

References

(1) FSAR Section 6.2.

Justification For Deviations - NUREG-1431 Section 3.05.03

13-Nov-99

JFD Number	JFD Text	
01	The brackets have been removed and the proper plant specific information has been provided. Condition A has been retained as the low head ECCS system is also utilized for residual heat removal.	
	ITS:	**NUREG:**
	LCO 3.05.03 COND B	LCO 3.05.03 COND B
	LCO 3.05.03 COND B RA B.1	LCO 3.05.03 COND B RA B.1
	LCO 3.05.03 COND C	LCO 3.05.03 COND C
02	NUREG 1431 under SR 3.5.3.1 provides a bracketed reference to SR 3.5.2.1 as being applicable in Mode 4. SR 3.5.2.1 requires position verification of ECCS valves, which if mispositioned would render more than one ECCS subsystem inoperable. This surveillance was not adopted in the Point Beach conversion as discussed in Justification for Deviation 19 of LCO 3.5.2 to this conversion package. Accordingly, this NUREG SR has not been adopted in LCO 3.5.3.	
	ITS:	**NUREG:**
	N/A	SR 3.05.03.01 - SR 3.05.02.01
03	NUREG 1431 contains a 31 day surveillance requirement which verifies that the ECCS piping is full of water. This surveillance is not contained in the CTS, and was not adopted, based on this surveillance being an unnecessary burden. The purpose of this SR is to ensure that the ECCS system piping is filled and vented. The ECCS piping at Point Beach is routed in such a manner as to preclude the need for periodic venting. All ECCS subsystem piping runs are routed below normal RWST level, thereby maintaining positive system pressure at all times. This pressure precludes inleakage through sources open to the atmosphere. Accordingly, this NUREG SR has not been adopted in LCO 3.5.3.	
	ITS:	**NUREG:**
	N/A	SR 3.05.03.01 - SR 3.05.02.03
04	NUREG 1431 SR 3.5.2.1, SR 3.5.2.3. and SR 3.5.2.7 were not adopted as part of Point Beach's conversion to the ITS as discussed in LCO 3.5.2. Accordingly, reference to NUREG SR 3.5.2.2, SR 3.5.2.4, and SR 3.5.2.8 have been renumbered so that the ITS references the appropriate SR.	
	ITS:	**NUREG:**
	SR 3.05.03.01 - SR 3.05.02.02	SR 3.05.03.01 - SR 3.05.02.04
		SR 3.05.03.01 - SR 3.05.02.04
	SR 3.05.03.01 - SR 3.05.02.05	SR 3.05.03.01 - SR 3.05.02.08

JFD Number	JFD Text	
05	Point Beach is a low head Safety Injection plant, which does not credit the operation of the Charging Pumps relative to an ECCS function. Only the Safety Injection and Residual Heat Removal Pumps are ECCS subsystems. Accordingly, the Bases for NUREG 1431 has been modified to reflect Point Beach's design.	
	ITS:	**NUREG:**
	B 3.05.03	B 3.05.03
06	The ECCS systems at Point Beach do not include hot leg recirculation as a phase of ECCS operation. The Point Beach design incorporates only an injection phase and a recirculation phase. The RHR subsystem normally supplies injection to the RCS via the upper plenum injection nozzles, and the SI subsystem supplies injection via the RCS cold legs. During the recirculation phase, the RHR subsystem will take suction from the containment sump, supplying direct injection into the RCS as well as providing suction supply to the SI subsystem. ECCS train operability to consist of an RHR pump system, an SI pump system, and the capability to support both the injection and recirculation phases. Changes have also been made where necessary in the Bases to address this issue. This change is necessary based on Point Beach's design and operation.	
	ITS:	**NUREG:**
	B 3.05.03	B 3.05.03
07	LCO 3.9.2 "Unborated Water Source Isolation Valves" was not adopted based on the Point Beach design. Accordingly, the references to LCO 3.9.5 and 6 have been revised to reflect the renumbering that has occurred in the 3.9 Section of the ITS.	
	ITS:	**NUREG:**
	B 3.05.03	B 3.05.03
08	The CTS requires each manual, power operated, and automatic valve necessary to insure system operability in the ECCS system which is not locked, sealed, or otherwise secured in position to be verified to be in its correct position every 31 days. This surveillance is applicable whenever ECCS is required to be operable. This surveillance is equivalent to SR 3.5.2.2 in NUREG 1431, and is required to be met in Mode 1, 2, and 3, but is not specified for performance in Mode 4. Based on the likelihood for valve mispositioning in the ECCS system not being significantly decreased in Mode 4, this surveillance requirement has been retained in the proposed ITS as SR 3.5.3.1-SR 3.5.2.1.	
	ITS:	**NUREG:**
	SR 3.05.03.01 - SR 3.05.02.01	N/A

JFD Number	JFD Text	
09	The CTS specifies that the ECCS components (pumps and valves) be tested to ensure that all components receive a safety injection signal during reactor shutdowns for refueling outages. This surveillance is required by the CTS anytime the system is required to be operable. The proposed ITS for Point Beach will retain this surveillance in Mode 4. The Safety Injection pumps are used exclusively for ECCS and ECCS support (e.g. accumulator fill operations) and accordingly are normally aligned for auto start. The RHR subsystem is a shared system, in that it is also utilized for shutdown cooling. The RHR system, when aligned for standby operations will be capable of auto starting in the ECCS configuration; however, during alignment to, and operation in, the shutdown cooling mode, this system must be manually realigned to perform its ECCS injection function. The note contained in the LCO Section of this LCO addresses these design and operational issues. This change to the NUREG is necessary to reflect the required automatic safety injection logic (manual actuation signal) as addressed in LCO 3.3.2 which is still required to be operable, thus requiring these SRs for continuity in operability requirements.	
	ITS:	**NUREG:**
	SR 3.05.03.01 - SR 3.05.02.03	N/A
		N/A
		N/A
	SR 3.05.03.01 - SR 3.05.02.04	N/A
		N/A
		N/A
10	NUREG 1431 under SR 3.5.3.1 provides a bracketed reference to SR 3.5.2.7 as being applicable in Mode 4. SR 3.5.2.7 requires position verification of ECCS throttle valves. This surveillance was not adopted in the Point Beach conversion as discussed in Justification for Deviation 20 of LCO 3.5.2 to this conversion package. Accordingly, this NUREG SR has not been adopted in LCO 3.5.3.	
	ITS:	**NUREG:**
	N/A	SR 3.05.03.01 - SR 3.05.02.07

3.5 EMERGENCY CORE COOLING SYSTEMS (ECCS)

3.5.3 ECCS – Shutdown

LCO 3.5.3 One ECCS train shall be OPERABLE.

Moved SR 3.5.3.1 Note to LCO IAW Approved TSTF 90

APPLICABILITY: MODE 4.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Required ECCS residual heat removal (RHR) subsystem inoperable.	A.1 Initiate action to restore required ECCS RHR subsystem to OPERABLE status.	Immediately
B. Required ECCS ~~[high head subsystem]~~ SI subsystem inoperable.	B.1 Restore required ECCS ~~[high head subsystem]~~ SI subsystem to OPERABLE status.	1 hour
C. Required Action and associated Completion Time [of Condition B] not met.	C.1 Be in MODE 5.	24 hours

Moved to LCO Section IAW Approved TSTF 90

SURVEILLANCE REQUIREMENTS

	SURVEILLANCE	FREQUENCY
SR 3.5.3.1	-------------------NOTE------------------- An RHR train may be considered OPERABLE during alignment and operation for decay heat removal, if capable of being manually realigned to the ECCS mode of operation. -- The following SRs are applicable for all equipment required to be OPERABLE: ~~[SR 3.5.2.1]~~ ~~[SR 3.5.2.7]~~ ~~[SR 3.5.2.3]~~ SR 3.5.2.~~8~~ SR 3.5.2.~~4~~	In accordance with applicable SRs

2 3 10 5 2 4/8 4

SR 3.5.2.1

Add SRs:
SR 3.5.2.3
SR 3.5.2.4

9

B 3.5 EMERGENCY CORE COOLING SYSTEMS (ECCS)

B 3.5.3 ECCS – Shutdown

BASES

BACKGROUND

The Background section for Bases 3.5.2, "ECCS – Operating," is applicable to these Bases, with the following modifications.

In MODE 4, the required ECCS train consists of two separate subsystems: ~~centrifugal charging~~ Safety Injection (SI) (high head) and residual heat removal (RHR) (low head).

5

Replace with Insert B 3.5.3-2

6

~~The ECCS flow paths consist of piping, valves, heat exchangers, and pumps such that water from the refueling water storage tank (RWST) can be injected into the Reactor Coolant System (RCS) following the accidents described in Bases 3.5.2.~~

APPLICABLE SAFETY ANALYSES

The Applicable Safety Analyses section of Bases 3.5.2 also applies to this Bases section.

Due to the stable conditions associated with operation in MODE 4 and the reduced probability of occurrence of a Design Basis Accident (DBA), the ECCS operational requirements are reduced. It is understood in these reductions that certain automatic safety injection (SI) actuation is not available. In this MODE, sufficient time exists for manual actuation of the required ECCS to mitigate the consequences of a DBA.

Only one train of ECCS is required for MODE 4. This requirement dictates that single failures are not considered during this MODE of operation. The ECCS trains satisfy Criterion 3 of the NRC Policy Statement.

LCO

In MODE 4, one of the two independent (and redund ant) ECCS trains is required to be OPERABLE to ensure that sufficient ECCS flow is available to the core following a DBA.

BASES

LCO (continued)

5 an SI

In MODE 4, an ECCS train consists of ~~a centrifugal charging~~ subsystem and an RHR subsystem. Each train includes the piping, instruments, and controls to ensure an OPERABLE flow path capable of taking suction from the RWST and transferring suction to the containment sump.

6

During an event requiring ECCS actuation, a flow path is required to provide an abundant supply of water from the RWST to the RCS via the ECCS pumps and their respective supply headers ~~to each of the four cold leg injection nozzles~~. In the long term, this flow path may be switched to take its supply from the containment sump ~~and to deliver its flow to the RCS hot and cold legs~~.

Insert B 3.5.3-1 Approved TSTF-90

APPLICABILITY

In MODES 1, 2, and 3, the OPERABILITY requirements for ECCS are covered by LCO 3.5.2.

In MODE 4 with RCS temperature below 350°F, one OPERABLE ECCS train is acceptable without single failure consideration, on the basis of the stable reactivity of the reactor and the limited core cooling requirements.

In MODES 5 and 6, plant conditions are such that the probability of an event requiring ECCS injection is extremely low. Core cooling requirements in MODE 5 are addressed by LCO 3.4.7, "RCS Loops – MODE 5, Loops Filled," and LCO 3.4.8, "RCS Loops – MODE 5, Loops Not Filled." MODE 6 core cooling requirements are addressed by LCO 3.9.~~5~~ 4 "Residual Heat Removal (RHR) and Coolant Circulation – High Water Level," and LCO 3.9.~~6~~ 5 "Residual Heat Removal (RHR) and Coolant Circulation – Low Water Level." 7

ACTIONS

A.1

With no ECCS RHR subsystem OPERABLE, the plant is not prepared to respond to a loss of coolant accident or to continue a cooldown using the RHR pumps and heat exchangers. The Completion Time of immediately to initiate actions that would restore at least one ECCS RHR subsystem to OPERABLE

BASES

ACTIONS (continued)

status ensures that prompt action is taken to restore the required cooling capacity. Normally, in MODE 4, reactor decay heat is removed from the RCS by an RHR loop. If no RHR loop is OPERABLE for this function, reactor decay heat must be removed by some alternate method, such as use of the steam generators. The alternate means of heat removal must continue until the inoperable RHR loop components can be restored to operation so that decay heat removal is continuous.

With both RHR pumps and heat exchangers inoperable, it would be unwise to require the plant to go to MODE 5, where the only available heat removal system is the RHR. Therefore, the appropriate action is to initiate measures to restore one ECCS RHR subsystem and to continue the actions until the subsystem is restored to OPERABLE status.

B.1

or containment sump via the RHR subsystem

5/6

SI

With no ~~ECCS high head~~ subsystem OPERABLE, due to the inoperability of the ~~centrifugal charging~~ pump or flow path from the RWST, the plant is not prepared to provide high pressure response to Design Basis Events requiring SI. The 1 hour Completion Time to restore at least one ~~ECCS high head~~ subsystem to OPERABLE status ensures that prompt action is taken to provide the required cooling capacity or to initiate actions to place the plant in MODE 5, where an ECCS train is not required.

C.1

When the Required Actions of Condition B cannot be completed within the required Completion Time, a controlled shutdown should be initiated. Twenty-four hours is a reasonable time, based on operating experience, to reach MODE 5 in an orderly manner and without challenging plant systems or operators.

BASES

SURVEILLANCE REQUIREMENTS

Approved TSTF 90

SR 3.5.3.1

The applicable Surveillance descriptions from Bases 3.5.2 apply. ~~This SR is modified by a Note that allows an RHR train to be considered OPERABLE during alignment and operation for decay heat removal, if capable of being manually realigned (remote or local) to the ECCS mode of operation and not otherwise inoperable. This allows operation in the RHR mode during MODE 4, if necessary.~~

REFERENCES

The applicable references from Bases 3.5.2 apply.

BASES INSERTS

INSERT B 3.5.3-1:

This LCO is modifed by a Note that allows an RHR train to be considered OPERABLE during alignment and operation for decay heat removal, if capable of being manually realigned (remote or local) to the ECCS mode of operation and not otherwise inoperable. This allows operation in the RHR mode during MODE 4.

INSERT B 3.5.3-2:

The ECCS flow paths consist of piping, valves, heat exchangers, and pumps necessary to provide water from the RWST into the RCS during the injection phase and from the containment sump into the RCS during the recirculation phase following the accidents described in Bases 3.5.2.

No Significant Hazards Considerations - NUREG-1431 Section 3.05.03

13-Nov-99

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

NSHC Number	NSHC Text
L.01	In accordance with the criteria set forth in 10 CFR 50.92, PBNP has evaluated this proposed Technical Specifications change and determined it does not represent a significant hazards consideration. The following is provided in support of this conclusion. 1. Does the change involve a significant increase in the probability or consequences of an accident previously evaluated? This change does not result in any hardware changes. The components covered by this Technical Specification are not assumed to be initiators of any analyzed event. The Shutdown Actions, Mode of Applicabilities, and required number of ECCS components are not precursors to any analyzed events. Therefore the probability associated with analyzed events is unchanged. The proposed Applicabilities, minimum equipment requirements and shutdown actions are based on stable unit conditions associated with MODE 4, the reduced thermal energy in the core, sufficient time for manual actuation of the remaining ECCS pumps to mitigate a Design Basis Accidents as necessary, and the assumption that single failures in the ECCS system are not assumed below Mode 3. As such, there is no significant increase in the consequence 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 establishes Applicabilities, Required Actions, and complements of components reflective of assumptions made in the Accident Analysis. As such, this change does not create the possibility of a new or different kind of accident from any accident previously evaluated. 3. Does this change involve a significant reduction in a margin of safety? This change proposes equipment Applicabilities, minimum numbers of components/subsystems and shutdown actions based on stable unit conditions associated with MODE 4, the reduced thermal energy in the core, and sufficient time for manual actuation of the remaining ECCS pumps, assuming no single failure within the ECCS subsystem. In addition, this change is reflective of assumptions made in the accident analysis for Point Beach. Therefore, this change does not involve a significant reduction in a margin of safety.

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

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

3.5 EMERGENCY CORE COOLING SYSTEMS (ECCS)

3.5.3 ECCS – Shutdown

LCO 3.5.3 One ECCS train shall be OPERABLE.

-------------------------NOTE-------------------------
An RHR train may be considered OPERABLE during alignment and operation for decay heat removal, if capable of being manually realigned to the ECCS mode of operation.

APPLICABILITY: MODE 4.

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. Required ECCS residual heat removal (RHR) subsystem inoperable.	A.1	Initiate action to restore required ECCS RHR subsystem to OPERABLE status.	Immediately
B. Required ECCS SI subsystem inoperable.	B.1	Restore required ECCS SI subsystem to OPERABLE status.	1 hour
C. Required Action and associated Completion Time of Condition B not met.	C.1	Be in MODE 5.	24 hours

SURVEILLANCE REQUIREMENTS

	SURVEILLANCE	FREQUENCY
SR 3.5.3.1	The following SRs are applicable for all equipment required to be OPERABLE: SR 3.5.2.1 SR 3.5.2.4 SR 3.5.2.2 SR 3.5.2.5 SR 3.5.2.3	In accordance with applicable SRs

B 3.5 EMERGENCY CORE COOLING SYSTEMS (ECCS)

B 3.5.3 ECCS – Shutdown

BASES

BACKGROUND

The Background section for Bases 3.5.2, "ECCS – Operating," is applicable to these Bases, with the following modifications.

In MODE 4, the required ECCS train consists of two separate subsystems: Safety Injection (SI) (high head) and residual heat removal (RHR) (low head).

The ECCS flow paths consist of piping, valves, heat exchangers, and pumps necessary to provide water from the RWST into the RCS during the injection phase and from the containment sump into the RCS during the recirculation phase following the accidents described in Bases 3.5.2.

APPLICABLE SAFETY ANALYSES

The Applicable Safety Analyses section of Bases 3.5.2 also applies to this Bases section.

Due to the stable conditions associated with operation in MODE 4 and the reduced probability of occurrence of a Design Basis Accident (DBA), the ECCS operational requirements are reduced. It is understood in these reductions that certain automatic safety injection (SI) actuation is not available. In this MODE, sufficient time exists for manual actuation of the required ECCS to mitigate the consequences of a DBA.

Only one train of ECCS is required for MODE 4. This requirement dictates that single failures are not considered during this MODE of operation. The ECCS trains satisfy Criterion 3 of the NRC Policy Statement.

LCO

In MODE 4, one of the two independent (and redundant) ECCS trains is required to be OPERABLE to ensure that sufficient ECCS flow is available to the core following a DBA.

BASES

LCO (continued)

In MODE 4, an ECCS train consists of an SI subsystem and an RHR subsystem. Each train includes the piping, instruments, and controls to ensure an OPERABLE flow path capable of taking suction from the RWST and transferring suction to the containment sump.

During an event requiring ECCS actuation, a flow path is required to provide an abundant supply of water from the RWST to the RCS via the ECCS pumps and their respective supply headers. In the long term, this flow path may be switched to take its supply from the containment sump.

This LCO is modifed by a Note that allows an RHR train to be considered OPERABLE during alignment and operation for decay heat removal, if capable of being manually realigned (remote or local) to the ECCS mode of operation and not otherwise inoperable. This allows operation in the RHR mode during MODE 4.

APPLICABILITY

In MODES 1, 2, and 3, the OPERABILITY requirements for ECCS are covered by LCO 3.5.2.

In MODE 4 with RCS temperature below 350°F, one OPERABLE ECCS train is acceptable without single failure consideration, on the basis of the stable reactivity of the reactor and the limited core cooling requirements.

In MODES 5 and 6, plant conditions are such that the probability of an event requiring ECCS injection is extremely low. Core cooling requirements in MODE 5 are addressed by LCO 3.4.7, "RCS Loops – MODE 5, Loops Filled," and LCO 3.4.8, "RCS Loops – MODE 5, Loops Not Filled." MODE 6 core cooling requirements are addressed by 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."

ACTIONS

A.1

With no ECCS RHR subsystem OPERABLE, the plant is not prepared to respond to a loss of coolant accident or to continue a cooldown using the RHR pumps and heat exchangers. The Completion Time of immediately to initiate actions that would restore at least one ECCS RHR subsystem to OPERABLE status ensures that prompt action is taken to restore the required cooling capacity. Normally, in MODE 4, reactor decay heat is removed from the RCS by an RHR loop. If no RHR loop is OPERABLE for this function, reactor decay heat must be removed by some alternate method, such as use of the steam generators. The alternate means of heat removal must continue until the inoperable RHR loop components can be restored to operation so that decay heat removal is continuous.

With both RHR pumps and heat exchangers inoperable, it would be unwise to require the plant to go to MODE 5, where the only available heat removal system is the RHR. Therefore, the appropriate action is to initiate measures to restore one ECCS RHR subsystem and to continue the actions until the subsystem is restored to OPERABLE status.

B.1

With no SI subsystem OPERABLE, due to the inoperability of the SI pump or flow path from the RWST or containment sump via the RHR subsystem, the plant is not prepared to provide high pressure response to Design Basis Events requiring SI. The 1 hour Completion Time to restore at least one SI subsystem to OPERABLE status ensures that prompt action is taken to provide the required cooling capacity or to initiate actions to place the plant in MODE 5, where an ECCS train is not required.

C.1

When the Required Actions of Condition B cannot be completed within the required Completion Time, a controlled shutdown should be initiated. Twenty-four hours is a reasonable time, based on operating experience, to reach MODE 5 in an orderly manner and without challenging plant systems or operators.

BASES

SURVEILLANCE REQUIREMENTS	<u>SR 3.5.3.1</u> The applicable Surveillance descriptions from Bases 3.5.2 apply.
REFERENCES	The applicable references from Bases 3.5.2 apply.

Cross-Reference Report - NUREG-1431 Section 3.05.04

ITS to CTS

13-Nov-99

ITS	CTS	DOC
B 3.05.04	BASES	A.03
LCO 3.05.04	15.03.03 APPL	A.01
	15.03.03 APPL	A.06
	15.03.03 OBJ	A.07
	15.03.03.A.01	A.05
LCO 3.05.04 COND A	NEW	L.01
LCO 3.05.04 COND A RA A.1	NEW	L.01
LCO 3.05.04 COND B	NEW	A.08
LCO 3.05.04 COND B RA B.1	NEW	A.08
LCO 3.05.04 COND C	NEW	A.08
LCO 3.05.04 COND C RA C.1	NEW	A.08
LCO 3.05.04 COND C RA C.2	NEW	A.08
SR 3.05.04.01	NEW	M.04
SR 3.05.04.01 NOTE	N/A	M.04
SR 3.05.04.02	15.03.03.A.01.A	M.02
SR 3.05.04.03	15.03.03.A.01.A	M.03
	15.04.01 T 15.04.01-02 03	A.04
	15.04.01 T 15.04.01-02 03 (6)	A.04

Cross-Reference Report - NUREG-1431 Section 3.05.04

CTS to ITS

13-Nov-99

CTS	ITS	DOC
15.03.03 APPL	LCO 3.05.04	A.06
	LCO 3.05.04	A.01
15.03.03 OBJ	LCO 3.05.04	A.07
15.03.03.A.01	DELETED	M.01
	LCO 3.05.04	A.05
15.03.03.A.01.A	SR 3.05.04.02	M.02
	SR 3.05.04.03	M.03
15.03.03.A.01.A NOTE *	DELETED	A.02
15.04.01 T 15.04.01-02 03	SR 3.05.04.03	A.04
15.04.01 T 15.04.01-02 03 (6)	SR 3.05.04.03	A.04
BASES	B 3.05.04	A.03

Description of Changes - NUREG-1431 Section 3.05.04

13-Nov-99

DOC Number	DOC Text	
A.01	In the conversion of Point Beach current Technical Specifications (CTS) to the proposed plant specific Improved Technical Specifications (ITS), certain wording preferences or conventions are adopted which do not result in technical changes (either actual or interpretational). Editorial changes, reformatting, and revised numbering are adopted to make the ITS consistent with the Standard Technical Specifications, Westinghouse Plants, NUREG-1431, Revision 1 (i.e., Improved Standard Technical Specifications (ISTS)).	
	CTS:	**ITS:**
	15.03.03 APPL	LCO 3.05.04
A.02	The CTS for 3.15.3.3.A.1.a contains a footnotes which dictate required RWST boron concentration based on whether the unit is operating pre or post refueling outage U1R25 and U2R23. The value proposed for inclusion into the Point Beach proposed ITS is the post U1R25 and U2R23 values. This change is administrative as both units will be operating under the limits proposed for inclusion into the ITS (post U1R25/U2R23) prior to issuance of the ITS. Accordingly, deletion of the pre U1R25 and U2R23 limitations is acceptable and administrative, as these values no longer impose any operational limitations.	
	CTS:	**ITS:**
	15.03.03.A.01.A NOTE *	DELETED
A.03	The Bases of the current Technical Specifications for this LCO have been completely replaced by the revised Bases reflecting the format and applicable content of the Improved Technical Specifications for Point Beach. The proposed Bases are based on NUREG 1431 Rev. 1. The proposed Bases for this LCO are consistent and supportive of the proposed LCO, and accordingly is administrative.	
	CTS:	**ITS:**
	BASES	B 3.05.04
A.04	CTS Table 15.4.1-2 item number 3 requires the RWST to be sampled for boron concentration weekly except during refueling shutdowns (Modes 1, 2, 3, 4, and 5) . The proposed Point Beach Improved Technical Specification (SR 3.5.4.3) will require RWST boron sampling on every 7 days in Modes 1, 2, 3, and 4. The requirement to sample RWST boron concentration in Cold Shutdown (Mode 5) has been deleted from the Technical Specifications, as this requirement is associated with maintaining operable boric acid flowpath sources, which has been relocated to licensee control through application of the Technical Specification selection criteria contained in 10CFR 50.36. Relocation of this information is addressed in LCO 3.5.2 of the Point Beach conversion package. As such, this change is administrative.	
	CTS:	**ITS:**
	15.04.01 T 15.04.01-02 03	SR 3.05.04.03
	15.04.01 T 15.04.01-02 03 (6)	SR 3.05.04.03

DOC Number	DOC Text	
A.05	The CTS 15.3.3.A.1.a states that the RWST is required to be operable prior to the reactor being made critical. However, the CTS does not contain any explicit Actions for an inoperable RWST, which would required Specification 15.3.0.b to be invoked whenever the RWST becomes inoperable. Specification 15.3.0.b will require the unit to be placed into Hot Shutdown (ITS Mode 3) within 7 hours and Cold Shutdown (ITS Mode 5) within 37 hours, implying an Applicability of Modes 1, 2, 3, and 4 (ITS Modes). Proposed LCO 3.5.4 will require the RWST to be operable in Modes 1, 2, 3, and 4. As such, this change is considered administrative as it is clarifying an ambiguous LCO Applicability and Action.	
	CTS:	**ITS:**
	15.03.03.A.01	LCO 3.05.04
A.06	The CTS provides an introductory statement (Applicability) which simply states which systems/components are addressed within a given section. This same information while worded differently is contained within the title of each ITS LCO. Accordingly, this change is a change in format with no change in technical requirement.	
	CTS:	**ITS:**
	15.03.03 APPL	LCO 3.05.04
A.07	The CTS provides an introductory statement (Objective) at the beginning of this Section of the Technical Specifications which provide a brief summary of the purpose for this Section. This information is contained in the Bases Section of the ITS. This information does not establish any regulatory requirements for the systems and components addressed within this Section. Accordingly, deletion of this information does not alter any requirement set forth in the Technical Specifications. This change is administrative and consistent with the format and presentation for the ITS as provided in NUREG 1431.	
	CTS:	**ITS:**
	15.03.03 OBJ	LCO 3.05.04

DOC Number	DOC Text
A.08	The CTS does not specify any remedial action for an inoperable RWST. The CTS contains Specification 15.3.0.b which is required to be entered in the event that an LCO cannot be satisfied because of failures or limitations beyond those specified in the permissible conditions of the LCO. Accordingly, CTS 15.3.0.B must be entered if the RWST becomes inoperable, which requires the unit to be placed into hot shutdown (ITS Mode 3) within 7 hours and cold shutdown (ITS Mode 5) within 37 hours. Proposed ITS LCO 3.5.4 Condition B provides a Condition for the RWST being inoperable for reasons other than boron concentration or temperature being outside of limits, allowing 1 hour to correct the condition, before requiring entry into Condition C which requires the unit to be placed into Mode 3 within 6 hours and Mode 5 within 36 hours (total of 7 hours to Mode 3 and 37 hours to Mode 5). Inclusion of these Conditions are administrative in that the Actions and associated time frame of the CTS and the ITS are the same. **CTS:** NEW **ITS:** LCO 3.05.04 COND B LCO 3.05.04 COND B RA B.1 LCO 3.05.04 COND C LCO 3.05.04 COND C RA C.1 LCO 3.05.04 COND C RA C.2
L.01	The CTS does not contain any remedial actions for RWST temperature or boron concentration out of limits. The CTS contains an Action (15.3.0.B) which is required to be entered in the event that an LCO cannot be satisfied because of failures or limitations beyond those specified in the permissible conditions of the LCO. Accordingly, CTS 15.3.0.B must be entered which requires the unit to be placed into hot shutdown (ITS Mode 3) within 7 hours and cold shutdown (ITS Mode 5) within 37 hours. Proposed ITS LCO 3.5.4 Condition A allows 8 hours to restore either RWST boron concentration or temperature to within limits before requiring the unit to be shutdown (Mode 3 in 6 hours and Mode 5 in 36 hours). An 8-hour Completion Time to restore RWST boron concentration or temperature to within limits is justified considering the contents of the tank are still available for injection following a Design Basis Accident and this time frame provides a reasonable amount of time to return the RWST to OPERABLE status. **CTS:** NEW **ITS:** LCO 3.05.04 COND A LCO 3.05.04 COND A RA A.1

13-Nov-99

DOC Number	DOC Text	
M.01	The CTS contains a provision exempting the requirement to maintain the RWST operable during low power physics testing. This provision has been deleted in the proposed Technical Specifications. Low power physics testing in the Improved Technical Specifications is a subset of Mode 2. While Mode 2 is typically a non limiting Mode, the operability requirements of the RWST are independent of physics testing, accordingly this provision has been deleted. This change represent a more restrictive changes as it involves the deletion of a flexibility that currently exists.	
	CTS:	**ITS:**
	15.03.03.A.01	DELETED
M.02	CTS 15.3.3.A.1.a specifies a minimum level requirement for the RWST, however, no periodic surveillance exists to verify this limit is met. Accordingly, a 7 day verification of RWST level is being proposed for the Point Beach ITS. The RWST volume is normally stable parameter, a 7 day Frequency is appropriate and has been shown to be acceptable through industry operating experience. This change is more restrictive that the CTS requirements and appropriate to verify LCO compliance.	
	CTS:	**ITS:**
	15.03.03.A.01.A	SR 3.05.04.02
M.03	The CTS only contains a lower RWST boron concentration limit. An upper limit has been proposed for inclusion into the periodic boron verification surveillance. The upper limit assures that the resulting containment sump pH following a LOCA will be maintained in an acceptable range so that the effects of chloride and caustic stress corrosion on mechanical systems and components will be minimized. Since the RWST volume is normally stable, a 7 day sampling Frequency to verify boron concentration is appropriate and has been shown to be acceptable through industry operating experience.	
	CTS:	**ITS:**
	15.03.03.A.01.A	SR 3.05.04.03
M.04	The CTS does not specify any RWST temperature limitations or periodic surveillances for RWST temperature. Accordingly, a periodic surveillance has been proposed for inclusion into the Point Beach ITS (SR 3.5.4.2) which requires verification of RWST water temperature every 24 hours. This surveillance is proposed for inclusion into the Point Beach ITS to preserve the assumption made in various accident analyses. The proposed Frequency is sufficient to identify a temperature change that would approach either the upper or lower limit. This change is more restrictive that the CTS and appropriate to verify LCO compliance.	
	CTS:	**ITS:**
	N/A	SR 3.05.04.01 NOTE
	NEW	SR 3.05.04.01

A.1

15.3.3 EMERGENCY CORE COOLING SYSTEM, AUXILIARY COOLING SYSTEMS, AIR RECIRCULATION FAN COOLERS, AND CONTAINMENT SPRAY

A.6

< See Sections 3.6 and 3.7 >

Applicablity:

Applies to the operating status of the Emergency Core Cooling System, Auxiliary Cooling Systems, Air Recirculation Fan Coolers, and Containment Spray.

A.7

Objective:

To define those limiting conditions for operation that are necessary: ~~(1) to remove~~ decay heat from the ~~core in~~ emergency or normal shutdown situations, (2) to remove heat from containment in normal operating and emergency situations, and (3) to remove airborne iodine from the containment atmosphere following a postulated Design Basis Accident.

Specification:

LCO 3.5.4

M.1

A.5

A. Safety Injection and Residual Heat Removal Systems

M.2

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

SR 3.5.4.2

a. The refueling water tank contains not less than 275,000 gal. of water with a

SR 3.5.4.2

boron concentration of at least 2700 ppm*

A.2

M.3

< See LCO 3.5.1 >

b. Each accumulator is pressurized to at least 700 psig and contains at least 1100 ft^3 but no more than 1136 ft^3 of water with a boron concentration of at least 2600 ppm.** Neither accumulator may be isolated.

< See LCOs 3.5.2 and 3.5.3 >

c. Two safety injection pumps are operable.

d. Two residual heat removal pumps are operable.

e. Two residual heat exchangers are operable.

A.2

*This value is in effect following U1R25 for Unit 1 and U2R23 for Unit 2; and takes effect prior to leaving the cold shutdown condition of those outages. Prior to U1R25, the Unit 1 minimum RWST boron concentration is 2000 ppm. Prior to U2R23, the Unit 2 minimum RWST boron concentration is 2000 ppm.

**This value is in effect following U1R25 for Unit 1 and U2R23 for Unit 2; and takes effect prior to leaving the cold shutdown condition of those outages. Prior to U1R25, the Unit 1 minimum SI accumulator boron concentration is 2000 ppm. Prior to U2R23, the Unit 2 minimum SI accumulator boron concentration is 2000 ppm.

< See LCO 3.5.1 >

< See Section 3.7 >

2. During power operation, the requirements of 15.3.3.D-1 may be modified to allow the following conditions. If the system is not restored to meet the conditions of 15.3.3.D-1 within the time period specified, the affected reactor(s) will be placed in the hot shutdown condition within six hours and in cold shutdown within 36 hours.

   a. One of the six required service water pumps may be out of service provided a pump is restored to operable status within 7 days. A second service water pump may be out of service provided a pump is restored to operable status within 72 hours. A third service water pump may be out of service provided two pumps are restored to operable status within 72 hours.

   b. The service water ring header continuous flowpath may be out of service for a period of 7 days. If less than four service water pumps are operable, service water system flow shall be evaluated within 24 hours of less than four service water pumps being operable. If it is determined that any equipment will not receive sufficient flow, the applicable LCOs for the affected equipment shall be entered. The LCOs can be exited if system realignment is completed to achieve the required flow rates for the affected equipment.

   c. Any or all automatic isolation valves required during accident conditions may be out of service for up to 72 hours provided at least four service water pumps are operable. This LCO can be exited provided the lines are isolated with a seismically qualified isolation valve or the valves are restored to operable status.

   d. The containment fan cooler outlet motor operated valves may be open for up to 72 hours provided at least five service water pumps are operable. This LCO can be exited provided the valves are returned to the closed position or the flowpath is isolated.

Basis

The normal procedure for starting the reactor is, first, to heat the reactor coolant to near operating temperature, by running the reactor coolant pumps. The reactor is then made critical by withdrawing control rods and/or diluting boron in the coolant.[1] With this mode of start-up, the energy stored in the reactor coolant during the approach to criticality is substantially equal to that during power operation and therefore to be conservative most engineered safety system components and auxiliary cooling systems, shall be fully operable. During low temperature physics tests there is a negligible amount of stored energy in the reactor coolant, therefore an accident comparable in severity to the Design Basis Accident is not possible, and the engineered safety systems are not required.

A.3

A.3

The operable status of the various systems and components is to be demonstrated by periodic tests, defined by Specification 15.4.5. A large fraction of these tests will be performed while the reactor is operating in the power range. If a component is found to be inoperable it will be possible in most cases to effect repairs and restore the system to full operability within a relatively short time. For a single component to be inoperable does not negate the ability of the system to perform its function, but it reduces the redundancy provided in the reactor design and thereby limits the ability to tolerate additional equipment failures. If it develops that (a) the inoperable component is not repaired within the specified allowable time period or (b) a second component in the same or related system is found to be inoperable, the reactor will initially be put in the hot shutdown condition to provide for reduction of the decay heat from the fuel, and consequent reduction of cooling requirements after a postulated loss-of-coolant accident. This will also permit improved access for repairs in some cases. After a limited time in hot shutdown, if the malfunction(s) are not corrected, the reactor will be placed in the cold shutdown condition, utilizing normal shutdown and cooldown procedures. For example, specification 15.3.3.A.2.a allows one accumulator to be isolated or otherwise inoperable for periods of up to one hour. An inoperable accumulator may be defined as one with its outlet MOV shut, no pressure instrumentation operable, or < See LCO 3.5.1 >s cross-connected with the accumulator on the other loop. If the inoperable accumulator is not restored witin one hour then the conditions of section 15.3.0 apply which requires the affected unit, if critical, to be in hot shutdown within seven hours and in cold shutdown within 37 hours if the condition is not corrected. In the cold shutdown condition there is no possibility of an accident that would release fission products or damage the fuel elements.

The specified repair times do not apply to regularly scheduled maintenance of the engineered safety systems, which is normally to be performed during refueling shutdowns. The limiting times to repair are based on:

1) Assuring with high reliability that the safety system will function properly if required to do so.

2) Allowances of sufficient time to effect repairs using safe and proper procedures.

A.3

A.3

Assuming the reactor has been operating at full rated power for at least 100 days, the magnitude of the decay heat decreases as follows after initiating hot shutdown.*

Time After Shutdown	Decay Heat % of Rated Power
1 min.	3.6
30 min.	1.55
1 hour	1.25
8 hours	0.7
48 hours	0.4

*Based on ANS 5.1-1979, "Decay Heat Power in Light-Water Reactors"

Thus, the requirement for core cooling in case of a postulated loss-of-coolant accident while in the hot shutdown condition is significantly reduced below the requirements for a postulated loss-of-coolant accident during power operation. Putting the reactor in the hot shutdown condition significantly reduces the potential consequences of a loss-of-coolant accident, and also allows more free access to some of the engineered safety system components in order to effect repairs.

When the failures involve the residual heat removal system, in order to insure redundant means of decay heat removal, the reactor system may remain in a condition with reactor coolant temperatures greater than 350°F so that the reactor coolant loops and associated steam generators may be utilized for redundant decay heat removal. However, when the remaining RHR loop must be relied upon for redundant decay heat removal capability, reactor coolant temperatures shall be maintained between 350°F and 140°F. < See Section 3.5.2 >

With respect to the core cooling function, there is some functional redundancy for certain ranges of break sizes.(2) < See LCOs 3.5.1, 3.5.2, and 3.5.3 >

The operability of the Refueling Water Storage Tank (RWST) as part of the ECCS ensures that a sufficient supply of borated water is available for injection by the ECCS in the event of either a LOCA or a steamline break. The limits on RWST

A.3

A.3

minimum volume and boron concentration ensure that: (1) sufficient water is available within containment to permit recirculation cooling flow to the core; (2) the reactor will remain subcritical in the cold condition (68 to 212 degrees-F) following a small break LOCA assuming complete mixing of the RWST, RCS, spray additive tank, containment spray system piping and ECCS water volumes with all control rods inserted except the most reactive control rod assembly (ARI-1); (3) the reactor will remain subcritical in the cold condition following a large break LOCA (break flow area greater than 3 ft^2) assuming complete mixing of the RWST, RCS, ECCS water and other sources of water that may eventually reside in the sump post-LOCA with all control rods assumed to be out (ARO); and (4) long term subcriticality is maintained following a steamline break assuming ARI-1 and fuel failure is precluded.

The containment cooling function is provided by two independent systems: (a) fan coolers and (b) containment spray which, with sodium hydroxide addition, provides the iodine removal function. During normal power operation, only three of the four fan coolers are required to remove heat lost from equipment and piping within the containment.[3] In the event of a Design Basis Accident, any one of the following combinations will provide sufficient cooling to reduce containment pressure: (1) four fan coolers, (2) two containment spray pumps, (3) two fan coolers plus one containment spray pump.[4] Sodium hydroxide addition via one spray pump reduces airborne iodine activity sufficiently to limit off-site doses to acceptable values. One or two fan coolers is permitted to be inoperable for up to 72 hours during power operation.

< See Section 3.6 >

Specification 15.3.3.B.2.c requires valves that provide the duplicate function be operable prior to initiating repairs on an inoperable valve. For the specific case of the containment spray pump discharge (SI-860) valves, SI-860A and SI-860D provide duplicate functions. Valves SI-860B and SI-860C are not required for system operability. Hence, prior to removing valve SI-860A from service, valve SI-860D must be operable and vice versa.

The component cooling system is different from the other systems discussed above in that the components are so located in the Auxiliary Building as to be accessible for repair after a loss-of-coolant accident. The component cooling water pump together with one component cooling heat exchanger can accommodate the heat removal load on one unit either following a loss-of-coolant accident, or during normal plant shutdown. If during the post-accident phase the component cooling water supply is lost, core and containment cooling could be maintained until repairs were effected.[5]

< See Section 3.7 >

TABLE 15.4.1-2

MINIMUM FREQUENCIES FOR EQUIPMENT AND SAMPLING TESTS

		Test	Frequency
1.	Reactor Coolant Samples	Gross Beta-gamma activity (excluding tritium)	5/week(7)
		Tritium activity	Monthly
		Radiochemical E Determination	Semiannually (2)(10)
		Isotopic Analysis for Dose Equivalent I-131 Concentration	Every two weeks(1)
		Isotopic Analysis for Iodine including I-131, I-133, and I-135	a.) Once per 4 hours whenever the specific activity exceeds 0.8 mCi/ gram Dose Equivalent I-131 or 100/E mCi/g(6) b.) One sample between 2 and 6 hours following a thermal power change exceeding 15% of rated power in a one-hour period.
		Chloride Concentration	5/week(8)
		Diss. Oxygen Conc.	5/week(6)
		Fluoride Conc.	Weekly
2.	Reactor Coolant Boron	Boron Concentration	Twice/week
3.	Refueling Water Storage Tank Water Sample	Boron Concentration	Weekly(6)
4.	Boric Acid Tanks	Boron Concentration	Twice/week and after each BAST concentration change when they are being relied upon as a source of borated water.
5.	Spray Additive Tank	NaOH Concentration	Monthly
6.	Accumulator	Boron Concentration	Monthly

< See Section 3.4 >

SR 3.5.4.3 A.4

< See Section 3.9 >

< See LCO 3.5.2 >

< See Section 3.6 >

< See LCO 3.5.1 >

Add New SR 3.5.4.1 - RWST Temp Verification M.4

TABLE 15.4.1-2 (Continued)

30.	Pressurizer Heaters	Verify that 100 KW of heaters are available.	Quarterly	< See Section 3.4 >
31.	CVCS Charging Pumps	Verify operability pumps.[17]	Quarterly	< See LCO 3.5.2 >
32.	Potential Dilution in Alarm < See Section 3.3 >	Verify operability of alarm.	Prior to placing plant in Progress cold shutdown.	
33.	Core Power Distribution < See Section 3.2 >	Perform power distribution maps using movable incore detector system to confirm hot channel factors.	Monthly [20]	
34.	Shutdown Margin	Perform shutdown margin calculation.	Daily[21]	< See Section 3.1 >

< See Section 3.4 > < See Section 3.1 >

.4

(1) ~~Required only during periods of power operation.~~
(2) Q determination will be started when the gross activity analysis of a filtered sample indicates ³10mCi/cc and will be redetermined if the primary coolant gross radioactivity of a filtered sample increases by more than 10mCi/cc.
(3) ~~Drop test shall be conducted at rated reactor coolant flow. Rods shall be dropped under both cold and~~ hot condition, but cold drop tests need not be timed.
(4) Drop tests will be conducted in the hot condition for rods on which maintenance was performed.
(5) ~~As accessible without disassembly of rotor.~~
(6) ~~Not required during periods of refueling shutdown.~~
(7) ~~At least once per week during periods of refueling shutdown.~~
(8) At least three times per week (with maximum time of 72 hours between samples) during periods of refueling shutdown.

< See Section 3.4 >

(9) ~~Not required during periods of cold or refueling shutdown, but must be performed prior to exceeding 200°F if it~~ has not been performed during the previous surveillance period.
(10) ~~Sample to be taken after a minimum of 2 EFPD and 20 days power operation since the reactor was last~~ subcritical for 48 hours or longer.
(11) ~~An approximately equal number of valves shall be tested each refueling outage such that all valves will be tested~~ within a five year period. If any valve fails its tests , an additional number of valves equal to the number originally tested shall be tested. If any of the additional tested valves fail, all remaining valves shall be tested.
(12) ~~The specified buses shall be determined energized in the required manner at least once per shift by verifying~~ correct static transfer switch alignment and indicated voltage on the buses.
(13) ~~Not required if the block valve is shut to isolate a PORV that is inoperable for reasons other than excessive seat~~ leakage.
(14) Only applicable when the overpressure mitigation system is in service.
(15) Required to be performed only if conditions will be established, as defined in Specification 15.3.15, where the PORVs are used for low temperature overpressure protection. The test must be performed prior to establishing these conditions.

< See Section 3.4 > < See Section 3.8 >

< See Section 3.4 and 3.6 >

Associated Specification removed with Unit 1 Amendment 176 and Unit 2 Amendment 180

< See Section 3.4 and 3.7 >

< See Section 3.0 >

15.3 Limiting Conditions for Operation

15.3.0 General Considerations

A. Many of the Limiting Conditions for Operation (LCO) presented in these specifications provide a temporary relaxation of the single failure criterion, which is consistent with overall reliability considerations, to allow time periods during which corrective action may be taken to restore the system to full operability. If the situation has not been corrected within the the specified time period, and the LCO prescribes no other specific action, action shall be initiated within one hour to place the affected unit in:

1. Hot shutdown within seven hours of entering this specification; AND
2. Cold shutdown within 37 hours of entering this specification.

This specification is applicable during power operation, low power operation, and shutdown with temperature ≥200°F.

- FYI -
CTS default
Action

Cond B and C

A.8

B. In the event an LCO cannot be satisfied because of equipment failures or limitations beyond those specified in the permissible conditions of the LCO, action shall be initiated within one hour to place the affected unit in:

1. Hot shutdown within seven hours of entering this specification; AND
2. Cold shutdown within 37 hours of entering this specification.

This specification is applicable during power operation, low power operation, and shutdown with temperature ≥200°F

C. Upon discovery of a Limiting Condition for Operation, the actions delineated in the specification shall be performed. If the requirements of the Limiting Condition for Operation are met or are no longer applicable prior to the expiration of the times delineated in the specification, completion of the specified actions is not required, unless otherwise stated.

Add New Cond A -
RWST Boron/Temp
Out of Limits

L.1

Justification For Deviations - NUREG-1431 Section 3.05.04

13-Nov-99

JFD Number	JFD Text	
01	The brackets have been removed and the proper plant specific information has been provided.	
	ITS:	**NUREG:**
	B 3.05.04	B 3.05.04
	SR 3.05.04.01	SR 3.05.04.01
	SR 3.05.04.02	SR 3.05.04.02
	SR 3.05.04.03	SR 3.05.04.03
02	The Bases for NUREG 1431 states that the ECCS and Containment Spray pumps take suction from separate redundant supply lines during the injection phase of a loss of coolant accident. The Point Beach ECCS and Containment Spray pumps are supplied from a common header with branch lines containing motor operated isolation valves used to isolate the RWST from the ECCS pumps during the recirculation phase of an accident. Multiple motor operated valves are used to prevent a single failure from establishing recirculation line up. As such, the Bases has been modified to delete reference to separate and redundant supply headers, reflective of Point Beach's design.	
	ITS:	**NUREG:**
	B 3.05.04	B 3.05.04
03	The Bases for LCO 3.5.4 contains a description of the Volume Control Tank to RWST suction supply auto swap over feature associated with the centrifugal charging pumps. The Bases also contains descriptive information only applicable to a plant with a Boron Injection Tank (BIT). The centrifugal charging pumps are not ECCS pumps for Point Beach and Point Beach does not have a BIT. Accordingly, all discussions related to these features have been deleted from the Bases of the proposed ITS.	
	ITS:	**NUREG:**
	B 3.05.04	B 3.05.04
04	The Bases for LCO 3.5.4 (RWST) describes the ECCS and Containment Spray pumps as having recirculation lines. This is true only for the ECCS pumps at Point Beach. Minimum flow protection for the Containment Spray pumps is provided by the spray educator line. Based on this statement not being correct as applied to the Point Beach design, and the fact that this LCO is for the RWST and not the ECCS and Containment Spray pumps, this statement has been deleted from the proposed ITS Bases for LCO 3.5.4.	
	ITS:	**NUREG:**
	B 3.05.04	B 3.05.04

Justification For Deviations - NUREG-1431 Section 3.05.04

13-Nov-99

JFD Number	JFD Text	
05	The Bases for LCO 3.5.4 contains a number of statement related to a design for which the suction supply to the Containment Spray pumps can be either from the RWST or the Containment Sump. The suction to the Containment Spray pumps at Point Beach is from the RWST alone. Accordingly, statements related to a Containment Sump suction supply to the Containment Spray Pumps has been deleted from the proposed Bases for LCO 3.5.4. Recirculation Spray can be provided by the RHR System only, which has been addressed within the Bases for RHR (LCO 3.5.2).	
	ITS:	**NUREG:**
	B 3.05.04	B 3.05.04
06	The current licensing basis for Point Beach does not include feedwater line break scenarios or inadvertent safety injection. Accordingly, reference to Feedwater line break events and inadvertent safety injection analyses in the Bases of the proposed ITS have been deleted. Minor wording changes have also been proposed to clarify statements made in the Bases.	
	ITS:	**NUREG:**
	B 3.05.04	B 3.05.04
07	NUREG 1431 refers to the Accumulators as an ECCS component, while the terminology and labeling at Point Beach refers to these components as Safety Injection (SI) Accumulators. The LCO title and associated Bases statements have been changed to reflect this site specific terminology.	
	ITS:	**NUREG:**
	B 3.05.04	B 3.05.04
	LCO 3.05.04	LCO 3.05.04
08	The Bases for NUREG 1431 states that the ECCS pumps are provided with motor operated isolation valves in each suction header to isolated the RWST once the recirculation mode of ECCS is entered. The Point Beach ECCS pumps are supplied from a common header, with isolation valves located on the suction supply to each of the ECCS pumps. As such, the Bases has been changed to state that isolation of the RWST is provided by valves on the supply lines versus each header, reflective of Point Beach's design.	
	ITS:	**NUREG:**
	B 3.05.04	B 3.05.04

JFD Number	JFD Text
09	NUREG 1431 contains a Surveillance Requirement requiring verification of RWST temperature once every 24 hours when ambient air temperature is outside of the required RWST temperature band. The proposed ITS for Point Beach will require performance of this Surveillance Requirement every 24 hours, with no allowance to suspend performance based on ambient air temperature. The RWST at Point Beach is located within a structure which surrounds the containment (containment facade) with no effective means monitoring ambient temperature on a continuous basis to establish the required performance interval. Accordingly, this provision has been omitted. **ITS:** B 3.05.04; SR 3.05.04.01 NOTE **NUREG:** B 3.05.04; SR 3.05.04.01 NOTE
10	The Bases have been modified to reflect the accident analyses assumptions and limiting analyses for RWST temperature. The lower limiting lower temperature limit (40 degrees) for the RWST was obtained from the Main Steam Line Break analysis, while the upper temperature limit is based on the containment integrity analysis. **ITS:** B 3.05.04 **NUREG:** B 3.05.04
11	The ECCS systems at Point Beach do not include hot leg recirculation as a mode of ECCS operation. The Bases has been changed to accurately reflect the Point Beach design. The Point Beach design incorporates an injection phase and a recirculation phase. The RHR subsystem normally supplies injection to the RCS via the upper plenum injection nozzles. The SI subsystem supplies injection via the RCS cold legs. To avoid excessive boron precipitation, ECCS can be operated in a simultaneous injection configuration which is a subset of the recirculation mode. In the simultaneous injection mode, suction is transferred to the containment sump, the RHR subsystem supplies upper plenum injection and suction to the SI subsystems, while the SI subsystems provide injection into the cold legs. **ITS:** B 3.05.04 **NUREG:** B 3.05.04

3.5 EMERGENCY CORE COOLING SYSTEMS (ECCS)

3.5.4 Refueling Water Storage Tank (RWST)

LCO 3.5.4 The RWST shall be OPERABLE.

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

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. RWST boron concentration not within limits. OR RWST borated water temperature not within limits.	A.1 Restore RWST to OPERABLE status.	8 hours
B. RWST inoperable for reasons other than Condition A.	B.1 Restore RWST to OPERABLE status.	1 hour
C. Required Action and associated Completion Time not met.	C.1 Be in MODE 3. AND C.2 Be in MODE 5.	6 hours 36 hours

SURVEILLANCE REQUIREMENTS

	SURVEILLANCE	FREQUENCY
SR 3.5.4.1	9 ~~-------NOTE------- Only required to be performed when ambient air temperature is < [35]°F or > [100]°F.~~	
1	40 Verify RWST borated water temperature is ≥ [35]°F and ≤ [100]°F.	24 hours
SR 3.5.4.2	Verify RWST borated water volume is ≥ [466,200 gallons ( )%]. 275,000 gallons 1	7 days
SR 3.5.4.3	Verify RWST boron concentration is ≥ [2000] ppm and ≤ [2200] ppm. 2700 1 3200 1	7 days

B 3.5 EMERGENCY CORE COOLING SYSTEMS (ECCS)

B 3.5.4 Refueling Water Storage Tank (RWST)

BASES

BACKGROUND

The RWST supplies borated water to the Chemical and Volume Control System (CVCS) during abnormal operating conditions, to the refueling pool during refueling, and to the ECCS and the Containment Spray System during accident conditions.

02 Motor s

02 are the supply line

The RWST supplies both trains of the ECCS and the Containment Spray System ~~through separate, redundant supply headers~~ the supply line during the injection phase of a loss of coolant accident (LOCA) recovery. ~~A motor~~ Motor operated isolation valves ~~is~~ are provided in ~~each header~~ the supply line to isolate the RWST from the ECCS once the system has been transferred to the recirculation mode. The recirculation mode is entered when pump suction is transferred to the containment sump following receipt of the RWST-Low Low (Level 1) signal. Use of a single RWST to supply both trains of the ECCS and Containment Spray System is acceptable since the RWST is a passive component, and passive failures are not required to be assumed to occur coincidentally with Design Basis Events.

03

~~The switchover from normal operation to the injection phase of ECCS operation requires changing centrifugal charging pump suction from the CVCS volume control tank (VCT) to the RWST through the use of isolation valves. Each set of isolation valves is interlocked so that the VCT isolation valves will begin to close once the RWST isolation valves are fully open. Since the VCT is under pressure, the preferred pump suction will be from the VCT until the tank is isolated. This will result in a delay in obtaining the RWST borated water. The effects of this delay are discussed in the Applicable Safety Analyses section of these Bases.~~

During normal operation in MODES 1, 2, and 3, the safety injection (SI) and residual heat removal (RHR) pumps are aligned to take suction from the RWST.

04

~~The ECCS and Containment Spray System pumps are provided with recirculation lines that ensure each pump can maintain~~

BASES

BACKGROUND (Continued)

04

~~minimum flow requirements when operating at or near shutoff head conditions.~~

When the suction for the ECCS ~~and Containment Spray System~~ pumps is transferred to the containment sump, the RWST flow paths must be isolated to prevent a release of the containment sump contents to the RWST, which could result in a release of contaminants to the atmosphere and the eventual loss of suction head for the ECCS pumps.

04

This LCO ensures that:

a. The RWST contains sufficient borated water to support the ECCS during the injection phase;

b. Sufficient water volume exists in the containment sump to support continued operation of the ECCS ~~and Containment Spray System~~ pumps at the time of transfer to the recirculation mode of cooling; and

04

c. The reactor remains subcritical following a LOCA.

Insufficient water in the RWST could result in insufficient cooling capacity when the transfer to the recirculation mode occurs. Improper boron concentrations could result in a reduction of SDM or excessive boric acid precipitation in the core following the LOCA, as well as excessive caustic stress corrosion of mechanical components and systems inside the containment.

APPLICABLE SAFETY ANALYSES

During accident conditions, the RWST provides a source of borated water to the ECCS and Containment Spray System pumps. As such, it provides containment cooling and depressurization, core cooling, and replacement inventory and is a source of negative reactivity for reactor shutdown (Ref. 1). The design basis transients and applicable safety analyses concerning each of these systems are discussed in the Applicable Safety Analyses section of B 3.5.2, "ECCS - Operating"; B 3.5.3, "ECCS - Shutdown"; and B 3.6.6, "Containment Spray and Cooling Systems." These analyses are used to assess changes to the RWST in order to evaluate

BASES

APPLICABLE SAFETY ANALYSES (continued)

their effects in relation to the acceptance limits in the analyses.

The RWST must also meet volume, boron concentration, and temperature requirements for non-LOCA events. The volume is not an explicit assumption in non-LOCA events since the required volume is a small fraction of the available volume. The deliverable volume limit is set by the LOCA and containment analyses. For the RWST, the deliverable volume is different from the total volume contained since, due to the design of the tank, more water can be contained than can be delivered. The minimum boron concentration is an explicit assumption in the main steam line break (MSLB) analysis to ensure the required shutdown capability. 3 ~~The importance of its value is small for units with a boron injection tank (BIT) with a high boron concentration. For units with no BIT or reduced BIT boron requirements, the minimum boron concentration limit is an important assumption in ensuring the required shutdown capability.~~ 6 ~~The maximum boron concentration is an explicit assumption in the inadvertent ECCS actuation analysis, although it is typically a nonlimiting event and the results are very insensitive to boron concentrations.~~ The maximum temperature ensures that the amount of cooling provided from the RWST during the heatup phase of a ~~feedline break~~ [6: LOCA] is [temperature] consistent with safety analysis assumptions; the minimum is an assumption in both the MSLB ~~and inadvertent ECCS actuation analyses, although the inadvertent ECCS actuation event is typically nonlimiting.~~

6/10
Replace with Insert B 3.5.4-1

3 ~~The MSLB analysis has considered a delay associated with the interlock between the VCT and RWST isolation valves, and the results show that the departure from nucleate boiling design basis is met. The delay has been established as [27] seconds, with offsite power available, or [37] seconds without offsite power. This response time includes [2] seconds for electronics delay, a [15] second stroke time for the RWST valves, and a [10] second stroke time for the VCT valves. Plants with a BIT need not be concerned with the delay since the BIT will supply highly borated water prior to RWST switchover, provided the BIT is between the pumps and the core.~~

BASES

APPLICABLE SAFETY ANALYSES (continued)

1 275,000 2700

For a large break LOCA analysis, the minimum water volume limit of [466,200] gallons and the lower boron concentration limit of [2000] ppm are used to compute the post LOCA sump boron concentration necessary to assure subcriticality. The large break LOCA is the limiting case since the safety analysis assumes that all control rods are out of the core.

11 Replace with Insert B 3.5.4-3

~~The upper limit on boron concentration of [2200] ppm is used to determine the maximum allowable time to switch to hot leg recirculation following a LOCA. The purpose of switching from cold leg to hot leg injection is to avoid boron precipitation in the core following the accident.~~

10 Replace with Insert B 3.5.4-2

~~In the ECCS analysis, the containment spray temperature is assumed to be equal to the RWST lower temperature limit of [35]°F. If the lower temperature limit is violated, the containment spray further reduces containment pressure, which decreases the rate at which steam can be vented out the break and increases peak clad temperature. The upper temperature limit of [100]°F is used in the small break LOCA analysis and containment OPERABILITY analysis. Exceeding this temperature will result in a higher peak clad temperature, because there is less heat transfer from the core to the injected water for the small break LOCA and higher containment pressures due to reduced containment spray cooling capacity.~~ 12 ~~For the containment response following an MSLB, the lower limit on boron concentration and the upper limit on RWST water temperature are used to maximize the total energy release to containment.~~

The RWST satisfies Criterion 3 of the NRC Policy Statement.

LCO

The RWST ensures that an adequate supply of borated water is available to cool and depressurize the containment in the event of a Design Basis Accident (DBA), to cool and cover the core in the event of a LOCA, to maintain the reactor subcritical following a DBA, and to ensure adequate level in the containment sump to support ECCS and Containment Spray System pump operation in the recirculation mode.

BASES

LCO (continued)

To be considered OPERABLE, the RWST must meet the water volume, boron concentration, and temperature limits established in the SRs.

APPLICABILITY

In MODES 1, 2, 3, and 4, RWST OPERABILITY requirements are dictated by ECCS and Containment Spray System OPERABILITY requirements. Since both the ECCS and the Containment Spray System must be OPERABLE in MODES 1, 2, 3, and 4, the RWST must also be OPERABLE to support their operation. Core cooling requirements in MODE 5 are addressed by LCO 3.4.7, "RCS Loops-MODE 5, Loops Filled," and LCO 3.4.8, "RCS Loops -MODE 5, Loops Not Filled." MODE 6 core cooling requirements are addressed by 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."

ACTIONS

A.1

With RWST boron concentration or borated water temperature not within limits, they must be returned to within limits within 8 hours. Under these conditions neither the ECCS nor the Containment Spray System can perform its design function. Therefore, prompt action must be taken to restore the tank to OPERABLE condition. The 8 hour limit to restore the RWST temperature or boron concentration to within limits was developed considering the time required to change either the boron concentration or temperature and the fact that the contents of the tank are still available for injection.

B.1

With the RWST inoperable for reasons other than Condition A (e.g., water volume), it must be restored to OPERABLE status within 1 hour.

In this Condition, neither the ECCS nor the Containment Spray System can perform its design function. Therefore, prompt action must be taken to restore the tank to OPERABLE status or to place the plant in a MODE in which the RWST is

BASES

ACTIONS (continued)

not required. The short time limit of 1 hour to restore the RWST to OPERABLE status is based on this condition simultaneously affecting redundant trains.

C.1 and C.2

If the RWST cannot be returned to OPERABLE status within the associated Completion Time, the plant must be brought to a MODE in which the LCO does not apply. To achieve this status, the plant must be brought to at least MODE 3 within 6 hours and to MODE 5 within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.5.4.1

The RWST borated water temperature should be verified every 24 hours to be within the limits assumed in the accident analyses band. This Frequency is sufficient to identify a temperature change that would approach either limit and has been shown to be acceptable through operating experience.

9

~~The SR is modified by a Note that eliminates the requirement to perform this Surveillance when ambient air temperatures are within the operating limits of the RWST. With ambient air temperatures within the band, the RWST temperature should not exceed the limits.~~

SR 3.5.4.2

The RWST water volume should be verified every 7 days to be above the required minimum level in order to ensure that a sufficient initial supply is available for injection and to support continued ECCS and Containment Spray System pump operation on recirculation. Since the RWST volume is normally stable and is protected by an alarm, a 7 day Frequency is appropriate and has been shown to be acceptable through operating experience.

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.5.4.3

The boron concentration of the RWST should be verified every 7 days to be within the required limits. This SR ensures that the reactor will remain subcritical following a LOCA. Further, it assures that the resulting sump pH will be maintained in an acceptable range so that boron precipitation in the core will not occur and the effect of chloride and caustic stress corrosion on mechanical systems and components will be minimized. Since the RWST volume is normally stable, a 7 day sampling Frequency to verify boron concentration is appropriate and has been shown to be acceptable through operating experience.

REFERENCES

1. FSAR, Chapter $\boxed{[6]}$ and Chapter $\boxed{[15]}$.

5 14

1

BASES INSERTS

INSERT B 3.5.4-1:

and large break LOCA, although the large break LOCA assumption is not the limiting value.

INSERT B 3.5.4-2:

In the large break LOCA analysis, the containment spray temperature is assumed to be 33°F, maximizing containment cooling capability, thereby minimizing containment pressure. Minimizing containment pressure increases RCS blowdown rate, increasing core reflood time, which results in higher peak clad temperatures. The upper temperature limit of 100°F is used in the containment integrity analysis. Exceeding this temperature will result in higher containment pressures due to reduced containment spray cooling capacity.

INSERT B 3.5.4-3:

The upper limit on boron concentration is used in determining the maximum allowable time to switch simultaneous injection following a LOCA. The purpose of switching simultaneous injection is to avoid boron precipitation in the core following the accident.

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? This change does not result in any hardware changes. The RWST is not assumed to be an initiator of any analyzed event. Establishing a Completion Time to restore the RWST to OPERABLE status does not affect the probability of an accident. The RWST volume will continue to be available during this time period. Therefore, the RWST will still be functional in that the RWST inventory is still available for injection into the core and containment. Because of the large RWST volume, boron concentration and temperature change very slowly, thus these parameters should not be significantly out of limits, thereby have an insignificant effect on analyzed events Therefore, the proposed 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 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 only provide an 8-hour Completion Time to restore the RWST boron or temperature to within limits before requiring a plant shutdown. 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 8-hour Completion Time allowed to restore the RWST to OPERABLE status prior to requiring a unit shutdown is based on the fact that the contents of the tank are still available for injection. Furthermore, any violation of these limits would generally result from minor deviations from the specified requirements (temperature/boron concentration). The probability of an event requiring the RWST as a source of water during this time period is small. Allowing 8-hours to return the RWST to OPERABLE will also minimize the potential for plant transients that can occur during the shutdown. As such, any reduction in a margin of safety will be insignificant and offset by the benefit of avoiding an unnecessary plant transient.

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.

3.5 EMERGENCY CORE COOLING SYSTEMS (ECCS)

3.5.4 Refueling Water Storage Tank (RWST)

LCO 3.5.4 The RWST shall be OPERABLE.

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

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. RWST boron concentration not within limits. OR RWST borated water temperature not within limits.	A.1 Restore RWST to OPERABLE status.	8 hours
B. RWST inoperable for reasons other than Condition A.	B.1 Restore RWST to OPERABLE status.	1 hour
C. Required Action and associated Completion Time not met.	C.1 Be in MODE 3. AND C.2 Be in MODE 5.	6 hours 36 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.5.4.1	Verify RWST borated water temperature is ≥ 40°F and ≤ 100°F.	24 hours
SR 3.5.4.2	Verify RWST borated water volume is ≥ 275,000 gallons.	7 days
SR 3.5.4.3	Verify RWST boron concentration is ≥ 2700 ppm and ≤ 3200 ppm.	7 days

B 3.5 EMERGENCY CORE COOLING SYSTEMS (ECCS)

B 3.5.4 Refueling Water Storage Tank (RWST)

BASES

BACKGROUND

The RWST supplies borated water to the Chemical and Volume Control System (CVCS) during abnormal operating conditions, to the refueling pool during refueling, and to the ECCS and the Containment Spray System during accident conditions.

The RWST supplies both trains of the ECCS and the Containment Spray System during the injection phase of a loss of coolant accident (LOCA) recovery. Motor operated isolation valves are provided in the supply line to isolate the RWST from the ECCS once the system has been transferred to the recirculation mode. The recirculation mode is entered when pump suction is transferred to the containment sump following receipt of the RWST - Low Low (Level 1) signal. Use of a single RWST to supply both trains of the ECCS and Containment Spray System is acceptable since the RWST is a passive component, and passive failures are not required to be assumed to occur coincidentally with Design Basis Events.

During normal operation in MODES 1, 2, and 3, the safety injection (SI) and residual heat removal (RHR) pumps are aligned to take suction from the RWST.

When the suction for the ECCS pumps is transferred to the containment sump, the RWST flow paths must be isolated to prevent a release of the containment sump contents to the RWST, which could result in a release of contaminants to the atmosphere and the eventual loss of suction head for the ECCS pumps.

This LCO ensures that:

a. The RWST contains sufficient borated water to support the ECCS during the injection phase;

b. Sufficient water volume exists in the containment sump to support continued operation of the ECCS pumps at the time of transfer to the recirculation mode of cooling; and

BASES

BACKGROUND (Continued)

c. The reactor remains subcritical following a LOCA.

Insufficient water in the RWST could result in insufficient cooling capacity when the transfer to the recirculation mode occurs. Improper boron concentrations could result in a reduction of SDM or excessive boric acid precipitation in the core following the LOCA, as well as excessive caustic stress corrosion of mechanical components and systems inside the containment.

APPLICABLE SAFETY ANALYSES

During accident conditions, the RWST provides a source of borated water to the ECCS and Containme nt Spray System pumps. As such, it provides containment cooling and depressurization, core cooling, and replacement inventory and is a source of negative reactivity for reactor shutdown (Ref. 1). The design basis transients and applicable safety analyses concerning each of these systems are discussed in the Applicable Safety Analyses section of B 3.5.2, "ECCS - Operating"; B 3.5.3, "ECCS - Shutdown"; and B 3.6.6, "Containment Spray and Cooling Systems." These analyses are used to assess changes to the RWST in order to evaluate their effects in relation to the acceptance limits in the analyses.

The RWST must also meet volume, boron concentration, and temperature requirements for non -LOCA events. The volume is not an explicit assumption in non -LOCA events since the required volume is a small fraction of the available volume. The deliverable volume limit is set by the LOCA and containment analyses. For the RWST, the deliverable volume is different from the total volume contained since, due to the design of the tank, more water can be contained than can be delivered. The minimum boron concentration is an explicit assumption in the main steam line break (MSLB) analysis to ensure the required shutdown capability. The maximum temperature ensures that the amount of cooling provided from the RWST during the heatup phase of a LOCA is consistent with safety analysis assumptions; the minimum temperature is an assumption in both the MSLB and large break LOCA, although the large break LOCA assumption is not the limiting value.

BASES

APPLICABLE SAFETY ANALYSES (continued)

For a large break LOCA analysis, the minimum water volume limit of 275,000 gallons and the lower boron concentration limit of 2700 ppm are used to compute the post LOCA sump boron concentration necessary to assure subcriticality. The large break LOCA is the limiting case since the safety analysis assumes that all control rods are out of the core.

The upper limit on boron concentration is used in determining the maximum allowable time to switch simultaneous injection following a LOCA. The purpose of switching simultaneous injection is to avoid boron precipitation in the core following the accident.

In the large break LOCA analysis, the containment spray temperature is assumed to be 33°F, maximizing containment cooling capability, thereby minimizing containment pressure. Minimizing containment pressure increases RCS blowdown rate, increasing core reflood time, which results in higher peak clad temperatures. The upper temperature limit of 100°F is used in the containment integrity analysis. Exceeding this temperature will result in higher containment pressures due to reduced containment spray cooling capacity.

The RWST satisfies Criterion 3 of the NRC Policy Statement.

LCO

The RWST ensures that an adequate supply of borated water is available to cool and depressurize the containment in the event of a Design Basis Accident (DBA), to cool and cover the core in the event of a LOCA, to maintain the reactor subcritical following a DBA, and to ensure adequate level in the containment sump to support ECCS and Containment Spray System pump operation in the recirculation mode.

To be considered OPERABLE, the RWST must meet the water volume, boron concentration, and temperature limits established in the SRs.

APPLICABILITY

In MODES 1, 2, 3, and 4, RWST OPERABILITY requirements are dictated by ECCS and Containment Spray System OPERABILITY

BASES

APPLICABILITY (continued)

requirements. Since both the ECCS and the Containment Spray System must be OPERABLE in MODES 1, 2, 3, and 4, the RWST must also be OPERABLE to support their operation. Core cooling requirements in MODE 5 are addressed by LCO 3.4.7, "RCS Loops – MODE 5, Loops Filled," and LCO 3.4.8, "RCS Loops – MODE 5, Loops Not Filled." MODE 6 core cooling requirements are addressed by 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."

ACTIONS

A.1

With RWST boron concentration or borat ed water temperature not within limits, they must be returned to within limits within 8 hours. Under these conditions neither the ECCS nor the Containment Spray System can perform its design function. Therefore, prompt action must be taken to restore the tank to OPERABLE condition. The 8 hour limit to restore the RWST temperature or boron concentration to within limits was developed considering the time required to change either the boron concentration or temperature and the fact that the contents of the tank are still available for injection.

B.1

With the RWST inoperable for reasons other than Condition A (e.g., water volume), it must be restored to OPERABLE status within 1 hour. In this Condition, neither the ECCS nor the Containment Spray System can perform its design function. Therefore, prompt action must be taken to restore the tank to OPERABLE status or to place the plant in a MODE in which the RWST is not required. The short time limit of 1 hour to restore the RWST to OPERABLE status is based on this condition simultaneously affecting redundant trains.

C.1 and C.2

If the RWST cannot be returned to OPERABLE status within the associated Completion Time, the plant must be brought to a MODE in which the LCO does not apply. To achieve this status, the plant must be brought to at least MODE 3 within

BASES

ACTIONS (continued)

6 hours and to MODE 5 within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.5.4.1

The RWST borated water temperature should be verified every 24 hours to be within the limits assumed in the accident analyses band. This Frequency is sufficient to identify a temperature change that would approach either limit and has been shown to be acceptable through operating experience.

SR 3.5.4.2

The RWST water volume should be verified every 7 days to be above the required minimum level in order to ensure that a sufficient initial supply is available for injection and to support continued ECCS and Containment Spray System pump operation on recirculation. Since the RWST volume is normally stable and is protected by an alarm, a 7 day Frequency is appropriate and has been shown to be acceptable through operating experience.

SR 3.5.4.3

The boron concentration of the RWST should be verified every 7 days to be within the required limits. This SR ensures that the reactor will remain subcritical following a LOCA. Further, it assures that the resulting sump pH will be maintained in an acceptable range so that boron precipitation in the core will not occur and the effect of chloride and caustic stress corrosion on mechanical systems and components will be minimized. Since the RWST volume is normally stable, a 7 day sampling Frequency to verify boron concentration is appropriate and has been shown to be acceptable through operating experience.

REFERENCES

1. FSAR, Chapter 5 and Chapter 14.

No Equivalent CTS Requirement Exists

Justification For Deviations - NUREG-1431 Section 3.05.06

13-Nov-99

JFD Number	JFD Text	
01	The Point Beach design does not include a Boric Acid Injection Tank. Accordingly, this LCO has not been incorporated as part of the Point Beach conversion to the ITS.	
	ITS:	**NUREG:**
	N/A	B 3.05.06
		LCO 3.05.06
		LCO 3.05.06 COND A
		LCO 3.05.06 COND A RA A.1
		LCO 3.05.06 COND B
		LCO 3.05.06 COND B RA B.1
		LCO 3.05.06 COND B RA B.2
		LCO 3.05.06 COND B RA B.3
		LCO 3.05.06 COND C
		LCO 3.05.06 COND C RA C.1
		SR 3.05.06.01
		SR 3.05.06.02
		SR 3.05.06.03

1

3.5 EMERGENCY CORE COOLING SYSTEMS (ECCS)

3.5.6 Boron Injection Tank (BIT)

LCO 3.5.6 The BIT shall be OPERABLE.

APPLICABILITY: MODES 1, 2, and 3.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. BIT inoperable.	A.1 Restore BIT to OPERABLE status.	1 hour
B. Required Action and associated Completion Time of Condition A not met.	B.1 Be in MODE 3. AND B.2 Borate to an SDM equivalent to [1]% Dk/k at 200°F. AND B.3 Restore BIT to OPERABLE status.	6 hours 6 hours 7 days
C. Required Action and associated Completion Time of Condition B not met.	C.1 Be in MODE 4.	12 hours

No Equivalent CTS Requirement Exists

Justification For Deviations - NUREG-1431 Section 3.05.05

13-Nov-99

JFD Number	JFD Text	
01	Point Beach is a low pressure Safety Injection Plant, which does not utilize the Charging system in a Safety Injection capacity. As discussed in the Bases Section of NUREG 1431, LCO 3.5.5 is only applicable to those units that utilize the centrifugal charging pumps for safety injection. Accordingly, this LCO has not been adopted in the Point Beach ITS.	
	ITS:	**NUREG:**
	N/A	B 3.05.05
		LCO 3.05.05
		LCO 3.05.05 COND A
		LCO 3.05.05 COND A RA A.1
		LCO 3.05.05 COND B
		LCO 3.05.05 COND B RA B.1
		LCO 3.05.05 COND B RA B.2
		SR 3.05.05.01
		SR 3.05.05.01 NOTE

3.5 EMERGENCY CORE COOLING SYSTEMS (ECCS)

3.5.5 Seal Injection Flow

LCO 3.5.5 Reactor coolant pump seal injection flow shall be ≤ [40] gpm with [centrifugal charging pump discharge header] pressure ≥ [2480] psig and the [charging flow] control va lve full open.

APPLICABILITY: MODES 1, 2, and 3.

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. Seal injection flow not within limit.	A.1	Adjust manual seal injection throttle valves to give a flow within limit with [centrifugal charging pump discharge header] pressure ³ [2480] psig and the [charging flow] control valve full open.	4 hours
B. Required Action and associated Completion Time not met.	B.1	Be in MODE 3.	6 hours
	AND		
	B.2	Be in MODE 4.	12 hours

Point Beach Nuclear Plant
Units 1 and 2

Technical Specifications Improvement Project

November 1999

Volume 8

Section 3.7

Cross-Reference Report - NUREG-1431 Section 3.07.01

ITS to CTS

13-Nov-99

ITS	CTS	DOC
B 3.07.01	15.03.04 OBJ	A.03
	BASES	A.04
LCO 3.07.01	15.03.04	A.01
	15.03.04 APPL	A.02
	15.03.04.A	M.01
	15.03.04.A.01	A.05
	15.03.04.A.01	M.01
LCO 3.07.01 COND NOTE	NEW	L.01
LCO 3.07.01 COND A	NEW	L.01
LCO 3.07.01 COND A RA A.1	NEW	L.01
LCO 3.07.01 COND B	NEW	L.01
LCO 3.07.01 COND B RA B.1	NEW	L.01
LCO 3.07.01 COND B RA B.2	NEW	L.01
LCO 3.07.01 COND B RA B.2 NOTE	NEW	L.01
LCO 3.07.01 COND C	NEW	M.01
LCO 3.07.01 COND C RA C.1	NEW	M.01
LCO 3.07.01 COND C RA C.2	NEW	M.01
LCO 3.07.01 T 3.07.01-01	15.03.04.A.01	A.05
	NEW	L.01
LCO 3.07.01 T 3.07.01-02	15.03.04.A.01	A.05
	NEW	M.02
SR 3.07.01.01	15.04.01 T 15.04.01-02 12	M.02
	15.04.01 T 15.04.01-02 12 (11)	LB.01
SR 3.07.01.01 NOTE	15.03.04.A	A.06

Cross-Reference Report - NUREG-1431 Section 3.07.01

CTS to ITS

13-Nov-99

CTS	ITS	DOC
15.03.04	LCO 3.07.01	A.01
15.03.04 APPL	LCO 3.07.01	A.02
15.03.04 OBJ	B 3.07.01	A.03
15.03.04.A	LCO 3.07.01	M.01
	SR 3.07.01.01 NOTE	A.06
15.03.04.A.01	LCO 3.07.01	M.01
	LCO 3.07.01	A.05
	LCO 3.07.01 T 3.07.01-01	A.05
	LCO 3.07.01 T 3.07.01-02	A.05
15.04.01 T 15.04.01-02 12	SR 3.07.01.01	M.02
15.04.01 T 15.04.01-02 12 (11)	SR 3.07.01.01	LB.01
BASES	B 3.07.01	A.04

Description of Changes - NUREG-1431 Section 3.07.01

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:**
	15.03.04	LCO 3.07.01
A.02	The CTS provides an introductory statement (Applicability), which simply states which systems/components are addressed within a given section. This same information, while worded differently, is contained within the title of each ITS LCO. Accordingly, this change is a change in format with no change in technical requirement.	
	CTS:	**ITS:**
	15.03.04 APPL	LCO 3.07.01
A.03	The CTS provides an introductory statement (Objective) at the beginning of this Section of the Technical Specifications which provide a brief summary of the purpose for this Section. This information is contained in the Bases Section of the ITS. This information does not establish any regulatory requirements for the systems and components addressed within this Section. Accordingly, deletion of this information does not alter any requirement set forth in the Technical Specifications. This change is administrative and consistent with the format and presentation for the ITS as provided in NUREG 1431.	
	CTS:	**ITS:**
	15.03.04 OBJ	B 3.07.01
A.04	The Bases of the current Technical Specifications for this section have been completely replaced by revised Bases that reflect the format and applicable content of PBNP ITS, consistent with the Standard Technical Specifications for Westinghouse Plants, NUREG-1431. The revised Bases are as shown in the PBNP ITS Bases.	
	CTS:	**ITS:**
	BASES	B 3.07.01

DOC Number	DOC Text
A.05	The CTS specifies that the minimum steam relieving capability of eight main steam safety valves shall be available. The ITS states that the MSSVs shall be operable as specified in Tables 3.7.1-1 and 3.7.1-2. ITS Table 3.7.1-1 specifies the maximum power level at which the unit can be operated based on the number of operable MSSVs, while Table 3.7.1-2 specifies the MSSV valve numbers and their associated lift settings. In specifying that the MSSVs must be operable and referring to these Tables, all eight MSSVs are required to be operable to fulfill the LCO. As such, this change is administrative. **CTS:** 15.03.04.A.01 **ITS:** LCO 3.07.01 LCO 3.07.01 T 3.07.01-01 LCO 3.07.01 T 3.07.01-02
A.06	The ITS contains a Note associated with SR 3.7.1.1 (MSSV setpoint verification), which allows MSSV setpoint testing to be performed after entry into Mode 3, but prior to entry into Mode 1 or 2. The CTS Mode of Applicability for the MSSVs is whenever the reactor coolant temperature is above 350 degrees with the reactor critical, which is equivalent to ITS Modes 1 and 2. CTS 15.4.0.1 states that surveillance requirements shall be met when the system or component is required to be operable. By applying Specification 15.4.0.1, the CTS required mode of performance for this surveillance has been determined to be equivalent to ITS Modes 1 and 2 making the ITS Note allowing entry into Mode 3 administrative. **CTS:** 15.03.04.A **ITS:** SR 3.07.01.01 NOTE

DOC Number	DOC Text
L.01	The CTS does not provide any specific Actions which address the inoperability of the MSSVs, which result in entry into CTS 15.3.0.b whenever an MSSV is determined to be inoperable. Entry into CTS 15.3.0.b requires the unit to be placed into Hot Shutdown (ITS Mode 3) within 7 hours at which time the CTS Applicability is exited and no further Technical Specification Actions are required. The ITS provide specific Conditions and Required Actions to address the inoperablity of MSSVs based on the number of inoperable valves and whether or not the Moderator Temperature Coefficient is positive, negative, or zero. If there are inoperable MSSV(s), it is necessary to limit the primary system power during steady state operation to a value that does not result in exceeding the combined steam flow capacity of the remaining OPERABLE MSSVs. This reduction is necessary to prevent primary and secondary system overpressurization and has been calculated in accordance with the conservative heat balance calculations provided in NRC Information Notice 94-60 which references Westinghouse NSAL 94-001. If the Moderator Temperature Coefficient is zero or negative, a power reduction alone is sufficient for a single inoperable MSSV on one or both Steam Generators. If the Moderator Temperature Coefficient is positive or if two or more MSSVs are inoperable on any Steam Generator, the power reduction must be accompanied by a similar reduction in the Power Range Neutron Flux-High setpoint. Reducing the Power Range Neutron Flux-High setpoint provides assurance that the reactor power will remain within the flow capacity of the remaining operable MSSVs in the event of a power increase or overshoot. If the reactor is not operating in excess of 5% power, the reduction in the Power Range Neutron Flux-High setpoint is not necessary as power increases and overshoots will not be significant enough to exceed the combined steam flow capacity of the remaining operable MSSVs.

CTS:	ITS:
NEW	LCO 3.07.01 COND NOTE
	LCO 3.07.01 COND A
	LCO 3.07.01 COND A RA A.1
	LCO 3.07.01 COND B
	LCO 3.07.01 COND B RA B.1
	LCO 3.07.01 COND B RA B.2
	LCO 3.07.01 COND B RA B.2 NOTE
	LCO 3.07.01 T 3.07.01-01

DOC Number	DOC Text
LB.01	The CTS specifies that an approximately equal number of MSSVs are to be tested for lift setpoint each refueling outage such that all valves are tested within a five year period. In addition, the CTS requires additional MSSVs to be tested based on setpoint testing failures. The sample selection size and increased sample population specified in the CTS are duplicative of the requirements specified by ASME Section XI and ASME/ANSI OM-1, 1981, as endorsed and required under 10 CFR 50.55a. Inclusion of these requirements via reference into 10 CFR 50.55a makes these requirement applicable to Point Beach without the need to duplicate these requirements in the Technical Specifications. This change is administrative.

CTS:	ITS:
15.04.01 T 15.04.01-02 12 (11)	SR 3.07.01.01

DOC Number	DOC Text
M.01	The CTS Mode of Applicability for the MSSVs is whenever reactor coolant temperature is heated above 350 degrees with the reactor critical, except during low power physics testing. The CTS does not provide any specific Actions which address the inoperability of the MSSVs, which result in entry into CTS 15.3.0.b whenever an MSSV is determined to be inoperable. Entry into CTS 15.3.0.b requires the unit to be placed into Hot Shutdown (ITS Mode 3) within 7 hours at which time the CTS Applicability is exited and no further Technical Specification Actions are required. The ITS establishes a Mode of Applicability for the MSSVs of Mode 1, 2, and 3 (RCS temperature of greater than or equal to 350 degrees). Similarly, the ITS contains a Condition and Required Action to Place the Unit in Mode 4 whenever the LCO's Required Actions and Associated Completion Times are not met, or one or more Steam Generators has three or more inoperable MSSVs. The revised Mode of Applicability and associated Actions provide assurance that the MSSV will be required to be operable whenever potential exist for a mainsteam system or RCS overpressurization as a result of a load rejection event. This change is an added restriction placed on plant operations.

CTS:	ITS:
15.03.04.A	LCO 3.07.01
15.03.04.A.01	LCO 3.07.01
NEW	LCO 3.07.01 COND C
	LCO 3.07.01 COND C RA C.1
	LCO 3.07.01 COND C RA C.2

DOC Number	DOC Text
M.02	The CTS requires periodic verification of MSSV setpoint in accordance with CTS Table 15.4.1-2, but does not list the valve numbers, nor their associated setpoints and tolerances. The proposed ITS adds a Table (3.7.1-2), which contains the MSSV number and associated setpoint. This Table also establishes an operability limit of plus or minus 3% of the MSSVs' lift setting between setpoint verifications. Following lift setpoint testing, SR 3.7.1.1 will require the MSSV to be left within 1% of their required lift setting. This change will allow the MSSVs to be considered operable with a deviation of up to 3%, relative to reporting requirements and increased sample population, but will require the valves to be left within 1% to account for setpoint drift between surveillance tests. The 3% operability limit is supported by Point Beach's accident analyses. As found MSSV setpoints have typically been approximately 1.6%. As such, the 1% as left value is an achievable/repeatable acceptance limit and is considered to be a conservative limit based on the accident analysis assumptions and MSSV setpoint drift observed to date.

CTS:	ITS:
15.04.01 T 15.04.01-02 12	SR 3.07.01.01
NEW	LCO 3.07.01 T 3.07.01-02

A.1

15.3.4 STEAM AND POWER CONVERSION SYSTEM

Applicability

A.2

Applies to the operating status of steam and power conversion system.

Objective

A.3

LCO 3.7.1

To define conditions of the steam and power conversion system steam-relieving capacity. Auxiliary Feedwater System and Service Water System operation is necessary to ensure the capability to remove decay heat from the core.

A.5

The MSSVs shall be OPERABLE as specified in Table 3.7.1-1 and Table 3.7.1-2.

Specification

Modes 1, 2, and 3

M.1

A. When the reactor coolant is heated above $350^{\circ}F$ the reactor shall not be taken critical unless the following conditions are met:

1. A minimum steam-relieving capability of eight (8) main steam safety valves available except for low power physics testing.

M.1

2. Auxiliary Feedwater System

< See LCO 3.7.5 >

a. Two Unit Operation - All four auxiliary feedwater pumps together with their associated flow paths and essential instrumentation shall be operable.

b. Single Unit Operation - Both motor driven auxiliary feedwater pumps and the turbine driven auxiliary feedwater pump associated with that unit together with their associated flow paths and essential instrumentation shall be operable.

Insert 3.7.1-1

M.1/L.1

Insert 3.7.1-2
ITS Table 3.7.1-1
Maximum Power
VS Operable
MSSVs

L.1

< See LCO 3.7.5 >

A.1

2. Single Unit Operation - One of the three operable auxiliary feedwater pumps associated with a unit may be out-of-service for the below specified times. The turbine driven auxiliary feedwater pump may be out-of-service for up to 72 hours. If the turbine driven auxiliary feedwater pump cannot be restored to service within that 72 hour time period, the reactor shall be in hot shutdown within the next 12 hours. Either one of the two motor driven auxiliary feedwater pumps may be out-of-service for up to 7 days. If the motor driven auxiliary feedwater pump cannot be restored to service within that 7 day period the operating unit shall be in hot shutdown within the next 12 hours.

D. The main steam stop valves (MS-2017 and MS-2018) and the non-return check valves (MS-2017A and MS-2018A) shall be operable. If one main steam stop valve or non-return check valve is inoperable but open, power operation may continue provided the inoperable valve is restored to operable status within 4 hours, otherwise the reactor shall be placed in a hot shutdown condition within the following 6 hours. With one or more main steam stop valves or non-return check valves inoperable, subsequent operation in the hot shutdown condition may proceed provided the inoperable valve or valves are maintained closed. An inoperable main steam stop valve or non-return check valve may however, be opened in the hot shutdown condition to cool down the affected unit and to perform testing to confirm operability.

< See LCO 3.7.2 >

Basis

A reactor shutdown from power requires removal of core decay heat. Immediate decay heat removal requirements are normally satisfied by the steam bypass to the condenser. Therefore, core decay heat can be continuously dissipated via the steam bypass to the condenser as feedwater in the steam generator is converted to steam by heat absorption. Normally, the capability to return feedwater flow to the steam generators is provided by operation of the turbine cycle feedwater system.

A.4

A.1

A.4

The eight main steam safety valves have a total combined rated capability of 6,664,000 lbs/hr. The total full power steam flow is 6,620,000 lbs/hr, therefore eight (8) main steam safety valves will be able to relieve the total full-power steam flow if necessary.

< See LCO 3.7.4, 3.7.5 and 3.7.6 >

In the unlikely event of complete loss of electrical power to the station, decay heat removal would continue to be assured for each unit by the availability of either the steam-driven auxiliary feedwater pump or one of the two motor-driven auxiliary steam generator feedwater pumps, and steam discharge to the atmosphere via the main steam safety valves or atmospheric relief valves. One motor-driven auxiliary feedwater pump can supply sufficient feedwater for removal of decay heat from a unit. The minimum amount of water in the condensate storage tanks ensures the ability to maintain each unit in a hot shutdown condition for at least one hour concurrent with a loss of all AC power.

An unlimited supply is available from the lake via either leg of the plant service water system for an indefinite time period.

Each of the AFW pumps possesses a low suction pressure trip that will protect it should a loss of feedwater occur. Additionally, should a steam generator tube rupture occur, the motor-operated steam admission valves for the turbine-driven AFW pumps serve as isolation boundaries for the affected steam generator.

The atmospheric steam dump lines are required to be operable because they are relied upon, following a steam generator tube rupture coincident with a loss of A.C. power, to cool down the Reactor Coolant System to RHR entry conditions. An atmospheric steam dump line is considered operable if it is capable of providing the controlled relief of main steam flow necessary to perform the RCS cooldown. Isolating an atmospheric steam dump line does not render it inoperable if the line can be unisolated and the RCS can still be cooled down to RHR entry conditions, through local or remote operation, within the time period required by the applicable FSAR accident analyses.

< See LCO 3.7.4 >

TABLE 15.4.1-2 (Continued)

		Test	Frequency
7.	Spent Fuel Pit < See LCO 3.7.15/16 >	a) Boron Concentration b) Water Level Verification	Monthly Weekly
8.	Secondary Coolant < See LCO 3.7.18 >	Gross Beta-gamma Activity or gamma isotopic analysis	Weekly[6]
		Iodine concentration	Weekly when gross Beta-gamma activity equals or exceeds 1.0 mCi/g [6]
9.	Control Rods < See Section 3.1 >	a) Rod drop times of all full length rods [3]	Each refueling or after maintenance that could affect proper functioning [4]
	shutdown prior to operation	b) Rodworth measurement	Following each refueling commencing power
10.	Control Rod	Partial movement of all rods	Every 2 weeks [18]
11.	Pressurizer Safety Valves	< See Section 3.4 >	Every five years [11]
12.	Main Steam Safety Valves	Set Point	~~Every five years~~ [11]
13.	Containment Isolation Trip	< See Section 3.6 and LCO 3.7.2 >	Each refueling shutdown
14.	Refueling System Interlocks	< See Section 3.9 >	Each refueling shutdown
15.	Service Water System	< See LCO 3.7.8 >	Each refueling shutdown
16.	Primary System Leakage	< See Section 3.4 >	Monthly [6]
17.	Diesel Fuel Supply	< See Section 3.8 >	Daily
18.	Deleted		
19.	Deleted		
20.	Boric Acid System < See Section 3.5 >	Storage Tank and piping temperatures ^{3} temperature required by Table 15.3.2-1	Daily[19]

LB.1 → In accordance with the Inservice Testing Program

SR 3.7.1.1 See Insert 3.7.1-3 ← A.6 M.2 LB.1

Insert 3.7.1-4 ITS Table 3.7.1-2 MSSV EPNs and Setpoints ← M.2

TABLE 15.4.1-2 (Continued)

30.	Pressurizer Heaters	Verify that 100 KW of heaters are available.	Quarterly < See Section 3.4 >
31.	CVCS Charging Pumps	Verify operability pumps.(17)	Quarterly < See Section 3.5 >
32.	Potential Dilution in Progress Alarm	Verify operability of alarm. < See Section 3.3 >	Prior to placing plant in cold shutdown.
33.	Core Power Distribution < See Section 3.2 >	Perform power distribution maps using movable incore detector system to confirm hot channel factors.	Monthly (20)
34.	Shutdown Margin	Perform shutdown margin calculation.	Daily(21) < See Section 3.1 >

(1) Required only during periods of power operation. < See Section 3.4 >

(2) Q determination will be started when the gross activity analysis of a filtered sample indicates ^{3}10mCi/cc and will be redetermined if the primary coolant gross radioactivity of a filtered sample increases by more than 10mCi/cc.

(3) Drop test shall be conducted at rated reactor coolant flow. Rods shall be dropped under both cold and hot condition, but cold drop tests need not be timed. < See Section 3.1 >

(4) Drop tests will be conducted in the hot condition for rods on which maintenance was performed.

(5) As accessible without disassembly of rotor.

(6) Not required during periods of refueling shutdown. < See LCO 3.4.16/3.3.2/3.5.4/3.4.13/3.7.18 >

(7) At least once per week during periods of refueling shutdown.

(8) At least three times per week (with maximum time of 72 hours between samples) during periods of refueling shutdown. < See Section 3.4 >

(9) Not required during periods of cold or refueling shutdown, but must be performed prior to exceeding 200°F if it has not been performed during the previous surveillance period. < See Section 3.3/3.6 >

(10) Sample to be taken after a minimum of 2 EFPD and 20 days power operation since the reactor was last subcritical for 48 hours or longer. < See Section 3.4 >

LB.1

(11) An approximately equal number of valves shall be tested each refueling outage such that ~~all valves will be tested within a five year period. If any valve fails its tests, an additional number of valves equal to the number originally tested shall be tested.~~ If any of the additional tested valves fail, all remaining valves shall be tested.

(12) The specified buses shall be determined energized in the required manner at least once per shift by verifying correct static transfer switch alignment and indicated voltage on the buses. < See Section 3.8 >

(13) Not required if the block valve is shut to isolate a PORV that is inoperable for reasons other than excessive seat leakage. < See Section 3.4 >

(14) Only applicable when the overpressure mitigation system is in service.

(15) Required to be performed only if conditions will be established, as defined in Specification 15.3.15, where the PORVs are used for low temperature overpressure protection. The test must be performed prior to establishing these conditions

Associated Specification removed with Unit 1 Amendment 176 and Unit 2 Amendment 180

L.1

INSERT 3.7.1-1:

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. One or more Steam Generators with one MSSV inoperable and Moderator Temperature Coefficient (MTC) zero or negative at all power levels.	A.1	Reduce THERMAL POWER to ≤ 49% RTP.	4 hours
B. One or more Steam Generators with two or more MSSVs inoperable. OR One or more Steam Generators with one MSSV inoperable and Moderator Temperature Coefficient (MTC) positive at any power level.	B.1	Reduce power to less than or equal to the Maximum Allowable % RTP specified in Table 3.7.1-1 for the number of OPERABLE MSSVs.	4 hours
	AND		
		--------NOTE-------- Only required in MODE 1. --------------------	
	B.2	Reduce the Power Range Neutron Flux - High reactor trip setpoint to less than the Maximum Allowable % RTP specified in Table 3.7.1-1 for the number of OPERABLE MSSVs.	36 hours
C. Required Action and associated Completion Time not met. OR One or more steam generators with three or more MSSVs inoperable.	C.1	Be in MODE 3.	6 hours
	AND		
	C.2	Be in MODE 4.	12 hours

M.1

INSERT 3.7.1-2:

1.1

Table 3.7.1-1 (page 1 of 1)
OPERABLE Main Steam Safety Valves versus
Maximum Allowable Power

NUMBER OF OPERABLE MSSVs PER STEAM GENERATOR REQUIRED OPERABLE	MAXIMUM ALLOWABLE POWER (% RTP)
4	≤ 100
3	≤ 49
2	≤ 29

INSERT 3.7.1-3:

SURVEILLANCE		FREQUENCY
SR 3.7.1.1	-------------------NOTE------------------- Only required to be performed in MODES 1 and 2. ---	A.6
	Verify each required MSSV lift setpoint per Table 3.7.1-2 in accordance with the Inservice Testing Program. Following testing, lift setting shall be within ±1%.	In accordance with the Inservice Testing Program

M.2

LB.1

INSERT 3.7.1-4:

Table 3.7.1-2 (page 1 of 1)
Main Steam Safety Valve Lift Settings

VALVE NUMBER		LIFT SETTING (psig ± 3%)
STEAM GENERATOR A	STEAM GENERATOR B	
MS 2010	MS 2005	1085
MS 2011	MS 2006	1100
MS 2012	MS 2007	1125
MS 2013	MS 2008	1125

M.2

Justification For Deviations - NUREG-1431 Section 3.07.01

13-Nov-99

JFD Number	JFD Text
01	The NUREG and associated Bases have been modified to incorporate the conservative heat balance calculation contained in NRC Information Notice 94-60 to derive the maximum allowable power level and Power Range High Neutron-Flux trip setpoint for continued operation whenever an MSSV is inoperable. In addition, the Required Actions have been modified to reflect the need to reduce the Power Range High Neutron-Flux trip setpoint whenever the Moderator Temperature Coefficient is positive or whenever two or more MSSVs are inoperable. The Actions contained in the NUREG are not sufficient to provide assurance that RCS and Secondary System pressures will be maintained within acceptable limits in the event of a power increase or overshoot which could occur with a positive Moderator Temperature Coefficient or with more than one MSSV inoperable on one or more Steam Generators. Whenever these conditions exist, it is necessary to limit the primary system power to a value that does not result in exceeding the combined steam flow capacity of the remaining OPERABLE MSSVs. Reducing the Power Range Neutron Flux-High setpoint provides assurance that the reactor will be tripped, maintaining power within the flow capacity of the remaining operable MSSVs in the event of a power increase. If the reactor is not operating in excess of 5% power, the reduction in the Power Range Neutron Flux-High setpoint is not necessary as power increases and overshoots will not be significant enough to exceed the combined steam flow capacity of the remaining operable MSSVs. Corresponding terminology changes have been made to Tables 3.7.1-1 and 3.7.1-2 to facilitate use of the revised Actions proposed. This change is consistent with the generic change TSTF 235, revision 1.

ITS:	NUREG:
B 3.07.01	B 3.07.01
LCO 3.07.01 COND A	LCO 3.07.01 COND A
LCO 3.07.01 COND A RA A.1	LCO 3.07.01 COND A RA A.1
LCO 3.07.01 COND B	N/A
LCO 3.07.01 COND B RA B.1	N/A
LCO 3.07.01 COND B RA B.2	N/A
LCO 3.07.01 COND B RA B.2 NOTE	N/A
LCO 3.07.01 COND C	LCO 3.07.01 COND B
LCO 3.07.01 COND C RA C.1	LCO 3.07.01 COND B RA B.1
LCO 3.07.01 COND C RA C.2	LCO 3.07.01 COND B RA B.2
LCO 3.07.01 T 3.07.01-01	LCO 3.07.01 T 3.07.01-01

Justification For Deviations - NUREG-1431 Section 3.07.01

13-Nov-99

JFD Number	JFD Text
02	The number of MSSVs listed in Table 1 has been reduced by one to a total of four as Point Beach has only four safety valves per steam generator. Similarly, the number of S/Gs contained in Table 2 has been reduced to two as Point Beach has only two steam generators and the designations have been changed from 1 and 2 to A and B to conform with plant-specific identification of equipment. Site specific steam generator safety valve setpoints have also been added.

ITS:	NUREG:
B 3.07.01	B 3.07.01
LCO 3.07.01 T 3.07.01-01	LCO 3.07.01 T 3.07.01-01
LCO 3.07.01 T 3.07.01-02	LCO 3.07.01 T 3.07.01-02

JFD Number	JFD Text
03	NUREG Table 3.7.1-1 is used in conjunction with the Required Actions of LCO 3.7.1 to establish the maximum allowable power level and reactor trip setpoint reductions which may be required when one or more MSSVs are determined to be inoperable. These values are site specific and have been calculated in accordance with a conservative heat balance algorithm contained in NRC Information Notice 94-60.

ITS:	NUREG:
B 3.07.01	B 3.07.01
LCO 3.07.01 T 3.07.01-01	LCO 3.07.01 T 3.07.01-01

JFD Number	JFD Text
04	Brackets have been removed and the appropriate plant specific information has been inserted.

ITS:	NUREG:
B 3.07.01	B 3.07.01

JFD Number	JFD Text
05	The Bases for LCO 3.7.1 state that normal feedwater flow is terminated by the loss of load event. No consequential loss of main feedwater will occur as a result of this event; however, loss of main feedwater is modeled as a worst case assumption. As such, the Bases have been changed to reflect loss of main feedwater as an analysis assumption and not a consequence of the event.

ITS:	NUREG:
B 3.07.01	B 3.07.01

JFD Number	JFD Text
06	Reference has been changed from the 1987 version of ASME/ANSI OM-1 to the 1981 version to reflect the version of the code in affect for the third inspection interval at Point Beach. In accordance with this version of the code, periodic safety valve testing consists of setpoint verifications, with the additional testing listed in the Bases only required after refurbishment of the MSSVs. Accordingly, the Bases have been modified to reflect ASME/ANSI OM-1, 1981. **ITS:** B 3.07.01 **NUREG:** B 3.07.01
07	The reference to valve chattering in the bases discussion of the MSSV design function has been deleted. The function of staggered setpoints to reduce the potential of valve chattering in the MSSV's is not discussed in the PBNP FSAR and was therefore deemed to be inappropriate. **ITS:** B 3.07.01 **NUREG:** B 3.07.01

3.7 PLANT SYSTEMS

3.7.1 Main Steam Safety Valves (MSSVs)

LCO 3.7.1 The MSSVs shall be OPERABLE as specified in Table 3.7.1-1 and Table 3.7.1-2.

APPLICABILITY: MODES 1, 2, and 3.

ACTIONS

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

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. One or more required MSSVs inoperable.	A.1	Reduce power to less than or equal to the applicable % RTP listed in Table 3.7.1-1.	4 hours
B. Required Action and associated Completion Time not met. <u>OR</u> One or more steam generators with less than [two] MSSVs OPERABLE.	B.1 <u>AND</u> B.2	Be in MODE 3. Be in MODE 4.	6 hours 12 hours

Replace with Insert 3.7.1-1

1

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.7.1.1	-------------------NOTE------------------- Only required to be performed in MODES 1 and 2. -- Verify each required MSSV lift setpoint per Table 3.7.1-2 in accordance with the Inservice Testing Program. Following testing, lift setting shall be within ±1%.	In accordance with the Inservice Testing Program

[1]

Table 3.7.1-1 (page 1 of 1)
OPERABLE Main Steam Safety Valves versus
~~Applicable Power in Percent of RATED THERMAL POWER~~ Maximum Allowable Power

~~MINIMUM NUMBER OF MSSVs PER STEAM GENERATOR REQUIRED OPERABLE~~ NUMBER OF OPERABLE MSSVs PER STEAM GENERATOR	~~APPLICABLE POWER (% RTP)~~ MAXIMUM ALLOWABLE POWER (%RTP)
~~5~~	~~≤ 100~~ [2]
4	≤ ~~80~~ 100 [3]
3	≤ ~~60~~ 49 [3]
2	≤ ~~40~~ 29 [3]

Table 3.7.1-2 (page 1 of 1)
Main Steam Safety Valve Lift Settings

VALVE NUMBER				LIFT SETTING (psig ± [3]%)
STEAM GENERATOR				
~~#1~~ A	~~#2~~ B	~~[#3]~~	~~[#4]~~	
MS 2010	MS 2005			1085
MS 2011	MS 2006			1100
MS 2012	MS 2007			1125
MS 2013	MS 2008			1125

INSERT 3.7.1-1:

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more Steam Generators with one MSSV inoperable and Moderator Temperature Coefficient (MTC) zero or negative at all power levels.	A.1 Reduce THERMAL POWER to $\leq$ 49% RTP.	4 hours
B. One or more Steam Generators with two or more MSSV inoperable <u>OR</u> One or more Steam Generators with one MSSV inoperable and Moderator Temperature Coefficient (MTC) positive at any power level.	B.1 Reduce power to less than or equal to the Maximum Allowable % RTP specified in Table 3.7.1-1 for the number of OPERABLE MSSVs. <u>AND</u>	4 hours
	--------NOTE--------- Only required in MODE 1. --------------------- B.2 Reduce the Power Range Neutron Flux - High reactor trip setpoint to less than or equal to the Maximum Allowable % RTP specified in Table 3.7.1-1 for the number of OPERABLE MSSVs.	36 hours
C. Required Action and associated Completion Time not met. <u>OR</u> One or more steam generators with three or more MSSVs inoperable.	C.1 Be in MODE 3. <u>AND</u>	6 hours
	C.2 Be in MODE 4.	12 hours

B 3.7 PLANT SYSTEMS

B 3.7.1 Main Steam Safety Valves (MSSVs)

BASES

BACKGROUND

The primary purpose of the MSSVs is to provide overpressure protection for the secondary system. The MSSVs also provide protection against overpressurizing the reactor coolant pressure boundary (RCPB) by providing a heat sink for the removal of energy from the Reactor Coolant System (RCS) if the preferred heat sink, provided by the Condenser and Circulating Water System, is not available.

2 4 1

~~Five~~ Four MSSVs are located on each main steam header, outside containment, upstream of the main steam isolation valves, as described in the FSAR, Section ~~[10.3.1]~~ 10.1 (Ref. 1). The MSSVs ~~capacity criteria is 110% of rated steam flow at~~ must have sufficient capacity to limit the secondary system pressure to ≤ 110% of the steam generator design pressure. ~~This meets~~ to meet the requirements of the ASME Code, Section III (Ref. 2). The MSSV design includes staggered setpoints, according to Table 3.7.1-2 in the accompanying LCO, so that only the needed valves will actuate. ~~Staggered setpoints reduce the potential for valve chattering that is due to steam pressure insufficient to fully open all valves following a turbine reactor trip.~~

1 7 1

APPLICABLE SAFETY ANALYSES

The design basis for the MSSVs comes from Reference 2 and its purpose is to limit the secondary system pressure to ≤ 110% of design pressure ~~when passing 100% of design steam flow. This design basis is sufficient to cope with any anticipated operational occurrence (AOO) or accident~~ for considered in the Design Basis Accident (DBA) and transient analysis.

4

The events that challenge the relieving capacity of the MSSVs, and thus RCS pressure, are those characterized as decreased heat removal events, which are presented in the FSAR, Section ~~[15.2]~~ 14.1.9 (Ref. 3). Of these, the full power turbine trip without steam dump is the limiting AOO. ~~This event also terminates normal feedwater flow to the steam generators.~~

5

4

safety analysis demonstrates that the

occurring from full power

BASES

5

APPLICABLE SAFETY ANALYSES (continued)

and loss of normal feedwater

The transient response for turbine trip without a direct reactor trip presents no hazard to the integrity of the RCS or the Main Steam System. ~~If a minimum reactivity feedback is assumed, the reactor is tripped on high pressurizer pressure. In this case, the pressurizer safety valves open, and RCS pressure remains below 110% of the design value. The MSSVs also open to limit the secondary steam pressure.~~

1

Replace with Insert B 3.7.1-1

~~If maximum reactivity feedback is assumed, the reactor is tripped on overtemperature ΔT. The departure from nucleate boiling ratio increases throughout the transient, and never drops below its initial value. Pressurizer relief valves and MSSVs are activated and prevent overpressurization in the primary and secondary systems.~~ The MSSVs are assumed to have two active and one passive failure modes. The active failure modes are spurious opening, and failure to reclose once opened. The passive failure mode is failure to open upon demand.

The MSSVs satisfy Criterion 3 of the NRC Policy Statement.

LCO

2

four

1

and

The accident analysis requires four MSSVs per steam generator to provide overpressure protection for design basis transients occurring at 102% RTP. An MSSV will be considered inoperable if it fails to open on demand. The LCO requires that ~~five~~ MSSVs be OPERABLE in compliance with Reference 2~~, even though this is not a requirement of the DBA analysis.~~ This ~~is~~ because operation with less than the ~~full number of MSSVs requires limitations on allowable THERMAL POWER (to meet ASME Code requirements). These limitations are according to Table 3.7.1-1 in the accompanying LCO, and Required Action A.2.~~

1

required

The OPERABILITY of the MSSVs is defined as the ability to open within the ~~setpoint~~ tolerances, relieve steam generator overpressure, and reseat when pressure has been reduced. The OPERABILITY of the MSSVs is determined by periodic surveillance testing in accordance with the Inservice Testing Program.

BASES

1 In MODES 1, 2, and 3, four MSSVs per steam generator are required to be OPERABLE to prevent Main Steam System overpressurization.

LCO (continued)

The lift settings, according to Table 3.7.1-2 in the accompanying LCO, correspond to ambient conditions of the valve at nominal operating temperature and pressure.

This LCO provides assurance that the MSSVs will perform their designed safety functions to mitigate the consequences of accidents that could result in a challenge to the RCPB.

1 or Main Steam System integrity

APPLICABILITY

~~In MODE 1 above 40% RTP, the number of MSSVs per steam generator required to be OPERABLE must be according to Table 3.7.1-1 in the accompanying LCO. Below 40% RTP in MODES 1, 2, and 3, only two MSSVs per steam generator are required to be OPERABLE.~~

In MODES 4 and 5, there are no credible transients requiring the MSSVs. The steam generators are not normally used for heat removal in MODES 5 and 6, and thus cannot be overpressurized; there is no requirement for the MSSVs to be OPERABLE in these

ACTIONS

The ACTIONS table is modified by a Note indicating that separate Condition entry is allowed for each MSSV.

A.1

1 action must be taken

2 four

With one or more MSSVs inoperable, ~~reduce power~~ so that the available MSSV relieving capacity meets Reference 2 requirements for the applicable THERMAL POWER.

Operation with less than all ~~five~~ MSSVs OPERABLE for each steam generator is permissible, if THERMAL POWER is

1 ~~proportionally~~ limited to the relief capacity of the remaining MSSVs. This is accomplished by restricting THERMAL POWER so that the energy transfer to the most limiting steam generator is not greater than the available relief capacity in that steam generator. ~~For example, if one MSSV is inoperable in one steam generator, the relief capacity of that steam generator is reduced by approximately 20%. To offset this reduction in relief capacity, energy transfer to that steam generator must be similarly reduced~~

1 Replace with Insert B 3.7.1-2

BASES

ACTIONS (continued)

1

Replace with Insert B 3.7.1-2

~~by at least 20%. This is accomplished by reducing THERMAL POWER by at least 20%, which conservatively limits the energy transfer to all steam generators to approximately 80% of total capacity, consistent with the relief capacity of the most limiting steam generator.~~

~~For each steam generator, at a specified pressure, the fractional relief capacity (FRC) of each MSSV is determined as follows:~~

$$FRC = \frac{A}{B} \quad 1$$

~~where:~~

~~A = the relief capacity of the MSSV; and~~

~~B = the total relief capacity of all the MSSVs of the steam generator~~

~~The FRC is the relief capacity necessary to address operation with reduced THERMAL POWER.~~

~~The reduced THERMAL POWER levels in the LCO prevent operation at power levels greater than the relief capacity of the remaining MSSVs. The reduced THERMAL POWER is determined as follows:~~

$$RP = \left[1 - (N_1 \times FRC_1 + N_2 \times FRC_2 + \ldots + N_5 \times FRC_5)\right] \times 100\% \quad 2$$

~~where:~~

~~RP = Reduced THERMAL POWER for the most limiting steam generator expressed as a percent of RTP;~~

~~N_1, N_2, ..., N_5 represent the status of the MSSV 1, 2, ..., 5, respectively.~~

~~= 0 if the MSSV is OPERABLE,~~
~~= 1 if the MSSV is inoperable;~~

~~FRC_1, FRC_2, ..., FRC_5 = the relief capacity of the MSSV 1, 2, ..., 5, respectively, as defined above.~~

BASES

ACTIONS (continued)

1

~~B~~ C.1 and ~~B~~ C.2

If the MSSVs cannot be restored to OPERABLE status within the associated Completion Time, or if one or more steam generators have ~~less than two MSSVs OPERABLE,~~ three or more inoperable MSSVs the unit must be placed in a MODE in which the LCO does not apply. To achieve this status, the unit must be placed in at least MODE 3 within 6 hours, and in MODE 4 within 12 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required unit conditions from full power conditions in an orderly manner and without challenging unit systems.

SURVEILLANCE REQUIREMENTS

SR 3.7.1.1

This SR verifies the OPERABILITY of the MSSVs by the verification of each MSSV lift setpoint in accordance with the Inservice Testing Program. The ASME Code, Section XI (Ref. 4), requires that safety and relief valve tests be performed in accordance with ANSI/ASME OM-1-~~1987~~ 1981 (Ref. 5). According to Reference 5, in addition to routine lift setpoint verifications, the following tests are required following equipment refurbishment:

6

a. Visual examination;

b. Seat tightness determination;

c. Setpoint pressure determination (lift setting);

d. Compliance with owner's seat tightness criteria; and

e. Verification of the balancing device integrity on balanced valves.

The ANSI/ASME Standard requires that all valves be tested every 5 years, and a minimum of 20% of the valves be tested every 24 months. The ASME Code specifies the activities and frequencies necessary to satisfy the requirements. Table 3.7.1-2 allows a ± [3]% setpoint tolerance for OPERABILITY; however, the valves are reset to ± 1% during the Surveillance to allow for drift.

5

BASES

SURVEILLANCE REQUIREMENTS (continued)

This SR is modified by a Note that allows entry into and operation in MODE 3 prior to performing the SR. The MSSVs may be either bench tested or tested in situ at hot conditions using an assist device to simulate lift pressure. If the MSSVs are not tested at hot conditions, the lift setting pressure shall be corrected to ambient conditions of the valve at operating temperature and pressure. 2

REFERENCES 2

1. FSAR, Section ~~[10.3.1]~~ 10.1
2. ASME, Boiler and Pressure Vessel Code, Section III, Article NC-7000, Class 2 Components.
3. FSAR, Section ~~[15.2]~~. 14.1.9
4. ASME, Boiler and Pressure Vessel Code, Section XI.
5. ANSI/ASME OM-1-~~[1987]~~ 1981 6
6. NRC Information Notice 94-60, "Potential Overpressurization of the Main Steam System," August 22, 1994. 1

LCO 3.7.1 BASES INSERTS

Insert B 3.7.1-1:

In Chapter 14 of the FSAR, one case of loss of electrical load analysis is performed assuming primary system pressure control via operation of the pressurizer power-operated relief valves and spray. This case demonstrates that the DNB Design Basis is met. Another analysis is performed assuming no primary system pressure control, reactor trip on high pressurizer pressure and operation of the pressurizer safety valves. This analysis demonstrates that RCS integrity is maintained by showing that the maximum RCS pressure does not exceed 110% of the design pressure. All cases analyzed demonstrate that the MSSVs maintain Main Steam System integrity by limiting the maximum steam pressure to less than 110% of the steam generator design pressure.

In addition to the decreased heat removal events, reactivity insertion events may also challenge the relieving capacity of the MSSVs. The uncontrolled rod cluster control assembly (RCCA) bank withdrawal at power event is characterized by an increase in core power and steam generation rate until reactor trip occurs when either the Overtemperature ΔT or Power Range Neutron Flux-High setpoint is reached. Steam flow to the turbine will not increase from its initial value for this event. The increased heat transfer to the secondary side causes an increase in steam pressure and may result in opening of the MSSVs prior to reactor trip, assuming no credit for operation of the atmospheric or condenser steam dump valves. The FSAR Section 14.1.2 safety analysis of the RCCA bank withdrawal at power event for a range of initial core power levels demonstrates that the MSSVs are capable of preventing secondary side overpressurization for this AOO.

The FSAR safety analyses discussed above assume that all of the MSSVs for each steam generator are OPERABLE. If there are inoperable MSSV(s), it is necessary to limit the primary system power during steady-state operation and AOOs to a value that does not result in exceeding the combined steam flow capacity of the remaining OPERABLE MSSVs. The required limitation on primary system power necessary to prevent secondary system overpressurization have been determined by conservative heat balance calculations. In some circumstances it is necessary to limit the primary side heat generation that can be achieved during an AOO by reducing the setpoint of the Power Range Neutron Flux-High reactor trip function. For example, if more than one MSSV on a single steam generator is inoperable, an uncontrolled RCCA bank withdrawal at power event occurring from a partial power level may result in an increase in reactor power that exceeds the combined steam flow capacity of the remaining OPERABLE MSSVs. Thus, for multiple inoperable MSSVs on the same steam generator it is necessary to prevent this power increase by lowering the Power Range Neutron Flux-High setpoint to an appropriate value. When the Moderator Temperature Coefficient (MTC) is positive, the reactor power may increase above the initial value during an RCS heatup event (e.g., turbine trip). Thus, for any number of inoperable MSSVs it is necessary to reduce the trip setpoint if a positive MTC may exist at partial power conditions.

Insert B 3.7.1-2:

A.1

In the case of a single inoperable MSSV on one or more steam generators, when the Moderator Temperature Coefficient is not positive, a reactor power reduction alone is sufficient to limit primary side heat generation to preclude overpressurization of the secondary side during any RCS heatup event. There is sufficient total steam flow capacity provided by the remaining OPERABLE MSSVs to preclude overpressurization in the event of an increase in reactor power due to reactivity insertion, such as in the event of an uncontrolled RCCA bank withdrawal at power. Therefore, Required Action A.1 requires an appropriate reduction in reactor power within 4 hours.

The maximum THERMAL POWER corresponding to the heat removal capacity of the remaining OPERABLE MSSVs is determined via a conservative heat balance calculation as described in Attachment 1 to Reference 6, with an appropriate allowance for instrument and channel uncertainties.

B.1 and B.2

In the case of multiple inoperable MSSVs on one or more steam generators, a reactor power reduction alone may be insufficient to limit steam production to within the total steam flow capacity provided by the remaining OPERABLE MSSVs. In the case of a single inoperable MSSV on one or more steam generators when the Moderator Temperature Coefficient is positive, the reactor power may increase as a result of an RCS heatup event such that flow capacity of the remaining OPERABLE MSSVs is insufficient.

The 4 hour Completion Time for Required Action B.1 is consistent with A.1. An additional 32 hours is allowed in Required Action B.2 to reduce the setpoints. The completion Time of 36 hours is based on a reasonable time to correct the MSSV inoperability, the time required to perform the power reduction, operating experience in resetting all channels of a protective function, and on the low probability of the occurrence of a transient that could result in steam generator overpressure during this period.

The maximum THERMAL POWER corresponding to the heat removal capacity of the remaining OPERABLE MSSVs is determined via a conservative heat balance calculation as described in the Attachment to Reference 6, with an appropriate allowance for Nuclear Instrumentation System trip channel uncertainties.

Required Action B.2 is modified by a Note, indicating that the Power Range Neutron Flux-High reactor trip setpoint reduction is only required in MODE 1. In MODES 2 and 3 the reactor protection system trips specified in LCO 3.3.1, "Reactor Trip System Instrumentation" provide sufficient protection.

The allowed Completion Times are reasonable based on operating experience to accomplish the Required Actions in an orderly manner without challenging unit systems.

No Significant Hazards Considerations - NUREG-1431 Section 3.07.01

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	The CTS does not provide any specific Actions which address the inoperability of the MSSVs, which result in entry into CTS 15.3.0.b whenever an MSSV is determined to be inoperable. Entry into CTS 15.3.0.b requires the unit to be placed into Hot Shutdown (ITS Mode 3) within 7 hours at which time the CTS Applicability is exited and no further Technical Specification Actions are required. The ITS provide specific Conditions and Required Actions to address the inoperability of MSSVs based on the number of inoperable valves and whether or not the Moderator Temperature Coefficient is positive, negative, or zero. The CTS does not specify any remedial or restoration actions for inoperable MSSVs. Accordingly, an inoperable MSSV results in entry into Specification 15.3.0.B, which would require the plant to be placed into Hot Shutdown within 7 hours. NUREG 1431 provides actions for inoperable MSSVs based on the number of inoperable valves and whether or not the Moderator Temperature Coefficient is positive, negative, or zero. If there are inoperable MSSV(s), it is necessary to limit the primary system power during steady state operation to a value that does not result in exceeding the combined steam flow capacity of the remaining OPERABLE MSSVs. This reduction is necessary to prevent primary and secondary system overpressurization and has been calculated in accordance with the conservative heat balance calculations provided in NRC Information Notice 94-60 which references Westinghouse NSAL 94-001. If the Moderator Temperature Coefficient is zero or negative, a power reduction alone is sufficient for a single inoperable MSSV on one or both Steam Generators. If the Moderator Temperature Coefficient is positive or if two or more MSSVs are inoperable on any Steam Generator, the power reduction must be accompanied by a similar reduction in the Power Range Neutron Flux-High setpoint. Reducing the Power Range Neutron Flux-High setpoint provides assurance that the reactor power will remain within the flow capacity of the remaining operable MSSVs in the event of a power increase or overshoot. If the reactor is not operating in excess of 5% power, the reduction in the Power Range Neutron Flux-High setpoint is not necessary as power increases and overshoots will not be significant enough to exceed the combined steam flow capacity of the remaining operable MSSVs. In accordance with the criteria set forth in 10 CFR 50.92, PBNP has evaluated this proposed Technical Specifications change and determined it does not represent a significant hazards consideration. The following is provided in support of this conclusion. 1. Does the change involve a significant increase in the probability or consequences of an accident previously evaluated? This change does not result in any hardware changes, which therefore does not result in any significant alteration to any previously evaluated accident precursors. The proposed Actions are sufficiently conservative to assure that previously evaluated acceptance limits will continue to be met. Therefore, the proposed 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

NSHC Number	NSHC Text
	accident previously evaluated? The proposed change does not require a physical alteration of the plant (no new or different type of equipment will be installed) or changes in parameters governing normal plant operation. The proposed change will allow continued operation with inoperable MSSVs, but at a reduced power level, with protective system setbacks as required to assure that the current acceptance limits are met. 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 Required Actions will establish sufficient margins in operation, such that the current acceptance limits for analyzed event will be preserved. In preserving these acceptance limits, the margin of safety is not significantly affected.
LB	In accordance with the criteria set forth in 10 CFR 50.92, PBNP has evaluated this proposed Technical Specifications change and determined it does not represent a significant hazards consideration. The following is provided in support of this conclusion. 1. Does the change involve a significant increase in the probability or consequences of an accident previously evaluated? This change involves deletion of a Specifications/information which is duplicative of information contained in the Code of Federal Regulations (CFRs). This information is more appropriately addressed by the CFRs and serves no purpose in the Technical Specifications. Deletion of this information will not result in an increase in the probability of an accident. Regulatory requirements do not alter plant design or configuration; therefore, this does not alter any event precursor. Accordingly, there will be no effect on the consequences of any accident. 2. Does the change create the possibility of a new or different kind of accident from any accident previously evaluated? The proposed change does not require a physical alteration of the plant (no new or different type of equipment will be installed) or changes in parameters governing normal plant operation. The proposed change deletes materials from the Technical Specifications which are adequately addressed in the CFRs. Thus, this change does not create the possibility of a new or different kind of accident from any accident previously evaluated. 3. Does this change involve a significant reduction in a margin of safety? The proposed change deletes materials from the Technical Specifications which are duplicative of requirements contained in the CFRs. These items are not an input to any accident analysis and, therefore, have no impact on margin of safety.

NSHC Number	NSHC Text
M	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.

3.7 PLANT SYSTEMS

3.7.1 Main Steam Safety Valves (MSSVs)

LCO 3.7.1 The MSSVs shall be OPERABLE as specified in Table 3.7.1-1 and Table 3.7.1-2.

APPLICABILITY: MODES 1, 2, and 3.

ACTIONS

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

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more Steam Generators with one MSSV inoperable and Moderator Temperature Coefficient (MTC) zero or negative at all power levels.	A.1 Reduce THERMAL POWER to ≤ 49% RTP.	4 hours
B. One or more Steam Generators with two or more MSSVs inoperable. OR One or more Steam Generators with one MSSV inoperable and Moderator Temperature Coefficient (MTC) positive at any power level.	B.1 Reduce power to less than or equal to the Maximum Allowable % RTP specified in Table 3.7.1-1 for the number of OPERABLE MSSVs. AND	4 hours (continued)

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
B. (continued)	--------NOTE--------- Only required in MODE 1. --------------------- B.2 Reduce the Power Range Neutron Flux - High reactor trip setpoint to less than or equal to the Maximum Allowable % RTP specified in Table 3.7.1-1 for the number of OPERABLE MSSVs.	 36 hours
C. Required Action and associated Completion Time not met. OR One or more steam generators with three or more MSSVs inoperable.	C.1 Be in MODE 3. AND C.2 Be in MODE 4.	6 hours 12 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.7.1.1	------------------NOTE------------------- Only required to be performed in MODES 1 and 2. --- Verify each required MSSV lift setpoint per Table 3.7.1-2 in accordance with the Inservice Testing Program. Following testing, lift setting shall be within ±1%.	In accordance with the Inservice Testing Program.

Table 3.7.1-1 (page 1 of 1)
OPERABLE Main Steam Safety Valves versus
Maximum Allowable Power

NUMBER OF OPERABLE MSSVs PER STEAM GENERATOR	MAXIMUM ALLOWABLE POWER (% RTP)
4	≤ 100
3	≤ 49
2	≤ 29

Table 3.7.1-2 (page 1 of 1)
Main Steam Safety Valve Lift Settings

VALVE NUMBER		LIFT SETTING (psig ± 3%)
STEAM GENERATOR		
A	B	
MS 2010	MS 2005	1085
MS 2011	MS 2006	1100
MS 2012	MS 2007	1125
MS 2013	MS 2008	1125

B 3.7 PLANT SYSTEMS

B 3.7.1 Main Steam Safety Valves (MSSVs)

BASES

BACKGROUND

The primary purpose of the MSSVs is to provide overpres sure protection for the secondary system. The MSSVs also provide protection against overpressurizing the reactor coolant pressure boundary (RCPB) by providing a heat sink for the removal of energy from the Reactor Coolant System (RCS) if the preferred heat sink, provided by the Condenser and Circulating Water System, is not available.

Four MSSVs are located on each main steam header, outside containment, upstream of the main steam isolation valves, as described in the FSAR, Section 10.1 (Ref. 1). The MSSVs must have sufficient capacity to limit the secondary system pressure to ≤ 110% of the steam generator design pressure to meet the requirements of the ASME Code, Section III (Ref. 2). The MSSV design includes staggered set points, according to Table 3.7.1-2 in the accompanying LCO, so that only the needed valves will actuate.

APPLICABLE SAFETY ANALYSES

The design basis for the MSSVs comes from Reference 2 and its purpose is to limit the secondary system pressure to ≤ 110% of design pressure for any anticipated operational occurrence (AOO) or accident considered in the Design Basis Accident (DBA) and transient analysis.

The events that challenge the relieving capacity of the MSSVs, and thus RCS pressure, are those characterized as decreased heat removal events, which are presented in the FSAR, Section 14.1.9 (Ref. 3). Of these, the full power turbine trip without steam dump is the limiting AOO.

The safety analysis demonstrates that the transient response for turbine trip occurring from full power without a direct reactor trip and loss of normal feedwater presents no hazard to the integrity of the RCS or the Main Steam System. In Chapter 14 of the FSAR, one case of loss of electrical load analysis is performed assuming primary system pressure

BASES

APPLICABLE SAFETY ANALYSES (continued)

control via operation of the pressurizer power-operated relief valves and spray. This analysis demonstrates that the DNB Design Basis is met. Another case is performed assuming no primary system pressure control, reactor trip on high pressurizer pressure and operation of the pressurizer safety valves. This analysis demonstrates that RCS integrity is maintained by showing that the maximum RCS pressure does not exceed 110% of the design pressure. All cases analyzed demonstrate that the MSSVs maintain Main Steam System integrity by limiting the maximum steam pressure to less than 110% of the steam generator design pressure.

In addition to the decreased heat removal events, reactivity insertion events may also challenge the relieving capacity of the MSSVs. The uncontrolled rod cluster control assembly (RCCA) bank withdrawal at power event is characterized by an increase in core power and steam generation rate until reactor trip occurs when either the Overtemperature ΔT or Power Range Neutron Flux-High setpoint is reached. Steam flow to the turbine will not increase from its initial value for this event. The increased heat transfer to the secondary side causes an increase in steam pressure and may result in opening of the MSSVs prior to reactor trip, assuming no credit for operation of the atmospheric or condenser steam dump valves. The FSAR Section 14.1.2 safety analysis of the RCCA bank withdrawal at power event for a range of initial core power levels demonstrates that the MSSVs are capable of preventing secondary side overpressurization for this AOO.

The FSAR safety analyses discussed above assume that all of the MSSVs for each steam generator are OPERABLE. If there are inoperable MSSV(s), it is necessary to limit the primary system power during steady-state operation and AOOs to a value that does not result in exceeding the combined steam flow capacity of the remaining OPERABLE MSSVs. The required limitation on primary system power necessary to prevent

BASES

APPLICABLE SAFETY ANALYSES (continued)

secondary system overpressurization have been determined by conservative heat balance calculations. In some circumstances it is necessary to limit the primary side heat generation that can be achieved during an AOO by reducing the setpoint of the Power Range Neutron Flux-High reactor trip function. For example, if more than one MSSV on a single steam generator is inoperable, an uncontrolled RCCA bank withdrawal at power event occurring from a partial power level may result in an increase in reactor power that exceeds the combined steam flow capacity of the remaining OPERABLE MSSVs. Thus, for multiple inoperable MSSVs on the same steam generator it is necessary to prevent this power increase by lowering the Power Range Neutron Flux-High setpoint to an appropriate value. When the Moderator Temperature Coefficient (MTC) is positive, the reactor power may increase above the initial value during an RCS heatup event (e.g., turbine trip). Thus, for any number of inoperable MSSVs it is necessary to reduce the trip setpoint if a positive MTC may exist at partial power conditions.

The MSSVs are assumed to have two active and one passive failure modes. The active failure modes are spurious opening, and failure to re-close once opened. The passive failure mode is failure to open upon demand.

The MSSVs satisfy Criterion 3 of the NRC Policy Statement.

LCO

The accident analysis requires four MSSVs per steam generator to provide overpressure protection for design basis transients occurring at 102% RTP. An MSSV will be considered inoperable if it fails to open on demand. The LCO requires that four MSSVs be OPERABLE in compliance with Reference 2 and the DBA analysis.

The OPERABILITY of the MSSVs is defined as the ability to open within the required tolerances, relieve steam generator overpressure, and reseat when pressure has been reduced. The OPERABILITY of the MSSVs is determined by periodic

BASES

LCO (continued)

surveillance testing in accordance with the Inservice Testing Program.

The lift settings, according to Table 3 .7.1-2 in the accompanying LCO, correspond to ambient conditions of the valve at nominal operating temperature and pressure.

This LCO provides assurance that the MSSVs will perform their designed safety functions to mitigate the consequences of accidents that could result in a challenge to the RCPB or Main Steam System integrity.

APPLICABILITY

In MODES 1, 2, and 3, four MSS Vs per steam generator are required to be OPERABLE to prevent Main Steam System overpressurization.

In MODES 4 and 5, there are no credible transients requiring the MSSVs. The steam generators are not normally used for heat removal in MODES 5 and 6, and thus cannot be overpressurized; there is no requirement for the MSSVs to be OPERABLE in these MODES.

ACTIONS

The ACTIONS table is modified by a Note in dicating that separate Condition entry is allowed for each MSSV.

A.1

With one or more MSSVs inoperable, action must be taken so that the available MSSV relieving capacity meets Reference 2 requirements for the applicable THERMAL POWER.

Operation with less than all four MSSVs OPERABLE for each steam generator is permissible, if THERMAL POWER is limited to the relief capacity of the remaining MSSVs. This is accomplished by restricting THERMAL POWER so that the energy transfer to the most limiting steam generator is not greater than the available relief capacity in that steam generator.

In the case of a single inoperable MSSV on one or more steam generators when the Moderator Temperature Coefficient is not

ACTIONS (continued)

positive, a reactor power reduction alone is sufficient to limit primary side heat generation to preclude overpressurization of the secondary side during any RCS heatup event. There is sufficient total steam flow capacity provided by the remaining OPERABLE MSSVs to preclude overpressurization in the event of an increase in reactor power due to reactivity insertion, such as in the event of an uncontrolled RCCA bank withdrawal at power. Therefore, Required Action A.1 requires an appropriate reduction in reactor power within 4 hours.

The maximum THERMAL POWER corresponding to the heat removal capacity of the remaining OPERABLE MSSVs is determined via a conservative heat balance calculation as described in Attachment 1 to Reference 6, with an appropriate allowance for instrument and channel uncertainties.

B.1 and B.2

In the case of multiple inoperable MSSVs on one or more steam generators, a reactor power reduction alone may be insufficient to limit steam production to within the total steam flow capacity provided by the remaining OPERABLE MSSVs. In the case of a single inoperable MSSV on one or more steam generators when the Moderator Temperature Coefficient is positive, the reactor power may increase as a result of an RCS heatup event such that flow capacity of the remaining OPERABLE MSSVs is insufficient.

The 4 hour Completion Time for Required Action B.1 is consistent with A.1. An additional 32 hours is allowed in Required Action B.2 to reduce the setpoints. The completion Time of 36 hours is based on a reasonable time to correct the MSSV inoperability, the time required to perform the power reduction, operating experience in resetting all channels of a protective function, and on the low probability of the occurrence of a transient that could result in steam generator overpressure during this period.

The maximum THERMAL POWER corresponding to the heat removal capacity of the remaining OPERABLE MSSVs is determined via a conservative heat balance calculation as described in the Attachment to Reference 6, with an appropriate allowance for Nuclear Instrumentation System trip channel uncertainties.

BASES

ACTIONS (continued)

Required Action B.2 is modified by a Note, indicating that the Power Range Neutron Flux-High reactor trip setpoint reduction is only required in MODE 1. In MODES 2 and 3 the reactor protection system trips specified in LCO 3.3.1, "Reactor Trip System Instrumentation" provide sufficient protection.

The allowed Completion Times are reasonable based on operating experience to accomplish the Required Actions in an orderly manner without challenging unit systems.

C.1 and C.2

If the MSSVs cannot be restored to OPERABLE status within the associated Completion Time, or if one or more steam generators have three or more inoperable MSSVs, the unit must be placed in a MODE in which the LCO does not apply. To achieve this status, the unit must be placed in at least MODE 3 within 6 hours, and in MODE 4 within 12 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required unit conditions from full power conditions in an orderly manner and without challenging unit systems.

SURVEILLANCE REQUIREMENTS

SR 3.7.1.1

This SR verifies the OPERABILITY of the MSSVs by the verification of each MSSV lift setpoint in accordance with the Inservice Testing Program. The ASME Code, Section XI (Ref. 4), requires that safety and relief valve tests be performed in accordance with ANSI/ASME OM-1-1981 (Ref. 5). According to Reference 5, in addition to routine lift setpoint verifications, the following tests are required following equipment refurbishment:

a. Visual examination;

b. Seat tightness determination;

c. Setpoint pressure determination (lift setting);

BASES

SURVEILLANCE REQUIREMENTS (continued)

d. Compliance with owner's seat tightness criteria; and

e. Verification of the balancing device integrity on balanced valves.

The ANSI/ASME Standard requires that all valves be tested every 5 years, and a minimum of 20% of the valves be tested every 24 months. The ASME Code specifies the activities and frequencies necessary to satisfy the requirements. Table 3.7.1-2 allows a ± 3% setpoint tolerance for OPERABILITY; however, the valves are reset to ± 1% during the Surveillance to allow for drift.

This SR is modified by a Note that allows entry into and operation in MODE 3 prior to performing the SR. The MSSVs may be either bench tested or tested in situ at hot conditions using an assist device to simulate lift pressure. If the MSSVs are not tested at hot conditions, the lift setting pressure shall be corrected to ambient conditions of the valve at operating temperature and pressure.

REFERENCES

1. FSAR, Section 10.1.

2. ASME, Boiler and Pressure Vessel Code, Section III, Article NC-7000, Class 2 Components.

3. FSAR, Section 14.1.9.

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

5. ANSI/ASME OM-1-1981.

6. NRC Information Notice 94-60, "Potential Overpressurization of the Main Steam System," August 22, 1994.

Cross-Reference Report - NUREG-1431 Section 3.07.02
ITS to CTS

13-Nov-99

ITS	CTS	DOC
B 3.07.02	15.03.04 OBJ	A.03
	BASES	A.07
IST	15.04.07.B	LB.02
LCO 3.07.02	15.03.04 APPL	A.02
	15.03.04.D	A.01
	15.03.04.D	A.04
	15.03.04.D	M.01
	15.04.07	A.01
	15.04.07 APPL	A.02
	15.04.07 OBJ	A.03
LCO 3.07.02 COND A	15.03.04.D	L.01
LCO 3.07.02 COND A RA A.1	15.03.04.D	L.01
LCO 3.07.02 COND B	15.03.04.D	A.01
LCO 3.07.02 COND B RA B.1	15.03.04.D	L.01
LCO 3.07.02 COND C	15.03.04.D	A.01
LCO 3.07.02 COND C NOTE	15.03.04.D	A.01
LCO 3.07.02 COND C RA C NOTE	15.03.04.D	M.02
LCO 3.07.02 COND C RA C.1	NEW	M.04
LCO 3.07.02 COND C RA C.2	15.03.04.D	M.04
LCO 3.07.02 COND C RA C.3	NEW	M.05
LCO 3.07.02 COND D	15.03.04.D	L.01
LCO 3.07.02 COND D RA D.1	15.03.04.D	L.01
LCO 3.07.02 COND D RA D.2	NEW	M.01
SR 3.07.02.01	15.04.07.A	A.01
	15.04.07.A	LA.01
	15.04.07.A	LB.01
SR 3.07.02.01 NOTE	15.04.07.A	A.06
SR 3.07.02.02	15.04.01 T 15.04.01-02 13	M.03

Cross-Reference Report - NUREG-1431 Section 3.07.02

CTS to ITS

13-Nov-99

CTS	ITS	DOC
15.03.04 APPL	LCO 3.07.02	A.02
15.03.04 OBJ	B 3.07.02	A.03
15.03.04.D	DELETED	A.05
	LCO 3.07.02	A.01
	LCO 3.07.02	A.04
	LCO 3.07.02	M.01
	LCO 3.07.02 COND A	L.01
	LCO 3.07.02 COND A RA A.1	L.01
	LCO 3.07.02 COND B	A.01
	LCO 3.07.02 COND B RA B.1	L.01
	LCO 3.07.02 COND C	A.01
	LCO 3.07.02 COND C NOTE	A.01
	LCO 3.07.02 COND C RA C NOTE	M.02
	LCO 3.07.02 COND C RA C.2	M.04
	LCO 3.07.02 COND D	L.01
	LCO 3.07.02 COND D RA D.1	L.01
15.04.01 T 15.04.01-02 13	SR 3.07.02.02	M.03
15.04.07	LCO 3.07.02	A.01
15.04.07 APPL	LCO 3.07.02	A.02
15.04.07 OBJ	LCO 3.07.02	A.03
15.04.07.A	DELETED	LA.01
	SR 3.07.02.01	A.01
	SR 3.07.02.01	LA.01
	SR 3.07.02.01	LB.01
	SR 3.07.02.01 NOTE	A.06
15.04.07.B	IST	LB.02
BASES	B 3.07.02	A.07

Description of Changes - NUREG-1431 Section 3.07.02

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:
15.03.04.D	LCO 3.07.02
	LCO 3.07.02 COND B
	LCO 3.07.02 COND C
	LCO 3.07.02 COND C NOTE
15.04.07	LCO 3.07.02
15.04.07.A	SR 3.07.02.01

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

CTS:	ITS:
15.03.04 APPL	LCO 3.07.02
15.04.07 APPL	LCO 3.07.02

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

CTS:	ITS:
15.03.04 OBJ	B 3.07.02
15.04.07 OBJ	LCO 3.07.02

DOC Number	DOC Text
A.04	The CTS states that the main steam stop and check valves (MS 2017, 2018, 2017A and 2018A) are required to be operable. This requirement is equivalent to ITS LCO 3.7.2, which requires two MSIVs and two non-return check valves to be operable. Specifying the noun name for these valves is sufficient to establish the regulatory requirement for maintaining these valves operable when required. There are no other valves contained within the main steam system which may be used to perform the required safety functions. This change is administrative. **CTS:** 15.03.04.D **ITS:** LCO 3.07.02
A.05	The CTS allows the main steam stop and non-return check valves to be opened in the hot shutdown condition to perform testing to confirm operability of these valves if the valves were previously closed in accordance with the CTS Actions. This allowance is duplicative of ITS LCO 3.0.5 which allows equipment removed from service or declared inoperable to be returned to service to perform testing required to demonstrate its operability. Based on ITS LCO 3.0.5 providing this allowance generically, removal of this component specific statement is administrative. **CTS:** 15.03.04.D **ITS:** DELETED
A.06	CTS specifies that closure timing of the MSIVs is to be performed under low flow conditions of 5% steam flow or less. The conditions under which this test is to be performed are discussed in description of change LA.1 of this section. However, the CTS requirement to perform this test prior to exceeding 5% steam flow is equivalent to the Note contained in ITS SR 3.7.2.1 requiring MSIV stroke timing to be completed prior to entering ITS Mode 1 (greater than 5% power). The CTS closure time limit of five seconds has been incorporated into SR 3.7.2.1. As such, this change is administrative. **CTS:** 15.04.07.A **ITS:** SR 3.07.02.01 NOTE
A.07	The Bases of the current Technical Specifications for this section have been completely replaced by revised Bases that reflect the format and applicable content of PBNP ITS, consistent with the Standard Technical Specifications for Westinghouse Plants, NUREG-1431. The revised Bases are as shown in the PBNP ITS Bases. **CTS:** BASES **ITS:** B 3.07.02

DOC Number	DOC Text
L.01	CTS allows four hours to restore one inoperable MSIV or non-return check valve to operable status during power operation (ITS Modes 1 and 2). If the inoperable valve is not restored to operable status with this four hour period, the CTS requires the unit to be placed into hot shutdown (ITS Mode 3) within the following 6 hours.

The ITS will allow an MSIV and non-return check valve to be inoperable simultaneously on the same steam generator for up to eight hours before requiring the unit to be placed into Mode 2. After entry into Mode 2, an additional eight hours is allowed to close and deactivate the MSIV and close the non-return check valve in the affected flowpath. If the valve is closed, indefinite operation in Mode 2 (less than 5% power) is allowed; however, if the valve cannot be closed, the unit is to be placed into Mode 3 within six hours and Mode 4 within 12 hours. As such, the ITS will allow multiple valves to be inoperable, continued operation below 5% power with isolated inoperable valves, and will ultimately extend the time allowed to reach Mode 3 from ten to twenty-four hours.

Allowing multiple valves to be inoperable simultaneously on the same steam generator is considered acceptable, as this condition does not result in an unanalyzed situation, but rather the inability to sustain a single failure of the other steam generator's MSIV and non-return check valve. The condition of multiple valves inoperable in the same flowpath is equivalent to a single MSIV inoperable as described in NUREG 1431.

Continued operation in Mode 2 with the affected flowpath isolated is acceptable, as the valves are required to be placed in the accident position, thereby fulfilling their required safety function.

Extending the limit allowed to reach Mode 3 is considered acceptable based on redundant capability of the opposite steam generator's MSIV and check valve, the passive nature of the steam generator as a boundary, and the low probability of an accident occurring during this time period that would require a closure of the MSIVs or non-return check valves.

CTS:	ITS:
15.03.04.D	LCO 3.07.02 COND A
	LCO 3.07.02 COND A RA A.1
	LCO 3.07.02 COND B RA B.1
	LCO 3.07.02 COND D
	LCO 3.07.02 COND D RA D.1

DOC Number	DOC Text
LA.01	CTS requires the main steam stop valves to be tested under low flow conditions, with reactor thermal power not to exceed five percent, in addition to specifying the method for timing valve stroke. These items are details which are not necessary to describe the actual regulatory requirement (performance of valve stroke timing). This information has been moved to plant procedures. This information provides details of processes which are not directly pertinent to the actual requirement, but rather describe an acceptable method of compliance. These details are not necessary to provide adequate protection of the public health and safety since the ITS still retains the requirement to perform the test. Changes to the testing conditions and methods will be controlled in accordance with the licensee's procedure revision process. Therefore, the level of safety is unaffected by the change. **CTS:** 15.04.07.A **ITS:** DELETED SR 3.07.02.01
LB.01	The CTS requires the main steam stop valves to be tested following plant shutdowns for major fuel reloadings. The main steam stop and non-return check valves are ASME Class 2 valves and as such are required to be tested on a frequency consistent with ASME Section XI, ASME/ANSI OM-1, 1981 as endorsed and required under 10 CFR 50.55a. Accordingly, testing frequency for these valves is established and required by regulation without the need to duplicate these requirements in the Technical Specifications. In fact, under the current PBNP IST Program, the main steam stop and non-return check valves are required to be tested on a cold shutdown frequency, which is more restrictive than the CTS. **CTS:** 15.04.07.A **ITS:** SR 3.07.02.01
LB.02	The CTS requires the main steam non-return check valves to be tested for operability during plant shutdowns for major fuel reloadings. The main steam non-return check valves are ASME Class 2 valves and as such are required to be tested in accordance with the criteria, methods, and frequency of testing established in ASME Section XI, ASME/ANSI OM-1, 1981 as endorsed and required under 10 CFR 50.55a. Accordingly, testing of these valves is required by regulation without the need to duplicate this requirement in the Technical Specifications. In fact, under the current PBNP IST Program, the main steam stop and non-return check valves are required to be tested on a cold shutdown frequency, which is more restrictive than the CTS. **CTS:** 15.04.07.B **ITS:** IST

Description of Changes - NUREG-1431 Section 3.07.02

13-Nov-99

DOC Number	DOC Text
M.01	The CTS requires the MSIVs and non-return check valves to be operable, but does not provide an explicit Mode of Applicability. If the MSIVs or non-return check valves are inoperable, the CTS will allow continued operation in hot shutdown providing that the valves are maintained closed. The CTS definition of Hot Shut Down requires the reactor to be greater than or equal to 540 degrees. Based on a Technical Specification structure which exits the Mode of Applicability for LCO non-compliance, the CTS applicability would be anytime the reactor coolant temperature is greater than or equal to 540 degrees. The ITS Mode of Applicability for this LCO has been proposed to be Mode 1, 2, and 3. Default Conditions and Required Actions have also been added to require the unit to be placed into Mode 3 within 6 hours and Mode 4 within 12 hours if the MSIVs or non-return check valves are not isolated in accordance with the proposed Actions. The MSIVs and non-return check valves must be operable in Modes 1, 2, and 3 as these are the Modes in which operation of these valves is necessary in the mitigation of DBAs. In Mode 4, steam generator energy is low and isolation is not necessary for DBA mitigation. In Modes 5 and 6, the MSIVs and non-return check valves are not required for isolation of secondary system pipe breaks, or mitigation of RCS cooldown events.

CTS:	ITS:
15.03.04.D	LCO 3.07.02
NEW	LCO 3.07.02 COND D RA D.2

DOC Number	DOC Text
M.02	The CTS allows an inoperable MSIV or non-return check valve to be opened in the hot shutdown condition to allow cooldown of the affected unit. This allowance is necessary to allow steam to be vented to the condenser from both steam generators, promoting uniform and simultaneous cooldown of both steam generators. The proposed ITS retains this allowance, while establishing a requirement to have administrative controls for closure of the valve(s). The addition of administrative controls is a more restrictive requirement than the CTS which will provide assurance that the valve(s) can be closed if necessary.

CTS:	ITS:
15.03.04.D	LCO 3.07.02 COND C RA C NOTE

DOC Number	DOC Text
M.03	CTS requires containment isolation valves (inclusive of the MSIVs) to be functionally tested each refueling shutdown, which the CTS defines as a shutdown to move fuel to and from the reactor core. The ITS SR 3.7.2.2 will require each MSIV to be actuated to its isolation position on an actual or simulated action signal once every 18 months. These tests are intended to ensure that MSIVs actuate to their required position upon receipt of an isolation signal. Accordingly, the CTS and the ITS require the same testing; however, the CTS does not define a specific frequency of performance for this surveillance. The CTS test interval is considered to be a plant evolution, which can vary significantly from outage to outage with no bounding limit. Changes in cycle lengths by default establish the required frequency. As such, the adoption of a bounding frequency (18 months) is a more restrictive change.

CTS:	ITS:
15.04.01 T 15.04.01-02 13	SR 3.07.02.02

DOC Number	DOC Text
M.04	The CTS allows operation to continue in hot shutdown with an inoperable MSIV or non-return check valve provided that the inoperable valve is closed. The proposed ITS will allow continued operation with an inoperable MSIV or non-return check valve as well, as outlined in Description of Change L.1, and M.2 of this LCO; however, the ITS will also require the MSIV in the affected flowpath to be closed and de-activated and the non-return check valve in the affected flowpath to be in the closed position. The MSIVs at Point Beach are check valves which close to inhibit forward flow. Forward flow through the MSIV is allowed by the check valve disk being held out of the flow steam by an air operator, which fails safe upon receipt of an actuation signal allowing the valve to close. Reverse flow to the Steam Generators from the Main Steam Lines (MSL) is prevented by the non-return check valves which are simple check valves. Accordingly, the MSL isolation function is accomplished through the use of two valves. Requiring the MSIV to be closed and deactivated in addition to closing the non-return check valve is intended to prevent either valve from being inadvertently opened due to changes in steam header or steam generator pressure. The proposed eight hour Completion Time for valve closure and deactivation is reasonable, considering the time required to isolate the flowpath and de-activate the MSIV. **CTS:** 15.03.04.D — **ITS:** LCO 3.07.02 COND C RA C.2 **CTS:** NEW — **ITS:** LCO 3.07.02 COND C RA C.1
M.05	The CTS allows continued operation in hot shutdown (ITS Mode 3) with an inoperable MSIV or non-return check valve providing the valve is closed, but the CTS does not specify a completion time for closure of the inoperable valve. The ITS will require that the inoperable valve be isolated within eight hours, in addition to establishing a requirement to verify that the MSIV and/or non-return check valve is closed once every seven days. The eight hour Completion Times for valve closure is reasonable, considering the time required to isolate the penetration. The 7 day Completion Time is reasonable, based on engineering judgment, in view of MSIV status indications available in the control room, and administrative controls to ensure that these valves are maintained in the closed position. **CTS:** NEW — **ITS:** LCO 3.07.02 COND C RA C.3

15.3.4 STEAM AND POWER CONVERSION SYSTEM

Applicability

A.2

Applies to the operating status of steam and power conversion system.

Objective

A.3

~~To define conditions of the steam and power conversion system steam-relieving capacity. Auxiliary Feedwater System and Service Water System operation is necessary to ensure the capability to remove decay heat from the core.~~

Specification

< See LCO 3.7.1, 3.7.4, 3.7.5 and 3.7.6 >

A. When the reactor coolant is heated above 350°F the reactor shall not be taken critical unless the following conditions are met:

1. A minimum steam-relieving capability of eight (8) main steam safety valves available, except for low power physics testing.

2. Auxiliary Feedwater System < See LCO 3.7.5 >

   a. Two Unit Operation - All four auxiliary feedwater pumps together with their associated flow paths and essential instrumentation shall be operable.

   b. Single Unit Operation - Both motor driven auxiliary feedwater pumps and the turbine driven auxiliary feedwater pump associated with that unit together with their associated flow paths and essential instrumentation shall be operable.

A.1

< See LCO 3.7.5 >

2. Single Unit Operation - One of the three operable auxiliary feedwater pumps associated with a unit may be out-of-service for the below specified times. The turbine driven auxiliary feedwater pump may be out-of-service for up to 72 hours. If the turbine driven auxiliary feedwater pump cannot be restored to service within that 72 hour time period, the reactor shall be in hot shutdown within the next 12 hours. Either one of the two motor driven auxiliary feedwater pumps may be out-of-service for up to 7 days. If the motor driven auxiliary feedwater pump cannot be restored to service within that 7 day period the operating unit shall be in hot shutdown within the next 12 hours.

See Insert 3.7.2-1

A.4/ M.1

L.1/ M.1/ M.2/ M.4/ M.5

See Insert 3.7.2-2

D. The main steam stop valves (MS-2017 and MS-2018) and the non-return check valves (MS-2017A and MS-2018A) shall be operable. If one main steam stop valve or non-return check valve is inoperable but open, power operation may continue provided the inoperable valve is restored to operable status within 4 hours, otherwise the reactor shall be placed in a hot shutdown condition within the following 6 hours. With one or more main steam stop valves or non-return check valves inoperable, subsequent operation in the hot shutdown condition may proceed provided the inoperable valve or valves are maintained closed. An inoperable main steam stop valve or non-return check valve may however, be opened in the hot shutdown condition to cool down the affected unit and to perform ~~testing to confirm operability.~~

A.5

Basis

A reactor shutdown from power requires removal of core decay heat. Immediate decay heat removal requirements are normally satisfied by the steam bypass to the condenser. Therefore, core decay heat can be continuously dissipated via the steam bypass to the condenser as feedwater in the steam generator is converted to steam by heat absorption. Normally, the capability to return feedwater flow to the steam generators is provided by operation of the turbine cycle feedwater system.

< See LCO 3.7.1 >

TABLE 15.4.1-2 (Continued)

		Test	Frequency
7.	Spent Fuel Pit < See LCO 3.7.15/16 >	a) Boron Concentration b) Water Level Verification	Monthly Weekly
8.	Secondary Coolant < See LCO 3.7.18 >	Gross Beta-gamma Activity or gamma isotopic analysis	Weekly (6)
		Iodine concentration	Weekly when gross Beta-gamma activity equals or exceeds 1.0 mCi/g (6)
9.	Control Rods < See Section 3.1 >	a) Rod drop times of all full length rods (3)	Each refueling or after maintenance that could affect proper functioning (4)
		b) Rodworth measurement	Following each refueling shutdown prior to commencing power operation
10.	Control Rod	Partial movement of all rods	Every 2 weeks (18)
11.	Pressurizer Safety Valves	< See Section 3.4 >	Every five years (11)
12.	Main Steam Safety Valves	< See LCO 3.7.1 >	Every five years (11)
13.	Containment Isolation Trip	Functioning	Each refueling shutdown
14.	Refueling System Interlocks	< See Section 3.9 >	Each refueling shutdown
15.	Service Water System	< See LCO 3.7.8 >	Each refueling shutdown
16.	Primary System Leakage	< See Section 3.4 >	Monthly (6)
17.	Diesel Fuel Supply	< See Section 3.8 >	Daily
18.	Deleted		
19.	Deleted		
20.	Boric Acid System < See Section 3.5 >	Storage Tank and piping temperatures ≥ temperature required by Table 15.3.2-1	Daily (19)

SR 3.7.2.2
See Insert 3.7.2-3

M.3

15.4.7 MAIN STEAM SYSTEM VALVES

A.2

Applicability

Applies to periodic testing and surveillance of the main steam stop valves (MS-2017 and MS-2018 and the non-return check valves (MS-2017A and MS-2018A).

A.3

~~Objective~~

~~To verify the ability of the main steam stop valves to close upon signal and to verify that the non-return check valves are operable.~~

SR 3.7.2.1 and Note
See Insert 3.7.2-4

A.6

LA.1

Specification

A. Main Steam Stop Valves

The main steam stop valves ~~shall be tested under low flow conditions of 5% steam flow or less~~ following plant shutdowns for major fuel reloading. ~~The test shall be performed during the plant startup prior to admitting steam to the turbine.~~ Closure time of five seconds or less shall be verified. ~~The five seconds shall be measured from the time of signal initiation until the valve indicates closed.~~

LB.1

LA.1

In accordance with the Inservice Testing Program

B. Non-Return Check Valves

~~The non-return check valves shall be tested for operability during shutdown for major fuel reloadings.~~

LB.2

~~Basis~~

~~The main steam stop valves serve to limit an excessive reactor coolant system cooldown rate and resultant reactivity insertion following a main steam break incident. Their ability to close upon signal should be verified at each scheduled refueling shutdown. A closure time of five seconds was selected as being consistent with the expected response time for instrumentation as detailed in the steam line break incident analysis. The test procedure need not require steam to be flowing in the pipe. The purpose of the non-return check valves is to prevent the blowdown of both steam generators in the event of a main steam line piping break upstream of the main steam stop valves. The non-return check valves are swinging disc check valves which are opened by normal steam flow.~~

A.7

During no-flow conditions the non-return check valves are shut. The position of the non-return check valves, and thus the ability of the valves to close and perform their safety function, can be verified locally when no steam flow conditions are established.

References

FSAR - Section 10.4
FSAR - Section 14.2.5

A.7

LCO 3.7.2 Inserts

Insert 3.7.2-1:

A.4

LCO 3.7.2	Two MSIVs and two non-return check valves shall be operable.

APPLICABILITY: MODES 1, 2, and 3

LCO 3.7.2 Inserts

Insert 3.7.2-2:

A. One Steam Generator flowpath with one or more inoperable valves.	A.1 Restore valve to OPERABLE status.	8 hours
B. Required Action and associated Completion Time of Condition A not met.	B.1 Be in MODE 2.	6 hours
C. ---------NOTE--------- Separate Condition entry is allowed for each MSIV and non-return check valve. --------------------- One or both MSIVs inoperable in MODE 2 or 3.	------------NOTE-------------- An inoperable flowpath may be opened under administrative controls to allow cool down of the affected unit. ------------------------------ C.1 Close and de-activate the MSIV in the affected flowpath. AND C.2 Close non-return check valve in the affected flowpath. AND C.3 Verify valves are closed.	8 hours 8 hours Once per 7 days
D. Required Action and associated Completion Time of Condition C not met.	D.1 Be in MODE 3. AND D.2 Be in MODE 4.	6 hours 12 hours

L.1

M.2

M.4

M.5

L.1

M.1

LCO 3.7.2 Inserts

Insert 3.7.2-3:

SURVEILLANCE		FREQUENCY
SR 3.7.2.2	Verify each MSIV actuates to the isolation position on an actual or simulated actuation signal.	18 months

M.3

Insert 3.7.2-4:

SR 3.7.2.1	-------------------NOTE------------------- Only required to be performed in MODES 1. -- Verify closure time of each MSIV is ≤ 5.0 seconds.	In accordance with the Inservice Testing Program

A.6

Justification For Deviations - NUREG-1431 Section 3.07.02

13-Nov-99

JFD Number	JFD Text
01	NUREG 1431 LCO 3.7.2 has been modified to reflect Point Beach's design. The MSIV LCO was written to address an MSIV which inhibits both forward and reverse flow. The MSIVs at Point Beach are check valves which close to inhibit forward flow. Forward flow through the MSIV is allowed by the check valve disk being held out of the flow steam by an air operator which fails safe upon receipt of an actuation signal allowing the valve to close. Reverse flow to the Steam Generators from the Main Steam Lines (MSLs) is prevented through the use of a simple check valve referred to as the MSL "non-return check valves". Accordingly, the MSL isolation function is accomplished through two valves, requiring modification of the LCO, Required Actions, Bases, and Surveillance Requirements to reflect the Point Beach Design Basis. The LCO Title has been modified to reflect both the MSIV and the non-return check valves. Condition A of NUREG 1431 LCO 3.7.2 has been modified to reflect the Point Beach equivalent to having an MSIV inoperable. This equivalent condition would be the inoperability of one or more valves (MSIV and non-return check valve) in the same SG flowpath. Eight hours has been retained as the restoration time for this Condition consistent with NUREG 1431. Condition C has been modified to address the Required Actions for inoperable MSIVs and non-return check valves in Modes 2 or 3. These Conditions are equivalent to Condition C of NUREG 1431 (inoperable MSIV in Mode 2 and 3); however, based on Point Beach's design, it is necessary to close both the MSIV and the non-return check valve in the affected flow path in order to provide isolation. Closure of both valves is necessary to prevent inadvertent opening of the inoperable valve due to differential pressure gradients that may develop due to heatups, cooldowns, or changes in steam demand. Eight hours has been retained for flowpath isolation and seven days for routine verification of isolation consistent with NUREG 1431. The Bases have been revised to reflect Point Beach's design and revised Conditions and Required Actions as discussed above.

ITS:	NUREG:
B 3.07.02	B 3.07.02
	B 3.07.02
LCO 3.07.02	LCO 3.07.02
	LCO 3.07.02
LCO 3.07.02 COND A	LCO 3.07.02 COND A
LCO 3.07.02 COND A RA A.1	LCO 3.07.02 COND A RA A.1
LCO 3.07.02 COND C	LCO 3.07.02 COND C
LCO 3.07.02 COND C NOTE	LCO 3.07.02 COND C NOTE
LCO 3.07.02 COND C RA C.1	LCO 3.07.02 COND C RA C.1
LCO 3.07.02 COND C RA C.2	N/A

JFD Number	JFD Text	
	LCO 3.07.02 COND C RA C.3	LCO 3.07.02 COND C RA C.2
02	The brackets have been removed and the proper plant specific information has been provided.	
	ITS:	**NUREG:**
	B 3.07.02	B 3.07.02
	LCO 3.07.02	LCO 3.07.02
	SR 3.07.02.01	SR 3.07.02.01
03	The CTS allows an inoperable MSIV or non-return check valve to be opened in the hot shutdown condition to allow cooldown of the affected unit. This CTS allowance has been retained as a Note associated with the Required Actions for these valves. This allowance is necessary to allow steam to be vented to the condenser from both steam generators, promoting uniform and simultaneous cooldown of both steam generators. The proposed ITS retains this allowance, while establishing a requirement to have administrative controls over these valves if opened.	
	ITS:	**NUREG:**
	B 3.07.02	B 3.07.02
	LCO 3.07.02 COND C RA C NOTE	N/A
04	The Applicability of NUREG 1431 LCO 3.7.2 has been modified based on Point Beach's MSIV and non-return check valve design. Deenergization of the MSIV will not isolate the MSIV flowpaths based on the MSIV and non-return check valve design as described in the Justification for Deviation 1 of this Section. The Applicability has been changed to establish entry into this LCO whenever sufficient energy is contained within the Steam Generators to require MSIV and non-return check valve isolation capability in the event of a Main Steam Line Break. This Applicability is consistent with the accident analysis assumptions for Point Beach.	
	ITS:	**NUREG:**
	B 3.07.02	B 3.07.02
	LCO 3.07.02	LCO 3.07.02
05	The Applicability section of the Bases has been reworded consistent with Point Beach having only two Steam Generators.	
	ITS:	**NUREG:**
	B 3.07.02	B 3.07.02

JFD Number	JFD Text
06	The Bases for Condition B contains a discussion related to closing the MSIV. Closure of the MSIV is performed in Condition C and is discussed within the Bases for the Required Actions associated with that Condition. Accordingly, the discussion contained in the Bases for Condition B has been deleted.

ITS:	NUREG:
B 3.07.02	B 3.07.02

JFD Number	JFD Text
07	NUREG SR 3.7.2.1 has been divided into two separate Surveillance Requirements. ITS SR 3.7.2.1 verifies the MSIV closure time while proposed ITS SR 3.7.2.2 verifies that the MSIVs will actuate on a simulated or actual actuation signal. This presentation is necessary to promote consistent application of the testing requirements in addition to deferring performance of MSIV stroke timing until prior to entry into Mode 1 as allowed by the CTS and discussed below.

Proposed ITS SR 3.7.2.1 and SR 3.7.2.2 are equivalent to CTS Surveillance Requirement 15.4.7.A, which requires the MSIVs to be stroke tested under low flow conditions (less than or equal to 5%) and CTS line item 13 of Table 15.4.1-2, which requires containment isolation valves (MSIVs) to be functionally tested. The CTS Applicability for containment isolation valves has been determined to be equivalent to ITS Modes 1 through 4 as discussed in LCO 3.6.3 of this conversion package. As such, functional testing of the MSIVs isolation capability is required prior to entry into Mode 4 under ITS LCO 3.6.3 (containment isolation) and prior to entry into ITS Mode 3 (ITS SR 3.7.2.2) under this LCO; however, stroke timing of the MSIVs (ITS SR 3.7.2.1) is not required until prior to exceeding 5% power. Deferred performance of the MSIV stroke timing is necessary to establish appropriate and representative testing conditions for the MSIVs, as discussed in Justification for Deviation 9 of this Section.

Additionally, the 18 month actuation test (SR 3.7.2.2) is intended to provide a continuation between the actuation logic testing contained in Section 3.3 of the ITS and the actuated components (MSIVs). NUREG 1431 requires Actuation Logic and Master and Slave Relay tests to be performed with the unit on line (bi-monthly and quarterly). These tests, when combined with the 18 month equipment actuation tests, prove equipment actuation capability from the channel output to the actuated equipment. Point Beach has not adopted the Surveillance Requirements for Master and Slave Relay testing based on design and licensing basis. Point Beach is not designed to allow on line testing without introducing unwarranted transients or intrusive testing techniques. Accordingly, Master and Slave testing has not been adopted as part of the conversion to the ITS. The 18 month actuation test encompasses Master and Slave Relay testing.

This change is consistent with proposed generic change TSTF 289.

ITS:	NUREG:
B 3.07.02	B 3.07.02
SR 3.07.02.01	SR 3.07.02.01
SR 3.07.02.02	N/A

13-Nov-99

JFD Number	JFD Text	
08	A discussion has been added to the Actions section, which addresses the MSIVs as being containment isolation valves. This discussion has been added to reinforce that the applicable Conditions and Required Actions of LCO 3.6.3 should also be entered if the MSIV is inoperable in such a fashion that its containment isolation capability is also impaired.	
	ITS:	**NUREG:**
	B 3.07.02	B 3.07.02
09	CTS 15.4.7.a requires the MSIVs to be stroke time tested under low flow conditions not to exceed 5% of steam flow, which has been determined to be equivalent to a required mode of performance for this surveillance of prior to entry into ITS Mode 1. The MSIVs at Point Beach are check valves which close to inhibit forward flow. Forward flow through the MSIV is allowed by the check valve disk being held out of the flow steam by an air operator which fails safe upon receipt of an actuation signal allowing the valve to close. As such, steam flow assists in closing the valve within its required Stoke time, requiring deferment in performance of this SR to establish conditions which are representative of the conditions under which the acceptance criteria was developed. This deviation from the NUREG is consistent with the CTS for Point Beach.	
	ITS:	**NUREG:**
	B 3.07.02	B 3.07.02
	SR 3.07.02.01 NOTE	SR 3.07.02.01 NOTE
10	NUREG 1431 provides an option of testing the MSIV per the Inservice Testing Program (IST) or once per 18 months. The option of testing these valves in accordance with the IST has been chosen. The MSIVs are Class 2 valves and are contained within the IST. Selection of this option is further discussed in Description of Change LB.1 of this LCO.	
	ITS:	**NUREG:**
	B 3.07.02	B 3.07.02
	SR 3.07.02.01	SR 3.07.02.01
11	The current licensing basis for Point Beach does not include feedwater line break scenarios. Accordingly, reference to Feedwater line break events in the Bases of the proposed ITS have been deleted	
	ITS:	**NUREG:**
	B 3.07.02	B 3.07.02

JFD Number	JFD Text
12	The Bases have been revised to list the MSIV isolation signals for Point Beach. This change is necessary to reflect Point Beach's design and licensing basis. **ITS:** B 3.07.02 — **NUREG:** B 3.07.02
13	The NUREG Bases provide a description of automatic power operated MSIV bypass valves. Point Beach's MSIV bypass valves are manual valves. Accordingly, the Bases have been modified to reflect Point Beach's design. **ITS:** B 3.07.02 — **NUREG:** B 3.07.02
14	The NUREG Bases have been modified to reflect the containment pressure and off site dose analyses reflective of Point Beach's current licensing basis. **ITS:** B 3.07.02 — **NUREG:** B 3.07.02
15	The Containment pressure analysis and radiological consequences for Steam Line Break event are both contained in the same section of Point Beach's FSAR. Accordingly, reference to separate sections of the FSAR are not necessary, reference numbers have been revised to reflect the appropriate FSAR Section and reference. **ITS:** B 3.07.02 — **NUREG:** B 3.07.02

and Non-Return Check Valves

1

3.7 PLANT SYSTEMS

3.7.2 Main Steam Isolation Valves (MSIVs)

2 Two

and two non-return check valves

LCO 3.7.2 ~~[Four]~~ MSIVs shall be OPERABLE.

4

APPLICABILITY: MODE 1.
MODES 2 and 3 ~~except when all MSIVs are closed and [de-activated]~~.

1

Replace with Insert 3.7.1-1

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
~~A. One MSIV inoperable in MODE 1.~~	~~A.1 Restore MSIV to OPERABLE status.~~	~~[8] hours~~
B. Required Action and associated Completion Time of Condition A not met.	B.1 Be in MODE 2.	6 hours
~~C. ---------NOTE--------- Separate Condition entry is allowed for each MSIV. --------------------- One or more MSIVs inoperable in MODE 2 or 3.~~	~~C.1 Close MSIV. AND C.2 Verify MSIV is closed.~~	~~[8] hours Once per 7 days~~

(continued)

See Insert 3.7.2-2

1

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
D. Required Action and associated Completion Time of Condition C not met.	D.1 Be in MODE 3. <u>AND</u> D.2 Be in MODE 4.	6 hours 12 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
SR 3.7.2.1 [9] -------------------NOTE------------------- Only required to be performed in MODES 1 ~~and 2~~ --- [2] Verify closure time of each MSIV is ≤ [4.6] 5.0 seconds ~~on an actual or simulated actuation signal~~. [7]	In accordance with the Inservice Testing Program ~~or [18] months]~~ [10]

SURVEILLANCE	FREQUENCY
SR 3.7.2.2 Verify each MSIV actuates to the isolation position on an actual or simulated actuation signal.	18 months

7

LCO 3.7.2 Inserts

INSERT 3.7.1-1:

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One Steam Generator flowpath with one or more inoperable valves.	A.1 Restore valve to OPERABLE status.	8 hours

INSERT 3.7.1-2:

C. ---------NOTE--------- Separate Condition entry is allowed for each MSIV and each non-return check valve. ---------------------	------------NOTE-------------- An inoperable flowpath may be opened under administrative controls to allow cool down of the affected unit. ------------------------------	3
One or both MSIVs inoperable in MODE 2 or 3.	C.1 Close and de-activate the MSIV in the affected flowpath.	8 hours
<u>OR</u>	<u>AND</u>	
One or both non-return check valves inoperable in MODE 2 or 3.	C.2 Close non-return check valve in the affected flowpath.	8 hours
	<u>AND</u>	
	C.3 Verify valves are closed and de-activated.	Once per 7 days

and Non-Return Check Valves

1

B 3.7 PLANT SYSTEMS

B 3.7.2 Main Steam Isolation Valves (MSIVs)

BASES

BACKGROUND

The MSIVs isolate steam flow from the secondary side of the steam generators following a ~~high energy line break (HELB). MSIV closure terminates flow from the unaffected (intact) steam generators.~~

11 Insert B 3.7.2-1

One MSIV is located in each main steam line outside, but close to containment. The MSIVs are downstream from the main steam safety valves (MSSVs) and auxiliary feedwater (AFW) pump turbine steam supply, to prevent MSSV and AFW isolation from the steam generators by MSIV closure. ~~Closing the MSIVs isolates each steam generator from the others, and isolates the turbine, Steam Bypass System, and other auxiliary steam supplies from the steam generators.~~

1 Insert B 3.7.2-2

The MSIVs close on a main steam isolation signal generated by ~~either low steam generator pressure or high containment pressure. The MSIVs fail closed on loss of control or actuation power.~~

12 Insert B 3.7.2-3

~~Each MSIV has an MSIV bypass valve. Although these bypass valves are normally closed, they receive the same emergency closure signal as do their associated MSIVs. The MSIVs may also be actuated manually.~~

13 Insert B 3.7.2-4

A description of the MSIVs is found in the FSAR, Section ~~[10.3]~~ (Ref. 1) .

2 10.1

and Non-Return Check Valves

2 14.2.5

APPLICABLE SAFETY ANALYSES

The design basis of the MSIVs is established by the containment analysis for ~~the~~ ~~large~~ steam line break (SLB) ~~inside containment~~ discussed in the FSAR, Section [6.2] (Ref. 2). ~~It is also affected by the accident analysis of the SLB events presented in the FSAR, Section [15.1.5] (Ref. 3).~~ The design precludes the blowdown of more than one steam generator, assuming a ~~single~~ active component failure (e.g., the failure of one MSIV to close on demand).

1

or Non-Return Check Valves

1

APPLICABLE SAFETY ANALYSES (continued)

14

Insert
B 3.7.2-5

~~The limiting case for the containment analysis is the SLB inside containment, with a loss of offsite power following turbine trip, and failure of the MSIV on the affect ed steam generator to close. At lower powers, the steam generator inventory and temperature are at their maximum, maximizing the analyzed mass and energy release to the containment. Due to reverse flow and failure of the MSIV to close, the additional mass and energy in the steam headers downstream from the other MSIV contribute to the total release. With the most reactive rod cluster control assembly assumed stuck in the fully withdrawn position, there is an increased possibility that the core will become critical and return to power. The core is ultimately shut down by the boric acid injection delivered by the Emergency Core Cooling System.~~

~~The accident analysis compares seve ral different SLB events against different acceptance criteria. The large SLB outside containment upstream of the MSIV is limiting for offsite dose, although a break in this short section of main steam header has a very low probability. The large SLB inside containment at hot zero power is the limiting case for a post trip return to power. The analysis includes scenarios with offsite power available, and with a loss of offsite power following turbine trip. With offsite power available, the reactor coolant pumps continue to circulate coolant through the steam generators, maximizing the Reactor Coolant System cooldown. With a loss of offsite power, the response of mitigating systems is delayed. Significant single failures considered include failure of an MSIV to close.~~

~~The MSIVs serve only a safety function and remain open during power operation. These valves operate under the following situations:~~

~~a. An HELB inside containment. In order to maximize the mass and energy release into containment, the analysis assumes that the MSIV in the affected steam generator remains open. For this accident scenario, steam is discharged into containment from all steam generators until the remaining MSIVs close. After MSIV closure, steam is discharged into containment only from the affected steam generator and from the residual steam in the main steam header downstream of the closed MSIVs in~~

BASES

and Non-Return Check Valves

1

APPLICABLE SAFETY ANALYSES (continued)

~~the unaffected loops. Closure of the MSIVs isolates the break from the unaffected steam generators.~~

14

Insert B 3.7.2-5

~~b. A break outside of containment and upstream from the MSIVs is not a containment pressurization concern. The uncontrolled blowdown of more than one steam generator must be prevented to limit the potential for uncontrolled RCS cooldown and positive reactivity addition. Closure of the MSIVs isolates the break and limits the blowdown to a single steam generator.~~

~~c. A break downstream of the MSIVs will be isolated by the closure of the MSIVs.~~

~~d. Following a steam generator tube rupture, closure of the MSIVs isolates the ruptured steam generator from the intact steam generators to minimize radiological releases.~~

11

~~e. The MSIVs are also utilized during other events such as a feedwater line break. This event is less limiting so far as MSIV OPERABILITY is concerned.~~

The MSIVs satisfy Criterion 3 of the NRC Policy Statement.

two MSIVs and two non-return check valves

1

are to

1

LCO

This LCO requires that ~~[four]~~ MSIVs in the steam lines be OPERABLE. The MSIVs are considered OPERABLE when the isolation times are within limits, and they close on an isolation actuation signal.

Insert B 3.7.2-6

This LCO provides assurance that the MSIVs will perform their design safety function to mitigate the consequences of accidents that could result in offsite exposures comparable to the 10 CFR 100 (Ref. ~~4~~) limits ~~or the NRC staff approved licensing basis~~.

15

3

14

1

and non-return check valves

APPLICABILITY

The MSIVs must be OPERABLE in MODES 1, ~~and in MODES~~ 2 and 3 ~~except when closed and de-activated~~ when there is significant mass and energy in the RCS and steam generators.

4

BASES

APPLICABILITY (continued)

4 ~~When the MSIVs are closed, they are already performing the safety function.~~

5 In MODE 4, normally ~~most of~~ the MSIVs and non-return check valves are closed, and the steam generator energy is low.

1

In MODE 5 or 6, the steam generators do not contain much energy because their temperature is below the boiling point of water; therefore, the MSIVs or non-return check valves are not required for isolation of potential high energy secondary system pipe breaks in these MODES.

ACTIONS

1

A.1

With one or more valves in a SG flowpath inoperable in MODE 1, action must be taken to restore the flowpath to OPERABLE status within [8] hours. Some repairs to the MSIV can be made with the unit hot. The [8] hour Completion Time is reasonable, considering the low probability of an accident occurring during this time period that would require a closure of the MSIVs or non-return check valves.

2

Insert B 3.7.2-7

8

The [8] hour Completion Time is greater than that normally allowed for containment isolation valves because the MSIVs are valves that isolate a closed system penetrating containment. These valves differ from other containment isolation valves in that the closed system provides an additional means for containment isolation.

B.1

2 If the MSIV or Non-Return Check Valve cannot be restored to OPERABLE status within [8] hours, the unit must be placed in a MODE in which the LCO does not apply. To achieve this status, the unit must be placed in MODE 2 within 6 hours and Condition C would be entered. The Completion Times are reasonable, based on operating experience, to reach MODE 2 ~~and to close the MSIVs in an orderly manner and~~ without challenging unit systems.

6

C.1 and C.2

Condition C is modified by a Note indicating that separate Condition entry is allowed for each MSIV or Non-Return Check Valve.

1

BASES

and Non-Return Check Valves 1

ACTIONS (continued) 3

Insert B 3.7.2-8

Since the MSIVs are required to be OPERABLE in MODES 2 and 3, the inoperable MSIVs may either be restored to OPERABLE status or closed. When closed, the MSIVs are already in the position required by the assumptions in the safety analysis.

1 and de-activated Insert B 3.7.2-9 1

2 The [8] hour Completion Time is consistent with that allowed in Condition A.

or non-return check valves 1

For inoperable MSIVs that cannot be restored to OPERABLE status within the specified Completion Time, but are ~~closed,~~ the ~~inoperable MSIVs~~ must be verified on a periodic basis to be closed. This is necessary to ensure that the assumptions in the safety analysis remain valid. The 7 day Completion Time is reasonable, based on engineering judgment, in view of ~~MSIV status~~ indications available in the control room, and other administrative controls, to ensure that these valves are in the closed position.

flowpath 1 isolated

D.1 and D.2

or non-return check valves 1

If the MSIVs cannot be restored to OPERABLE status or are not closed within the associated Completion Time, the unit must be placed in a MODE in which the LCO does not apply. To achieve this status, the unit must be placed at least in MODE 3 within 6 hours, and in MODE 4 within 12 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required unit conditions from MODE 2 conditions in an orderly manner and without challenging unit systems.

SURVEILLANCE REQUIREMENTS

SR 3.7.2.1

5.0 2

This SR verifies that MSIV closure time is ≤ [4.6] seconds ~~on an actual or simulated actuation signal. The MSIV closure time is assumed in the accident and containment analyses.~~ This Surveillance is normally performed upon returning the unit to operation following a refueling outage. The MSIVs should not be tested at power, since even a part stroke exercise increases the risk of a valve closure

6

1

BASES

SURVEILLANCE REQUIREMENTS (continued)

when the unit is generating power. As the MSIVs are not tested at power, they are exempt from the ASME Code, Section XI (Ref. ~~5~~ 4 15), requirements during operation in MODE 1 9 ~~or 2~~.

10 The Frequency is in accordance with the [7] Inservice Testing Program ~~or [18] months~~. ~~The [18] month Frequency for valve closure time is based on the refueling cycle.~~ Operating experience has shown that these components usually pass the Surveillance when performed at the ~~[18] month~~ Frequency required by the Inservice Testing Program. Therefore, the Frequency is acceptable from a reliability standpoint.

This test is conducted in MODE ~~3 with the unit~~ 2 under low steam flow conditions at operating temperature and pressure~~, as discussed in Reference 5 exercising requirements~~. This SR is modified by a Note that allows entry into and operation in MODE~~3~~s 2 and 3 prior to performing the SR. This allows a delay of testing ~~until MODE 3,~~ to establish conditions consistent with those under which the acceptance criterion was generated.

Insert B 3.7.2-10

7

REFERENCES

1. FSAR, Section ~~[10.3]~~ 10.1 2
2. FSAR, Section ~~[6.2]~~ 14.2.5 2
3. ~~FSAR, Section [15.1.5].~~
4. 10 CFR 100.11. 15
5. ASME, Boiler and Pressure Vessel Code, Section XI.

NUREG-0800, Standard Review Plan 15.1.5, Appendix A, "Radiological Consequence of Main Steam Line Failures Outside of a PWR", Rev. 2, July 1981.

14

LCO 3.7.2 BASES INSERTS

Insert B 3.7.2-1:

steam line break. In addition, the MSIVs are used to isolate the affected steam generator in the event of a steam generator tube rupture.

Insert B 3.7.2-2:

The MSIVs isolate the turbine, Condenser Steam Dump System, and other auxiliary steam supplies (with the exception of the turbine driven auxiliary feedwater pump) from the steam generators. The MSIVs in conjunction with the non-return check valves, isolate the steam generators from each other.

Insert B 3.7.2-3:

Containment Pressure High-High, Steam Flow High-High coincident with a Safety Injection, or Steam Flow High coincident with Low Tavg and a Safety Injection. The MSIVs may also be manually actuated.

Insert B 3.7.2-4:

Each MSIV has a normally closed bypass valve.

Insert B 3.7.2-5:

The SLB containment pressure calculation is a parameter by parameter comparison of a reference 2-loop plant to Point Beach. Each parameter is evaluated to determine if the Point Beach value is conservative, non-conservative or nominal. The effects of the non-conservative parameters are quantified using a conservative heat balance to determine how much they increase peak containment pressure. Non-conservative parameters quantified in the calculation include additional FW and AFW, higher initial containment pressure, longer fan cooler delay time and lower fan cooler heat removal rates. The effect of one conservative parameter, containment heat sink surface area, is also quantified to determine how much it decreases peak containment pressure. Quantified increases and decreases are added to and subtracted from the most limiting result from the reference 2-loop plant analysis. Another conservative parameter is the trip reactivity worth for PBNP. The excess trip reactivity worth is used to show that there is no return to criticality during a steam line break. Avoiding a return to criticality can significantly reduce the mass and energy release rate to containment. The calculation uses the fact that there is no return to criticality to eliminate the need to evaluate many parameters that affect reactivity and the amount of energy created by a return to criticality. By comparing and quantifying the effects of the conservative and non-conservative

Insert B 3.7.2-5 (continued):

parameters, it is shown that the peak containment pressure is 51.3 psig. This peak pressure is less than the containment design pressure of 60 psig. The analysis of the Main Steam Line Break (MSLB) offsite radiological consequences uses the analytical methods and assumptions outlined in the Standard Review Plan (Reference 5). For the pre-accident iodine spike, it is assumed that a reactor transient has occurred prior to the MSLB and has raised the RCS iodine concentration to the allowed Technical Specification value of 50 μCi/gm of dose equivalent (DE) I-131 at 100% power. For the accident-initiated iodine spike, the reactor trip associated with the MSLB creates an iodine spike in the RCS which increases the iodine release rate from the fuel to the RCS to a value of 500 times greater than the release rate corresponding to the maximum equilibrium RCS Technical Specification concentration of 0.8 μCi/gm of DE I-131. The affected SG will rapidly depressurize and release to the outside atmosphere the radioiodines initially contained in the secondary coolant and the radioiodines which are transferred from the primary coolant through SG tube leakage. A portion of the iodine activity initially contained in the intact SGs and noble gas activity due to tube leakage is released to atmosphere as well. The amount of primary to secondary SG tube leakage in each of th e two SGs is assumed to be equal to the Technical Specification limit for a single SG of 0.35 gpm. No credit for iodine removal is taken for any steam released to the condenser prior to reactor trip an d concurrent loss of offsite power. The SG connected to the ruptured main stream line is assumed to boil dry. The entire liquid inventory of this SG is assumed to be steamed off and all of the iodine initially in this SG is released to the environment. Also, iodine carried over to the faulted SG by SG tube leaks is assumed to be released directly to the environment with no credit taken for iodine retention in the SG.

Following a steam generator tube rupture, closure of the MSIVs isolates the ruptured steam generator from the intact steam generator to minimize radiological releases.

In addition to providing SG isolation during a SLB or SGTR, the MSIVs are also containment isolation valves. The containment isolation function of these valves is addressed under LCO 3.6.3.

Insert B 3.7.2-6:

The steam line non-return check valves are considered to be operable when they are capable of closing in response to reverse flow.

LCO 3.7.2 BASES INSERTS

Insert B 3.7.2-7:

The MSIVs are containment isolation valves, and as such the applicable Conditions and Required Actions of LCO 3.6.3 must be entered if containment isolation capability is lost. The 8 hour Completion Time associated with this LCO for an MSIV is greater than that normally allowed for containment isolation valves because the MSIVs are valves that isolate a closed system penetrating containment.

Insert B 3.7.2-8:

In addition, the Required Actions are modified by a note which allows the MSIVs and non-return check valves to be opened under administrative controls for the plant cooldowns. These administrative controls consist of establishing a dedicated operator, who is in communication with the control room. In this way, the penetration can be rapidly isolated if necessary. This allowance is necessary to prevent significant differential temperature and pressures from developing between the SGs when cooling the plant down using the condenser steam dumps.

Insert B 3.7.2-9:

Similarly, since the non-return check valves are required to be OPERABLE in MODES 2 and 3, the inoperable non-return check valve may either be restored to OPERABLE status or closed. When closed, the non-return check valves is also in its required position. In order to prevent inadvertent opening of the MSIV or non-return check valves, due to differential pressure changes between the SG and the steam lines, the Required Actions requires that both the MSIV and non-return check valve in the affected flowpath be closed and the MSIV de-activated whenever either valve is inoperable. Deactivation of the MSIV may be accomplished through removing power to the actuation solenoids or by isolation and venting of the air operator.

LCO 3.7.2 BASES INSERTS

Insert B 3.7.2-10:

SR 3.7.2.2

This SR verifies that each MSIV will actuate to its isolation position on a actuation isolation signal. The 18 month Frequency is based on a refueling cycle interval and the potential for an unplanned transient if the Surveillance were performed with the reactor at power. Operating experience has shown that these components normally pass this Surveillance when performed at the 18 month Frequency. Therefore, the Frequency was concluded to be acceptable from a reliability standpoint