JAFNPP

IMPROVED STANDARD TECHNICAL SPECIFICATIONS (ISTS) CONVERSION

ITS: 3.4.7

Residual Heat Removal (RHR) Shutdown Cooling System - Hot Shutdown

MARKUP OF CURRENT TECHNICAL SPECIFICATIONS (CTS)

DISCUSSION OF CHANGES (DOCs) TO THE CTS

NO SIGNIFICANT HAZARDS CONSIDERATION (NSHC) FOR LESS RESTRICTIVE CHANGES

MARKUP OF NUREG-1433, REVISION 1, SPECIFICATION

JUSTIFICATION FOR DIFFERENCES (JFDs) FROM NUREG-1433, REVISION 1

MARKUP OF NUREG-1433, REVISION 1, BASES

JUSTIFICATION FOR DIFFERENCES (JFDs) FROM NUREG-1433, REVISION 1, BASES

RETYPED PROPOSED IMPROVED TECHNICAL SPECIFICATIONS (ITS) AND BASES

JAFNPP

IMPROVED STANDARD TECHNICAL SPECIFICATIONS (ISTS) CONVERSION

ITS: 3.4.7

Residual Heat Removal (RHR) Shutdown Cooling System - Hot Shutdown

MARKUP OF CURRENT TECHNICAL SPECIFICATIONS (CTS)

M1

INSERT NEW SPECIFICATION 3.4.7

Insert new Specification 3.4.7, "Residual Heat Removal (RHR) Shutdown Cooling System - Hot Shutdown" as shown in the JAFNPP Improved Technical Specifications.

JAFNPP

IMPROVED STANDARD TECHNICAL SPECIFICATIONS (ISTS) CONVERSION

ITS: 3.4.7

Residual Heat Removal (RHR) Shutdown Cooling System - Hot Shutdown

DISCUSSION OF CHANGES (DOCs) TO THE CTS

ADMINISTRATIVE CHANGES

None

TECHNICAL CHANGES - MORE RESTRICTIVE

M1 A Specification (ITS 3.4.7) is being added requiring two RHR shutdown cooling subsystems to be Operable in MODE 3 with reactor steam dome pressure less than the shutdown cooling permissive pressure. In MODE 3, the RHR shutdown cooling subsystems are not required to mitigate any events or accidents in the safety analyses. The RHR shutdown cooling subsystems were identified as important contributors to risk reduction and, therefore, included in the JAFNPP ITS in accordance with Criterion 4 of 10 CFR 50.36(c)(2)(ii). Appropriate Actions and a Surveillance Requirement are also being added. The addition of the new Specification is a more restrictive change necessary to ensure residual heat removal capability is available.

TECHNICAL CHANGES - LESS RESTRICTIVE (GENERIC)

None

TECHNICAL CHANGES - LESS RESTRICTIVE (SPECIFIC)

None

TECHNICAL CHANGES - RELOCATIONS

None

JAFNPP

IMPROVED STANDARD TECHNICAL SPECIFICATIONS (ISTS) CONVERSION

ITS: 3.4.7

Residual Heat Removal (RHR) Shutdown Cooling System - Hot Shutdown

NO SIGNIFICANT HAZARDS CONSIDERATION (NSHC) FOR LESS RESTRICTIVE CHANGES

NO SIGNIFICANT HAZARDS CONSIDERATION
ITS: 3.4.7 - RHR SHUTDOWN COOLING SYSTEM-HOT SHUTDOWN

TECHNICAL CHANGES - LESS RESTRICTIVE (SPECIFIC)

There are no plant specific less restrictive changes identified for this Specification.

CTS

[DOC M1]

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.8 7 PA1 Residual Heat Removal (RHR) Shutdown Cooling System—Hot Shutdown

CLB1

LCO 3.4.8 Two RHR shutdown cooling subsystems shall be OPERABLE~~, and, with no recirculation pump in operation, at least one RHR shutdown cooling subsystem shall be in operation~~.

CLB1

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

~~1. Both RHR shutdown cooling subsystems and recirculation pumps may be removed from operation for up to 2 hours per 8 hour period.~~

X1 ~~2.~~ One RHR shutdown cooling subsystem may be inoperable for up to 2 hours for the performance of Surveillances.

PA2 less than

APPLICABILITY: MODE 3, with reactor steam dome pressure ~~<~~ ~~R~~the RHR cut in permissive pressure~~s~~.

PA3

ACTIONS

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

1. LCO 3.0.4 is not applicable.
2. Separate Condition entry is allowed for each RHR shutdown cooling subsystem.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or two RHR shutdown cooling subsystems inoperable.	A.1 Initiate action to restore RHR shutdown cooling subsystem(s) to OPERABLE status. AND	Immediately (continued)

Typ. All Pages

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. (continued)	A.2 Verify an alternate method of decay heat removal is available for each inoperable RHR shutdown cooling subsystem. AND A.3 Be in MODE 4.	1 hour 24 hours
B. No RHR shutdown cooling subsystem in operation. AND No recirculation pump in operation.	B.1 Initiate action to restore one RHR shutdown cooling subsystem or one recirculation pump to operation. AND B.2 Verify reactor coolant circulation by an alternate method. AND B.3 Monitor reactor coolant temperature and pressure.	Immediately 1 hour from discovery of no reactor coolant circulation AND Once per 12 hours thereafter Once per hour

CLB1

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.4.~~8~~7.1 PA1	--------------------NOTE-------------------- PA3 Not required to be met until 2 hours after reactor steam dome pressure is ~~≥~~ less than PA2 ~~[~~the RHR cut in permissive pressure~~]~~. -- Verify ~~one~~ each required CLB1 RHR shutdown cooling subsystem ~~or recirculation pump is operating~~ manual, power operated, and automatic valve in the flow path that is not locked, sealed, or otherwise secured in position, is aligned or can be aligned to its correct position.	~~12 hours~~ 31 days CLB1

JAFNPP

IMPROVED STANDARD TECHNICAL SPECIFICATIONS (ISTS) CONVERSION

ITS: 3.4.7

Residual Heat Removal (RHR) Shutdown Cooling System - Hot Shutdown

MARKUP OF NUREG-1433, REVISION 1
SPECIFICATION

JAFNPP

IMPROVED STANDARD TECHNICAL SPECIFICATIONS (ISTS) CONVERSION

ITS: 3.4.7

Residual Heat Removal (RHR) Shutdown Cooling System - Hot Shutdown

JUSTIFICATION FOR DIFFERENCES (JFDs) FROM NUREG-1433, REVISION 1

RETENTION OF EXISTING REQUIREMENT (CLB)

CLB1 The requirement in ISTS LCO 3.4.8 (ITS LCO 3.4.7) associated with a recirculation pump or RHR shutdown cooling subsystem being in operation has been deleted. The requirement that two RHR shutdown cooling subsystems are Operable is considered acceptable. Requirements for RHR shutdown cooling subsystem operations are adequately controlled by JAFNPP plant operating procedures and policies. There are no explicit requirements in the CTS for RHR shutdown cooling subsystem operability or that a recirculation pump should be in operation in Hot Shutdown. However, ITS SR 3.4.9.1 will require the reactor coolant temperature to be monitored during RCS heatup and cooldown operations and RCS inservice leak and hydrostatic testing while SR 3.4.9.2 will require these readings to be taken during a planned startup where the reactor will achieve criticality. These SRs will necessitate placing either the recirculation pump or a RHR shutdown cooling subsystem in service to ensure circulation for temperature monitoring or cooling. The RHR shutdown cooling subsystems remove decay heat to reduce temperature of the reactor coolant to < 212°F in preparation for performing Refueling or Cold Shutdown maintenance operations, or for maintaining the reactor stable in the Hot Shutdown conditions. Therefore, an RHR shutdown cooling subsystem will normally be in operation when a cooldown is in progress or to maintain reactor coolant temperature. Without it an alternate will be necessary to achieve the plant objective. During a heatup, the recirculation pumps will normally be in operation to ensure circulation and temperature monitoring. If plant conditions are maintained operations will consider the situation but normally either the recirculation pump will be in service or an RHR shutdown cooling subsystem will in operation to ensure adequate temperature monitoring. This will ensure the plant objective is being met and to ensure the core is in a safe condition. JAFNPP is operated in Hot Shutdown, in a safe manner to ensure decay heat is removed so that no core damage could result, therefore the system is normally continuously operated during these conditions. The requirements in ISTS LCO 3.4.8 are not needed since the Surveillances in ITS 3.4.9 will clearly require temperature monitoring capability and this is accomplished with either any RHR shutdown cooling subsystem or recirculation pump in operation.

Furthermore, to be consistent with this modification, the allowances in ISTS LCO 3.4.7 Note 1 that both RHR shutdown cooling subsystems and recirculation pumps may be removed from operation for up to 2 hours per 8 hour period, the requirements in ISTS 3.4.8 ACTION B for no RHR shutdown cooling subsystem and no recirculation pump in operation have been deleted. ISTS SR 3.4.8.1 (ITS SR 3.4.7.1) requires the verification that one RHR shutdown cooling subsystem or recirculation pump is in operation every 12 hours. Since the requirement to be in

RETENTION OF EXISTING REQUIREMENT (CLB)

CLB1 (continued)

operation has been deleted this SR has been revised to verify each required RHR shutdown cooling subsystem manual, power operated, and automatic valve in the flow path that is not locked, sealed, or otherwise secured in position, is aligned or can be aligned to its correct position every 31 days. The Frequency is consistent with similar Surveillances in other LCOs and is considered adequate for this condition. In Hot Shutdown, the RHR shutdown cooling system is of prime focus of plant operations and since the controls for this system are in the control room the Frequency is considered adequate. This Frequency is consistent with the same type of Surveillance in other LCOs (e.g., ITS 3.5.1).

PLANT SPECIFIC WORDING PREFERENCE OR MINOR EDITORIAL IMPROVEMENT (PA)

PA1 NUREG-1433 Specification 3.4.5, "RCS Pressure Isolation Valve (PIV) Leakage", is not incorporated in ITS. Subsequent ITS Specifications and Bases have been renumbered accordingly.

PA2 Editorial change to maintain consistency with the Writer's Guide for the Restructured Technical Specifications.

PA3 The brackets have been removed and the proper plant specific nomenclature has been provided.

PLANT SPECIFIC DIFFERENCE IN DESIGN OR DESIGN BASIS (DB)

None

DIFFERENCE BASED ON APPROVED TRAVELER (TA)

None

DIFFERENCE BASED ON PENDING TRAVELER (TP)

None

DIFFERENCE FOR OTHER REASONS THAN ABOVE (X)

None

JAFNPP

IMPROVED STANDARD TECHNICAL SPECIFICATIONS (ISTS) CONVERSION

ITS: 3.4.7

Residual Heat Removal (RHR) Shutdown Cooling System - Hot Shutdown

MARKUP OF NUREG-1433, REVISION 1, BASES

B 3.4 REACTOR COOLANT SYSTEM (RCS)

PA1

B 3.4.8 7 Residual Heat Removal (RHR) Shutdown Cooling System—Hot Shutdown

BASES

BACKGROUND PA2 212

Irradiated fuel in the shutdown reactor core generates heat during the decay of fission products and increases the temperature of the reactor coolant. This decay heat must be removed to reduce the temperature of the reactor coolant to ≤ ~~200~~ 212 °F. ~~This decay heat removal is~~ in preparation for performing refueling or maintenance operations, or ~~for keeping~~ maintaining the reactor in the Hot Shutdown condition. PA3 the decay heat must be removed for PA3 Cold Shutdown

The two redundant, manually controlled shutdown cooling ~~subsystems~~ loops PA3 of the RHR System provide decay heat removal. Each loop consists of two motor driven pumps, a heat exchanger, and associated piping and valves. Both loops have a common suction from the same reactor water PA4 recirculation loop. Each pump discharges the reactor coolant, after circulation through the respective heat exchanger, to the reactor via the associated reactor water PA4 recirculation loop. The RHR heat exchangers transfer heat to the RHR Service Water System (LCO 3.7.1, "Residual Heat Removal Service Water (RHRSW) System"). DB1 (Ref. 1)

APPLICABLE SAFETY ANALYSES

Decay heat removal by operation of the RHR System in the shutdown cooling mode is not required for mitigation of any event or accident evaluated in the safety analyses (Ref. 1). Decay heat removal is, however, an important safety function that must be accomplished or core damage could result. ~~Although~~ The RHR shutdown cooling subsystem ~~does not~~ meet~~s a specific~~ ~~criterion of the NRC Policy Statement, it was identified in the NRC Policy Statement as a significant contributor to risk reduction. Therefore, the RHR Shutdown Cooling System is retained as a Technical Specification~~ 10 CFR 50.36(c)(2)(ii) (Ref. 2). 5 X1 4

LCO

Two RHR shutdown cooling subsystems are required to be OPERABLE~~, and when no recirculation pump is in operation, one shutdown cooling subsystem must be in operation~~. An OPERABLE RHR shutdown cooling subsystem consists of one OPERABLE RHR pump, one heat exchanger, and the associated piping and valves. The two subsystems have a common suction source and are allowed to have a common heat exchanger and CLB1

(continued)

Typ. All Pages

BASES

(and two RHR service water pumps) PA3 PA3

LCO (continued)

RHR (and one RHR service water pump) RHR

common discharge piping. Thus, to meet the LCO, both pumps in one loop or one pump in each of the two loops must be OPERABLE. Since the piping and heat exchangers are passive components that are assumed not to fail, they are allowed to be common to both subsystems. Each shutdown cooling subsystem is considered OPERABLE if it can be manually aligned (~~remote or local~~) from the control room or locally (PA4) in the shutdown cooling mode for removal of decay heat. In MODE 3, one RHR shutdown cooling subsystem can provide the required cooling, but two subsystems are required to be OPERABLE to provide redundancy. Operation of one subsystem can maintain or reduce the reactor coolant temperature as required. However, to ensure adequate core flow to allow for accurate average reactor coolant temperature monitoring, nearly continuous operation is required.

~~Note 1 permits both RHR shutdown cooling subsystems to be shut down for a period of 2 hours in an 8 hour period.~~ The Note ~~2~~ allows one RHR shutdown cooling subsystem to be inoperable for up to 2 hours for the performance of Surveillance tests. (CLB1) These tests may be on the affected RHR System or on some other plant system or component that necessitates placing the RHR shutdown cooling subsystem ~~System~~ in an inoperable status during the performance. This is permitted because the core heat generation can be low enough and the heatup rate slow enough (X2) to allow some changes to the RHR shutdown cooling subsystems or other operations requiring ~~RHR flow interruption and~~ loss of redundancy. PA3

APPLICABILITY

In MODE 3 with reactor steam dome pressure below ~~(~~the RHR shutdown cooling suction valve isolation logic (Function 6.a of LCO 3.3.6.1, Primary Containment Isolation Instrumentation) (PA3) cut in permissive pressure ~~(i.e., the actual pressure at which the interlock resets)~~ (PA4) the RHR System ~~may be operated~~ is required to be OPERABLE so that it in the shutdown cooling mode to remove decay heat to reduce or maintain coolant temperature. (CLB1) Otherwise, a recirculation pump is ~~required to be in operation~~ normally in operation to circulate coolant to provide for temperature monitoring. (CLB1)

In MODES 1 and 2, and in MODE 3 with reactor steam dome pressure greater than or equal to ~~(~~the RHR cut in permissive pressure~~)~~, this LCO is not applicable. (PA4) (PA3) Operation of the RHR System in the shutdown cooling mode is not allowed above this pressure because the RCS pressure may exceed the design pressure of the shutdown cooling piping. Decay heat removal at reactor pressures greater than or equal to the RHR cut in permissive pressure is typically accomplished by condensing

(continued)

BASES

APPLICABILITY (continued)

the steam in the main condenser. Additionally, in MODE 2 below this pressure, the OPERABILITY requirements for the Emergency Core Cooling Systems (ECCS) (LCO 3.5.1, "ECCS—Operating") do not allow placing the RHR shutdown cooling subsystem into operation. PA5

The requirements for 8 decay heat removal in MODES 4 and 5 are discussed in LCO 3.4.9, "Residual Heat Removal (RHR) Shutdown Cooling System—Cold Shutdown"; LCO 3.9.8, 7 "Residual Heat Removal (RHR)—High Water Level"; and LCO 3.9.9, "Residual Heat Removal (RHR)—Low Water Level." 8 PA5

ACTIONS

A Note to the ACTIONS excludes the MODE change restriction of LCO 3.0.4. This exception allows entry into the applicable MODE(S) while relying on the ACTIONS even though the ACTIONS may eventually require plant shutdown. This exception is acceptable due to the redundancy of the OPERABLE subsystems, the low pressure at which the plant is operating, the low probability of an event occurring during operation in this condition, and the availability of alternate methods of decay heat removal capability.

A second Note has been provided to modify the ACTIONS related to RHR shutdown cooling subsystems. Section 1.3, Completion Times, specifies once a Condition has been entered, subsequent divisions, subsystems, components or variables expressed in the Condition, discovered to be inoperable or not within limits, will not result in separate entry into the Condition. Section 1.3 also specifies Required Actions of the Condition continue to apply for each additional failure, with Completion Times based on initial entry into the Condition. However, the Required Actions for inoperable shutdown cooling subsystems provide appropriate compensatory measures for separate inoperable shutdown cooling subsystems. As such, a Note has been provided that allows separate Condition entry for each inoperable RHR shutdown cooling subsystem.

<u>A.1, A.2, and A.3</u> the CLB1

With one required RHR shutdown cooling subsystem inoperable for decay heat removal, except as permitted by the LCO Note 2, the inoperable subsystem must be restored to OPERABLE status

(continued)

BASES

ACTIONS

A.1, A.2, and A.3 (continued)

without delay. In this condition, the remaining OPERABLE subsystem can provide the necessary decay heat removal. The overall reliability is reduced, however, because a single failure in the OPERABLE subsystem could result in reduced RHR shutdown cooling capability. Therefore, an alternate method of decay heat removal must be provided.

With both RHR shutdown cooling subsystems inoperable, an alternate method of decay heat removal must be provided in addition to that provided for the initial RHR shutdown cooling subsystem inoperability. This re-establishes backup decay heat removal capabilities, similar to the requirements of the LCO. The 1 hour Completion Time is based on the decay heat removal function and the probability of a loss of the available decay heat removal capabilities.

The required cooling capacity of the alternate method should be ensured by verifying (by calculation or demonstration) its capability to maintain or reduce temperature. Decay heat removal by ambient losses can be considered as, or contributing to, the alternate method capability. Alternate methods that can be used include (but are not limited to) ~~the Spent Fuel Pool Cooling System and the Reactor Water Cleanup System.~~

DB2

Condensate and Main Steam Systems, Reactor Water Cleanup System (by itself or using feed and bleed in combination with the Control Rod Drive System or Condensate System), or a combination of an RHR pump and Safety/relief valve(s)

However, due to the potentially reduced reliability of the alternate methods of decay heat removal, it is also required to reduce the reactor coolant temperature to the point where MODE 4 is entered.

CLB1

B.1, B.2, and B.3

With no RHR shutdown cooling subsystem and no recirculation pump in operation, except as permitted by LCO Note 1, reactor coolant circulation by the RHR shutdown cooling subsystem or recirculation pump must be restored without delay.

Until RHR or recirculation pump operation is re-established, an alternate method of reactor coolant circulation must be placed into service. This will provide the necessary circulation for monitoring coolant temperature. The 1 hour Completion Time is based on the coolant circulation function

(continued)

PA1

BASES

ACTIONS

B.1, B.2, and B.3 (continued)

CLB1

~~and is modified such that the 1 hour is applicable separately for each occurrence involving a loss of coolant circulation. Furthermore, verification of the functioning of the alternate method must be reconfirmed every 12 hours thereafter. This will provide assurance of continued temperature monitoring capability.~~

~~During the period when the reactor coolant is being circulated by an alternate method (other than by the required RHR shutdown cooling subsystem or recirculation pump), the reactor coolant temperature and pressure must be periodically monitored to ensure proper function of the alternate method. The once per hour Completion Time is deemed appropriate.~~

SURVEILLANCE REQUIREMENTS

SR 3.4.8.1 7

PA1

Insert SR 3.4.7.1

CLB1

~~This Surveillance verifies that one RHR shutdown cooling subsystem or recirculation pump is in operation and circulating reactor coolant. The required flow rate is determined by the flow rate necessary to provide sufficient decay heat removal capability. The Frequency of 12 hours is sufficient in view of other visual and audible indications available to the operator for monitoring the RHR subsystem in the control room.~~

CLB1

This Surveillance is modified by a Note allowing sufficient time to align the RHR System for shutdown cooling operation after clearing the pressure interlock that isolates the system, ~~or for placing a recirculation pump in operation~~. The Note takes exception to the requirements of the Surveillance being met (i.e., ~~forced coolant circulation is~~ not required for this initial 2 hour period), which also allows entry into the Applicability of this Specification in accordance with SR 3.0.4 since the Surveillance will not be "not met" at the time of entry into the Applicability.

subsystem

Verify

OPERABILITY

valves are aligned or can be aligned is

REFERENCES

None.

1. 10 CFR 50.36 (c)(2)(ii)

X11

1. UFSAR, chapter 14

DB1

CLB1

INSERT SR 3.4.7.1

Verifying the correct alignment for manual, power operated, and automatic valves in the RHR shutdown cooling flow path provides assurance that the proper flow paths will exist for RHR operation. This SR does not apply to valves that are locked, sealed, or otherwise secured in position since these were verified to be in the correct position prior to locking, sealing, or securing. A valve that can be manually (from the control room or locally) aligned is allowed to be in a non-RHR shutdown cooling position provided the valve can be repositioned. This SR does not require any testing or valve manipulation; rather, it involves verification that those valves capable of potentially being mispositioned are in the correct position. This SR does not apply to valves that cannot be inadvertently misaligned, such as check valves.

The 31 day Frequency of this SR was derived from the Inservice Testing Program requirements for performing valve testing at least once every 92 days. The Frequency of 31 days is further justified because the valves are operated under procedural control. This Frequency has been shown to be acceptable through operating experience.

JAFNPP

IMPROVED STANDARD TECHNICAL SPECIFICATIONS (ISTS) CONVERSION

ITS: 3.4.7

Residual Heat Removal (RHR) Shutdown Cooling System - Hot Shutdown

JUSTIFICATION FOR DIFFERENCES (JFDs) FROM NUREG-1433, REVISION 1, BASES

RETENTION OF EXISTING REQUIREMENT (CLB)

CLB1 The requirement in ISTS LCO 3.4.8 (ITS LCO 3.4.7) associated with a recirculation pump or RHR shutdown cooling subsystem being in operation has been deleted. The requirement that two RHR shutdown cooling subsystems are Operable is considered acceptable. Requirements for RHR shutdown cooling subsystem operations are adequately controlled by JAFNPP plant operating procedures and policies. There are no explicit requirements in the CTS for RHR shutdown cooling subsystem operability or that a recirculation pump should be in operation in Hot Shutdown. However, ITS SR 3.4.9.1 will require the reactor coolant temperature to be monitored during RCS heatup and cooldown operations and RCS inservice leak and hydrostatic testing while SR 3.4.9.2 will require these readings to be taken during a planned startup where the reactor will achieve criticality. These SRs will necessitate placing either the recirculation pump or a RHR shutdown cooling subsystem in service to ensure circulation for temperature monitoring or cooling. The RHR shutdown cooling subsystems remove decay heat to reduce temperature of the reactor coolant to $\leq$ 212°F in preparation for performing Refueling or Cold Shutdown maintenance operations, or for maintaining the reactor stable in the Hot Shutdown conditions. Therefore, an RHR shutdown cooling subsystem will normally be in operation when a cooldown is in progress or to maintain reactor coolant temperature. Without it an alternate will be necessary to achieve the plant objective. During a heatup, the recirculation pumps will normally be in operation to ensure circulation and temperature monitoring. If plant conditions are maintained operations will consider the situation but normally either the recirculation pump will be in service or an RHR shutdown cooling subsystem will in operation to ensure adequate temperature monitoring. This will ensure the plant objective is being met and to ensure the core is in a safe condition. JAFNPP is operated in Hot Shutdown, in a safe manner to ensure decay heat is removed so that no core damage could result, therefore the system is normally continuously operated during these conditions. The requirements in ISTS LCO 3.4.8 are not needed since the Surveillances in ITS 3.4.9 will clearly require temperature monitoring capability and this is accomplished with either any RHR shutdown cooling subsystem or recirculation pump in operation.

Furthermore, to be consistent with this modification, the allowances in ISTS LCO 3.4.7 Note 1 that both RHR shutdown cooling subsystems and recirculation pumps may be removed from operation for up to 2 hours per 8 hour period, the requirements in ISTS 3.4.8 ACTION B for no RHR shutdown cooling subsystem and no recirculation pump in operation have been deleted. ISTS SR 3.4.8.1 (ITS SR 3.4.7.1) requires the verification that one RHR shutdown cooling subsystem or recirculation pump is in operation every 12 hours. Since the requirement to be in

RETENTION OF EXISTING REQUIREMENT (CLB)

CLB1 (continued)

operation has been deleted this SR has been revised to verify each required RHR shutdown cooling subsystem manual, power operated, and automatic valve in the flow path that is not locked, sealed, or otherwise secured in position, is aligned or can be aligned to its correct position every 31 days. The Frequency is consistent with similar Surveillances in other LCOs and is considered adequate for this condition. In Hot Shutdown, the RHR shutdown cooling system is of prime focus of plant operations and since the controls for this system are in the control room the Frequency is considered adequate. This Frequency is consistent with the same type of Surveillance in other LCOs (e.g., ITS 3.5.1).

PLANT SPECIFIC WORDING PREFERENCE OR MINOR EDITORIAL IMPROVEMENT (PA)

PA1 NUREG-1433 Specification 3.4.5, "RCS Pressure Isolation Valve (PIV) Leakage," is not incorporated in ITS. Subsequent ITS Specifications and Bases have been renumbered accordingly.

PA2 The Bases have been revised to be consistent with changes made to the Specifications.

PA3 Editorial changes have been made for enhanced clarification, correction, or improvement with no change in intent.

PA4 The Bases have been modified to reflect plant specific terminology.

PA5 The correct LCO number has been included.

PLANT SPECIFIC DIFFERENCE IN DESIGN OR DESIGN BASIS (DB)

DB1 The Bases have been modified to reflect JAFNPP specific References.

DB2 The Bases have been revised to reflect the JAFNPP specific design.

DIFFERENCE BASED ON APPROVED TRAVELER (TA)

None

JUSTIFICATION FOR DIFFERENCES FROM NUREG-1433, REVISION 1
ITS BASES: 3.4.7 - RHR SHUTDOWN COOLING SYSTEM-HOT SHUTDOWN

DIFFERENCE BASED ON PENDING TRAVELER (TP)

None

DIFFERENCE FOR OTHER REASONS THAN ABOVE (X)

X1 NUREG-1433, Revision 1, Bases references to "the NRC Policy Statement" has been replaced with 10 CFR 50.36(c)(2)(ii), in accordance with 60 FR 36953 effective August 18, 1995.

JAFNPP

IMPROVED STANDARD TECHNICAL SPECIFICATIONS (ISTS) CONVERSION

ITS: 3.4.7

Residual Heat Removal (RHR) Shutdown Cooling System - Hot Shutdown

RETYPED PROPOSED IMPROVED TECHNICAL SPECIFICATIONS (ITS) AND BASES

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.7 Residual Heat Removal (RHR) Shutdown Cooling System—Hot Shutdown

LCO 3.4.7 Two RHR shutdown cooling subsystems shall be OPERABLE.

-------------------------NOTE-------------------------
One RHR shutdown cooling subsystem may be inoperable for up to 2 hours for the performance of Surveillances.

APPLICABILITY: MODE 3, with reactor steam dome pressure less than the RHR cut in permissive pressure.

ACTIONS

-------------------------NOTES-------------------------
1. LCO 3.0.4 is not applicable.
2. Separate Condition entry is allowed for each RHR shutdown cooling subsystem.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or two RHR shutdown cooling subsystems inoperable.	A.1 Initiate action to restore RHR shutdown cooling subsystem(s) to OPERABLE status.	Immediately
	<u>AND</u>	
	A.2 Verify an alternate method of decay heat removal is available for each inoperable RHR shutdown cooling subsystem.	1 hour
	<u>AND</u>	
	A.3 Be in MODE 4.	24 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.4.7.1	------------------NOTE------------------ Not required to be met until 2 hours after reactor steam dome pressure is less than the RHR cut in permissive pressure. --- Verify each required RHR shutdown cooling subsystem manual, power operated, and automatic valve in the flow path that is not locked, sealed, or otherwise secured in position, is aligned or can be aligned to its correct position.	31 days

B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.7 Residual Heat Removal (RHR) Shutdown Cooling System—Hot Shutdown

BASES

BACKGROUND

Irradiated fuel in the shutdown reactor core generates heat during the decay of fission products and increases the temperature of the reactor coolant. This decay heat must be removed to reduce the temperature of the reactor coolant to ≤ 212°F in preparation for performing Refueling or Cold Shutdown maintenance operations, or the decay heat must be removed for maintaining the reactor in the Hot Shutdown condition.

The two redundant, manually controlled shutdown cooling subsystems (loops) of the RHR System provide decay heat removal. Each loop consists of two motor driven pumps, a heat exchanger, and associated piping and valves. Both loops have a common suction from the same reactor water recirculation loop. Each pump discharges the reactor coolant, after circulation through the respective heat exchanger, to the reactor via the associated reactor water recirculation loop. The RHR heat exchangers transfer heat to the RHR Service Water System (LCO 3.7.1, "Residual Heat Removal Service Water (RHRSW) System").

APPLICABLE SAFETY ANALYSES

Decay heat removal by operation of the RHR System in the shutdown cooling mode is not required for mitigation of any event or accident evaluated in the safety analyses (Ref. 1). Decay heat removal is, however, an important safety function that must be accomplished or core damage could result. The RHR shutdown cooling subsystem meets Criterion 4 of 10 CFR 50.36(c)(2)(ii) (Ref. 2).

LCO

Two RHR shutdown cooling subsystems are required to be OPERABLE. An OPERABLE RHR shutdown cooling subsystem consists of one OPERABLE RHR pump, one heat exchanger, and the associated piping and valves. The two subsystems have a common suction source and are allowed to have a common heat exchanger and common discharge piping. Thus, to meet the LCO, both RHR pumps (and two RHR service water pumps) in one loop or one RHR pump (and one RHR service water pump) in

(continued)

BASES

LCO (continued)

each of the two loops must be OPERABLE. Since the piping and heat exchangers are passive components that are assumed not to fail, they are allowed to be common to both subsystems. Each shutdown cooling subsystem is considered OPERABLE if it can be manually aligned (from the control room or locally) in the shutdown cooling mode for removal of decay heat. In MODE 3, one RHR shutdown cooling subsystem can provide the required cooling, but two subsystems are required to be OPERABLE to provide redundancy. Operation of one subsystem can maintain or reduce the reactor coolant temperature as required. However, to ensure adequate core flow to allow for accurate average reactor coolant temperature monitoring, nearly continuous operation is required.

The Note allows one RHR shutdown cooling subsystem to be inoperable for up to 2 hours for the performance of Surveillance tests. These tests may be on the affected RHR System or on some other plant system or component that necessitates placing the RHR shutdown cooling subsystems in an inoperable status during the performance. This is permitted because the core heat generation can be low enough and the heatup rate slow enough to allow some changes to the RHR shutdown cooling subsystems or other operations requiring loss of redundancy.

APPLICABILITY

In MODE 3 with reactor steam dome pressure below the RHR cut-in permissive pressure (i.e., the actual pressure at which the shutdown cooling suction valve isolation logic interlock resets (Function 6.a of LCO 3.3.6.1, Primary Containment Isolation Instrumentation)) the RHR System is required to be OPERABLE so that it may be operated in the shutdown cooling mode to remove decay heat to reduce or maintain coolant temperature. Otherwise, a recirculation pump is normally in operation to circulate coolant to provide for temperature monitoring.

In MODES 1 and 2, and in MODE 3 with reactor steam dome pressure greater than or equal to the RHR cut-in permissive pressure, this LCO is not applicable. Operation of the RHR System in the shutdown cooling mode is not allowed above this pressure because the RCS pressure may exceed the design pressure of the shutdown cooling piping. Decay heat removal

(continued)

BASES

APPLICABILITY (continued)

at reactor pressures greater than or equal to the RHR cut in permissive pressure is typically accomplished by condensing the steam in the main condenser. Additionally, in MODE 2 below this pressure, the OPERABILITY requirements for the Emergency Core Cooling Systems (ECCS) (LCO 3.5.1, "ECCS – Operating") do not allow placing the RHR shutdown cooling subsystem into operation.

The requirements for decay heat removal in MODES 4 and 5 are discussed in LCO 3.4.8, "Residual Heat Removal (RHR) Shutdown Cooling System – Cold Shutdown"; LCO 3.9.7, "Residual Heat Removal (RHR) – High Water Level"; and LCO 3.9.8, "Residual Heat Removal (RHR) – Low Water Level."

ACTIONS

A Note to the ACTIONS excludes the MODE change restriction of LCO 3.0.4. This exception allows entry into the applicable MODE(S) while relying on the ACTIONS even though the ACTIONS may eventually require plant shutdown. This exception is acceptable due to the redundancy of the OPERABLE subsystems, the low pressure at which the plant is operating, the low probability of an event occurring during operation in this condition, and the availability of alternate methods of decay heat removal capability.

A second Note has been provided to modify the ACTIONS related to RHR shutdown cooling subsystems. Section 1.3, Completion Times, specifies once a Condition has been entered, subsequent divisions, subsystems, components or variables expressed in the Condition, discovered to be inoperable or not within limits, will not result in separate entry into the Condition. Section 1.3 also specifies Required Actions of the Condition continue to apply for each additional failure, with Completion Times based on initial entry into the Condition. However, the Required Actions for inoperable shutdown cooling subsystems provide appropriate compensatory measures for separate inoperable shutdown cooling subsystems. As such, a Note has been provided that allows separate Condition entry for each inoperable RHR shutdown cooling subsystem.

(continued)

BASES

ACTIONS (continued)

A.1, A.2, and A.3

With one required RHR shutdown cooling subsystem inoperable for decay heat removal, except as permitted by the LCO Note, the inoperable subsystem must be restored to OPERABLE status without delay. In this condition, the remaining OPERABLE subsystem can provide the necessary decay heat removal. The overall reliability is reduced, however, because a single failure in the OPERABLE subsystem could result in reduced RHR shutdown cooling capability. Therefore, an alternate method of decay heat removal must be provided.

With both RHR shutdown cooling subsystems inoperable, an alternate method of decay heat removal must be provided in addition to that provided for the initial RHR shutdown cooling subsystem inoperability. This re-establishes backup decay heat removal capabilities, similar to the requirements of the LCO. The 1 hour Completion Time is based on the decay heat removal function and the probability of a loss of the available decay heat removal capabilities.

The required cooling capacity of the alternate method should be ensured by verifying (by calculation or demonstration) its capability to maintain or reduce temperature. Decay heat removal by ambient losses can be considered as, or contributing to, the alternate method capability. Alternate methods that can be used include (but are not limited to) the Condensate and Main Steam Systems, Reactor Water Cleanup System (by itself or using feed and bleed in combination with the Control Rod Drive System or Condensate System), or a combination of an RHR pump and safety/relief valve(s).

However, due to the potentially reduced reliability of the alternate methods of decay heat removal, it is also required to reduce the reactor coolant temperature to the point where MODE 4 is entered.

SURVEILLANCE REQUIREMENTS

SR 3.4.7.1

Verifying the correct alignment for manual, power operated, and automatic valves in the RHR shutdown cooling flow path provides assurance that the proper flow paths will exist for RHR operation. This SR does not apply to valves that are

(continued)

BASES

SURVEILLANCE REQUIREMENTS	SR 3.4.7.1 (continued)

locked, sealed, or otherwise secured in position since these were verified to be in the correct position prior to locking, sealing, or securing. A valve that can be manually (from the control room or locally) aligned is allowed to be in a non-RHR shutdown cooling position provided the valve can be repositioned. This SR does not require any testing or valve manipulation; rather, it involves verification that those valves capable of potentially being mispositioned are in the correct position. This SR does not apply to valves that cannot be inadvertently misaligned, such as check valves.

The 31 day Frequency of this SR was derived from the Inservice Testing Program requirements for performing valve testing at least once every 92 days. The Frequency of 31 days is further justified because the valves are operated under procedural control. This Frequency has been shown to be acceptable through operating experience.

This Surveillance is modified by a Note allowing sufficient time to verify RHR shutdown cooling subsystem OPERABILITY after clearing the pressure interlock that isolates the system. The Note takes exception to the requirements of the Surveillance being met (i.e., valves are aligned or can be aligned is not required for this initial 2 hour period), which also allows entry into the Applicability of this Specification in accordance with SR 3.0.4 since the Surveillance will not be "not met" at the time of entry into the Applicability.

REFERENCES

1. UFSAR, Chapter 14.
2. 10 CFR 50.36(c)(2)(ii).

JAFNPP

IMPROVED STANDARD TECHNICAL SPECIFICATIONS (ISTS) CONVERSION

ITS: 3.4.8

Residual Heat Removal (RHR) Shutdown Cooling System - Cold Shutdown

MARKUP OF CURRENT TECHNICAL SPECIFICATIONS (CTS)

DISCUSSION OF CHANGES (DOCs) TO THE CTS

NO SIGNIFICANT HAZARDS CONSIDERATION (NSHC) FOR LESS RESTRICTIVE CHANGES

MARKUP OF NUREG-1433, REVISION 1, SPECIFICATION

JUSTIFICATION FOR DIFFERENCES (JFDs) FROM NUREG-1433, REVISION 1

MARKUP OF NUREG-1433, REVISION 1, BASES

JUSTIFICATION FOR DIFFERENCES (JFDs) FROM NUREG-1433, REVISION 1, BASES

RETYPED PROPOSED IMPROVED TECHNICAL SPECIFICATIONS (ITS) AND BASES

JAFNPP

IMPROVED STANDARD TECHNICAL SPECIFICATIONS (ISTS) CONVERSION

ITS: 3.4.8

Residual Heat Removal (RHR) Shutdown Cooling System - Cold Shutdown

MARKUP OF CURRENT TECHNICAL SPECIFICATIONS (CTS)

M1

Insert New Specification 3.4.8

Insert new ITS Specification 3.4.8, "Residual Heat Removal (RHR) Shutdown Cooling System-Cold Shutdown" as shown in the JAFNPP Improved Technical Specifications.

JAFNPP

IMPROVED STANDARD TECHNICAL SPECIFICATIONS (ISTS) CONVERSION

ITS: 3.4.8

Residual Heat Removal (RHR) Shutdown Cooling System - Cold Shutdown

DISCUSSION OF CHANGES (DOCs) TO THE CTS

ADMINISTRATIVE CHANGES

None

TECHNICAL CHANGES - MORE RESTRICTIVE

M1 A Specification (ITS 3.4.8) is being added requiring two RHR shutdown cooling subsystems to be Operable in MODE 4. In MODE 4, the RHR shutdown cooling subsystems are not required to mitigate any events or accidents in the safety analyses. The RHR shutdown cooling subsystems were identified as important contributors to risk reduction and, therefore, included in the JAFNPP ITS in accordance with Criterion 4 of 10 CFR 50.36(c)(2)(ii). Appropriate Actions and a Surveillance Requirement have also been added. The addition of the new Specification is a more restrictive change necessary to ensure residual heat removal capability is available.

TECHNICAL CHANGES - LESS RESTRICTIVE (GENERIC)

None

TECHNICAL CHANGES - LESS RESTRICTIVE (SPECIFIC)

None

TECHNICAL CHANGES - RELOCATIONS

None

JAFNPP

IMPROVED STANDARD TECHNICAL SPECIFICATIONS (ISTS) CONVERSION

ITS: 3.4.8

Residual Heat Removal (RHR) Shutdown Cooling System - Cold Shutdown

NO SIGNIFICANT HAZARDS CONSIDERATION (NSHC) FOR LESS RESTRICTIVE CHANGES

NO SIGNIFICANT HAZARDS CONSIDERATION
ITS: 3.4.8 - RHR SHUTDOWN COOLING SYSTEM-COLD SHUTDOWN

TECHNICAL CHANGES - LESS RESTRICTIVE (SPECIFIC)

There are no plant specific less restrictive changes identified for this Specification.

JAFNPP

IMPROVED STANDARD TECHNICAL SPECIFICATIONS (ISTS) CONVERSION

ITS: 3.4.8

Residual Heat Removal (RHR) Shutdown Cooling System - Cold Shutdown

MARKUP OF NUREG-1433, REVISION 1
SPECIFICATION

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.9 8 Residual Heat Removal (RHR) Shutdown Cooling System—Cold Shutdown

PA1

LCO 3.4.9 Two RHR shutdown cooling subsystems shall be OPERABLE. ~~and, with no recirculation pump in operation, at least one RHR shutdown cooling subsystem shall be in operation~~. CLB1

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

~~1. Both RHR shutdown cooling subsystems and recirculation pumps may be removed from operation for up to 2 hours per 8 hour period.~~

CLB1 2. One RHR shutdown cooling subsystem may be inoperable for up to 2 hours for the performance of Surveillances.

APPLICABILITY: MODE 4.

ACTIONS

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

Separate Condition entry is allowed for each shutdown cooling subsystem.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or two RHR shutdown cooling subsystems inoperable.	A.1 Verify an alternate method of decay heat removal is available for each inoperable RHR shutdown cooling subsystem.	1 hour AND Once per 24 hours thereafter

(continued)

Typ. All Pages

ACTIONS (continued)

CONDITION	REQUIRED ACTION		COMPLETION TIME
B. No RHR shutdown cooling subsystem in operation. AND No recirculation pump in operation.	B.1	Verify reactor coolant circulating by an alternate method.	1 hour from discovery of no reactor coolant circulation AND Once per 12 hours thereafter
	AND B.2	Monitor reactor coolant temperature.	Once per hour

CLB1

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.4.9.1 8 PA1	Verify ~~one~~ each RHR shutdown cooling subsystem ~~or recirculation pump is operating~~ manual, power operated, and automatic valve in the flow path that is not locked, sealed, or otherwise secured in position, is aligned or can be aligned to its correct position.	~~12 hours~~ 31 days CLB1

JAFNPP

IMPROVED STANDARD TECHNICAL SPECIFICATIONS (ISTS) CONVERSION

ITS: 3.4.8

Residual Heat Removal (RHR) Shutdown Cooling System - Cold Shutdown

JUSTIFICATION FOR DIFFERENCES (JFDs) FROM NUREG-1433, REVISION 1

RETENTION OF EXISTING REQUIREMENT (CLB)

CLB1 The requirement in ISTS LCO 3.4.9 (ITS LCO 3.4.8) associated with a recirculation pump or RHR shutdown cooling subsystem being in operation has been deleted. The requirement that two RHR shutdown cooling subsystems are Operable is considered acceptable. Requirements for RHR shutdown cooling subsystem operations are adequately controlled by JAFNPP plant operating procedures and policies. There are no explicit requirements in the CTS for RHR shutdown cooling subsystem operability or that a recirculation pump should be in operation in Cold Shutdown. However, ITS SR 3.4.9.1 will require the reactor coolant temperature to be monitored during RCS heatup and cooldown operations and RCS inservice leak and hydrostatic testing while SR 3.4.9.2 will require these readings to be taken during a planned startup where the reactor will achieve criticality. These Surveillance will necessitate placing either the recirculation pump or a RHR shutdown cooling subsystem in service to ensure circulation for temperature monitoring or cooling. The RHR shutdown cooling subsystems remove decay heat to reduce temperature of the reactor coolant to $\leq$ 212°F in preparation for performing Refueling or Cold Shutdown maintenance operations, or for maintaining the reactor stable in the Cold Shutdown conditions. Therefore, an RHR shutdown cooling subsystem will normally be in operation when a cooldown is in progress or to maintain reactor coolant temperature. Without it an alternate will be necessary to achieve the plant objective. During a heatup, the recirculation pumps will normally be in operation to ensure circulation and temperature monitoring. If the plant is maintaining plant conditions the plant will consider the situation but normally either the recirculation pump will be in service or an RHR shutdown cooling subsystem will in operation to ensure adequate temperature monitoring. This will ensure the plant objective is being met and to ensure the core is in a safe condition. JAFNPP is operated in Cold Shutdown, in a safe manner to ensure decay heat is removed so that no core damage could result, therefore the system is normally continuously operated during these conditions. The requirements in ISTS LCO 3.4.9 are not needed since the Surveillances in ITS 3.4.9 will clearly require temperature monitoring capability and this is accomplished with either one RHR shutdown cooling subsystem or one recirculation pump in operation.

Furthermore, to be consistent with this modification, the allowances in ISTS LCO 3.4.9 Note 1 that both RHR shutdown cooling subsystems and recirculation pumps may be removed from operation for up to 2 hours per 8 hour period, the requirements in ISTS 3.4.9 ACTION B for no RHR shutdown cooling subsystem and no recirculation pump in operation have been deleted. ISTS SR 3.4.9.1 (ITS SR 3.4.8.1) requires the verification that one RHR shutdown cooling subsystem or recirculation pump is in operation every 12 hours. Since the requirement to be in

RETENTION OF EXISTING REQUIREMENT (CLB)

CLB1 (continued)

operation has been deleted this SR has been revised to verify each required RHR shutdown cooling subsystem manual, power operated, and automatic valve in the flow path that is not locked, sealed, or otherwise secured in position, is aligned or can be aligned to its correct position every 31 days. The Frequency is consistent with similar Surveillances in other LCOs and is considered adequate for this condition. In Cold Shutdown, the RHR shutdown cooling system is of prime focus of plant operations and since the controls for this system are in the control room the Frequency is considered adequate. This Frequency is consistent with the same type of Surveillance in other LCOs (e.g., ITS 3.5.1).

PLANT SPECIFIC WORDING PREFERENCE OR MINOR EDITORIAL IMPROVEMENT (PA)

PA1 NUREG-1433 Specification 3.4.5, "RCS Pressure Isolation Valve (PIV) Leakage," is not incorporated in ITS. Subsequent ITS Specifications and Bases have been renumbered accordingly.

PLANT SPECIFIC DIFFERENCE IN DESIGN OR DESIGN BASIS (DB)

None

DIFFERENCE BASED ON APPROVED TRAVELER (TA)

None

DIFFERENCE BASED ON PENDING TRAVELER (TP)

None

DIFFERENCE FOR OTHER REASONS THAN ABOVE (X)

None

JAFNPP

IMPROVED STANDARD TECHNICAL SPECIFICATIONS (ISTS) CONVERSION

ITS: 3.4.8

Residual Heat Removal (RHR) Shutdown Cooling System - Cold Shutdown

MARKUP OF NUREG-1433, REVISION 1, BASES

B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.9 B PA1 Residual Heat Removal (RHR) Shutdown Cooling System—Cold Shutdown

BASES

the decay heat must be Removed for PA3

BACKGROUND PA2

Irradiated fuel in the shutdown reactor core generates heat during the decay of fission products and increases the temperature of the reactor coolant. This decay heat must be removed to maintain the temperature of the reactor coolant ≤ ~~200~~ 212 °F. ~~This decay heat removal is~~ in preparation for performing refueling ~~or maintenance~~ operations, or ~~for keeping~~ Maintaining PA3 the reactor in the Cold Shutdown condition.

The two redundant, manually controlled shutdown cooling ~~subsystems~~ (loops) PA3 of the RHR System provide decay heat removal. Each loop consists of two motor driven pumps, a heat exchanger, and associated piping and valves. Both loops have a common suction from the same recirculation loop. Each pump discharges the reactor coolant, after circulation through the respective heat exchanger, to the reactor via ~~the associated~~ (a) PA6 recirculation loop. The RHR heat exchangers transfer heat reactor water PA4 to the RHR Service Water System. DB2

reactor water PA4

APPLICABLE SAFETY ANALYSES

Decay heat removal by operation of the RHR System in the shutdown cooling mode is not required for mitigation (Ref. 1) of any event or accident evaluated in the safety analyses. Decay heat removal is, however, an important safety function that must be accomplished or core damage could result. ~~Although the RHR Shutdown Cooling System does not meet a specific criterion of the NRC Policy Statement, it was identified in the NRC Policy Statement as a significant contributor to risk reduction. Therefore, the RHR Shutdown Cooling System is retained as a Technical Specification.~~ S X1 4 X1 10 CFR 50.36(c)(2)(ii) (Ref. 2)

LCO DB1

Two RHR shutdown cooling subsystems are required to be OPERABLE~~, and when no recirculation pump is in operation, one RHR shutdown cooling subsystem must be in operation~~. CLB1 An OPERABLE RHR shutdown cooling subsystem consists of one OPERABLE RHR pump, one heat exchanger, one or two RHR service water pumps providing water to the heat exchanger, as required for temperature control and the associated piping and valves. The two subsystems PA3 have a common suction source and are allowed to have a common heat exchanger and common discharge piping. Thus, to meet the LCO, both RHR pumps

(continued)

(and two RHR service water pumps) PA3

BASES

LCO (continued)

in one loop or one RHR pump in each of the two loops must be OPERABLE. Since the piping and heat exchangers are passive components that are assumed not to fail, they are allowed to be common to both subsystems. In MODE 4, the RHR cross tie ~~valve (2E11-F010)~~ valves (10 MOV-20 and 10 RHR-09) DB1 may be opened to allow pumps in one loop to discharge through the opposite recirculation loop to make a complete subsystem. Additionally, each shutdown cooling subsystem is considered OPERABLE if it can be manually aligned ~~(remote or local)~~ from the control room or locally PA4 in the shutdown cooling mode for removal of decay heat. In MODE 4, one RHR shutdown cooling subsystem can provide the required cooling, but two subsystems are required to be OPERABLE to provide redundancy. Operation of one subsystem can maintain or reduce the reactor coolant temperature as required. However, to ensure adequate core flow to allow for accurate average reactor coolant temperature monitoring, nearly continuous operation is required.

~~Note 1 permits both RHR shutdown cooling subsystems to be shut down for a period of 2 hours in an 8 hour period.~~ CLB1 The Note ~~2~~ allows one RHR shutdown cooling subsystem to be inoperable for up to 2 hours for the performance of Surveillance tests. These tests may be on the affected RHR System or on some other plant system or component that necessitates placing the RHR ~~System~~ shutdown cooling subsystems PA3 in an inoperable status during the performance. This is permitted because the core heat generation can be low enough and the heatup rate slow enough to allow some changes to the RHR subsystems or other operations requiring ~~RHR flow interruption and~~ loss of shutdown cooling redundancy. CLB1

APPLICABILITY

In MODE 4, the RHR ~~Shutdown Cooling~~ System normally in operation PA3 ~~may be operated~~ is required to be OPERABLE so that it PA2 in the shutdown cooling mode to remove decay heat to maintain coolant temperature below ~~200~~ 212 PA2 °F. Otherwise, a recirculation pump is ~~required to be in operation~~ to circulate coolant to provide for temperature monitoring CLB1 PA3.

In MODES 1 and 2, and in MODE 3 with reactor steam dome pressure greater than or equal to the RHR cut in permissive pressure, this LCO is not applicable. Operation of the RHR System in the shutdown cooling mode is not allowed above this pressure because the RCS pressure may exceed the design pressure of the shutdown cooling piping. Decay heat removal at reactor pressures greater than or equal to the RHR cut in permissive pressure is typically accomplished by condensing

(continued)

BASES

APPLICABILITY (continued)

the steam in the main condenser. Additionally, in MODE 2 below this pressure, the OPERABILITY requirements for the Emergency Core Cooling Systems (ECCS) (LCO 3.5.1, "ECCS—Operating") do not allow placing the RHR shutdown cooling subsystem into operation.

The requirements for decay heat removal in MODE 3 below the cut in permissive pressure and in MODE 5 are discussed in LCO 3.4.8 7, "Residual Heat Removal (RHR) Shutdown Cooling System—Hot Shutdown"; LCO 3.9.8, "Residual Heat Removal (RHR)—High Water Level"; and LCO 3.9.9 8, "Residual Heat Removal (RHR)—Low Water Level." 7 PA5

ACTIONS

A Note has been provided to modify the ACTIONS related to RHR shutdown cooling subsystems. Section 1.3, Completion Times, specifies once a Condition has been entered, subsequent divisions, subsystems, components or variables expressed in the Condition, discovered to be inoperable or not within limits, will not result in separate entry into the Condition. Section 1.3 also specifies Required Actions of the Condition continue to apply for each additional failure, with Completion Times based on initial entry into the Condition. However, the Required Actions for inoperable shutdown cooling subsystems provide appropriate compensatory measures for separate inoperable shutdown cooling subsystems. As such, a Note has been provided that allows separate Condition entry for each inoperable RHR shutdown cooling subsystem.

<u>A.1</u>

With one of the two required RHR shutdown cooling subsystems inoperable, except as permitted by the LCO Note 2, the remaining subsystem is capable of providing the required decay heat removal. However, the overall reliability is reduced. Therefore, an alternate method of decay heat removal must be provided. With both RHR shutdown cooling subsystems inoperable, an alternate method of decay heat removal must be provided in addition to that provided for the initial RHR shutdown cooling subsystem inoperability. This re-establishes backup decay heat removal capabilities, similar to the requirements of the LCO. The 1 hour

(continued)

BASES

ACTIONS

A.1 (continued)

Completion Time is based on the decay heat removal function and the probability of a loss of the available decay heat removal capabilities. Furthermore, verification of the functional availability of these alternate method(s) must be reconfirmed every 24 hours thereafter. This will provide assurance of continued heat removal capability.

The required cooling capacity of the alternate method should be ensured by verifying (by calculation or demonstration) its capability to maintain or reduce temperature. Decay heat removal by ambient losses can be considered as, or contributing to, the alternate method capability. Alternate methods that can be used include (but are not limited to) ~~the Spent Fuel Pool Cooling System and the Reactor Water Cleanup System~~.

DB1
Condensate and Main Steam Systems, Reactor Water Cleanup System (by itself or using feed and bleed in combination with the Control Rod Drive System or Condensate System), or a combination of an RHR pump and safety/relief valve(s)

CLB1

~~B.1 and B.2~~

~~With no RHR shutdown cooling subsystem and no recirculation pump in operation, except as permitted by LCO Note 1, and until RHR or recirculation pump operation is re-established, an alternate method of reactor coolant circulation must be placed into service. This will provide the necessary circulation for monitoring coolant temperature. The 1 hour Completion Time is based on the coolant circulation function and is modified such that the 1 hour is applicable separately for each occurrence involving a loss of coolant circulation. Furthermore, verification of the functioning of the alternate method must be reconfirmed every 12 hours thereafter. This will provide assurance of continued temperature monitoring capability.~~

~~During the period when the reactor coolant is being circulated by an alternate method (other than by the required RHR Shutdown Cooling System or recirculation pump), the reactor coolant temperature and pressure must be periodically monitored to ensure proper function of the alternate method. The once per hour Completion Time is deemed appropriate.~~

(continued)

BASES (continued)

SURVEILLANCE REQUIREMENTS

SR 3.4.9.1 8 PA1

CLB1

Insert SR 3.4.8.1

CLB1

~~This Surveillance verifies that one RHR shutdown cooling subsystem or recirculation pump is in operation and circulating reactor coolant. The required flow rate is determined by the flow rate necessary to provide sufficient decay heat removal capability. The Frequency of 12 hours is sufficient in view of other visual and audible indications available to the operator for monitoring the RHR subsystem in the control room.~~

REFERENCES

None. 1. 10 CFR 50.36 (c)(2)(ii). X1

1. UFSAR, Chapter 14 DB2

CLB1

Insert SR 3.4.8.1

Verifying the correct alignment for manual, power operated, and automatic valves in the RHR shutdown cooling flow path provides assurance that the proper flow paths will exist for RHR operation. This SR does not apply to valves that are locked, sealed, or otherwise secured in position since these were verified to be in the correct position prior to locking, sealing, or securing. A valve that can be manually (from the control room or locally) aligned is allowed to be in a non-RHR shutdown cooling position provided the valve can be repositioned. This SR does not require any testing or valve manipulation; rather, it involves verification that those valves capable of potentially being mispositioned are in the correct position. This SR does not apply to valves that cannot be inadvertently misaligned, such as check valves.

The 31 day Frequency of this SR was derived from the Inservice Testing Program requirements for performing valve testing at least once every 92 days. The Frequency of 31 days is further justified because the valves are operated under procedural control. This Frequency has been shown to be acceptable through operating experience.

JAFNPP

IMPROVED STANDARD TECHNICAL SPECIFICATIONS (ISTS) CONVERSION

ITS: 3.4.8

Residual Heat Removal (RHR) Shutdown Cooling System - Cold Shutdown

JUSTIFICATION FOR DIFFERENCES (JFDs) FROM NUREG-1433, REVISION 1, BASES

RETENTION OF EXISTING REQUIREMENT (CLB)

CLB1 The requirement in ISTS LCO 3.4.9 (ITS LCO 3.4.8) associated with a recirculation pump or RHR shutdown cooling subsystem being in operation has been deleted. The requirement that two RHR shutdown cooling subsystems are Operable is considered acceptable. Requirements for RHR shutdown cooling subsystem operations are adequately controlled by JAFNPP plant operating procedures and policies. There are no explicit requirements in the CTS for RHR shutdown cooling subsystem operability or that a recirculation pump should be in operation in Cold Shutdown. However, ITS SR 3.4.9.1 will require the reactor coolant temperature to be monitored during RCS heatup and cooldown operations and RCS inservice leak and hydrostatic testing while SR 3.4.9.2 will require these readings to be taken during a planned startup where the reactor will achieve criticality. These Surveillance will necessitate placing either the recirculation pump or a RHR shutdown cooling subsystem in service to ensure circulation for temperature monitoring or cooling. The RHR shutdown cooling subsystems remove decay heat to reduce temperature of the reactor coolant to ≤ 212°F in preparation for performing Refueling or Cold Shutdown maintenance operations, or for maintaining the reactor stable in the Cold Shutdown conditions. Therefore, an RHR shutdown cooling subsystem will normally be in operation when a cooldown is in progress or to maintain reactor coolant temperature. Without it an alternate will be necessary to achieve the plant objective. During a heatup, the recirculation pumps will normally be in operation to ensure circulation and temperature monitoring. If the plant is maintaining plant conditions the plant will consider the situation but normally either the recirculation pump will be in service or an RHR shutdown cooling subsystem will in operation to ensure adequate temperature monitoring. This will ensure the plant objective is being met and to ensure the core is in a safe condition. JAFNPP is operated in Cold Shutdown, in a safe manner to ensure decay heat is removed so that no core damage could result, therefore the system is normally continuously operated during these conditions. The requirements in ISTS LCO 3.4.9 are not needed since the Surveillances in ITS 3.4.9 will clearly require temperature monitoring capability and this is accomplished with either one RHR shutdown cooling subsystem or one recirculation pump in operation.

Furthermore, to be consistent with this modification, the allowances in ISTS LCO 3.4.9 Note 1 that both RHR shutdown cooling subsystems and recirculation pumps may be removed from operation for up to 2 hours per 8 hour period, the requirements in ISTS 3.4.9 ACTION B for no RHR shutdown cooling subsystem and no recirculation pump in operation have been deleted. ISTS SR 3.4.9.1 (ITS SR 3.4.8.1) requires the verification that one RHR shutdown cooling subsystem or recirculation pump is in operation every 12 hours. Since the requirement to be in

RETENTION OF EXISTING REQUIREMENT (CLB)

CLB1 (continued)

operation has been deleted this SR has been revised to verify each required RHR shutdown cooling subsystem manual, power operated, and automatic valve in the flow path that is not locked, sealed, or otherwise secured in position, is aligned or can be aligned to its correct position every 31 days. The Frequency is consistent with similar Surveillances in other LCOs and is considered adequate for this condition. In Cold Shutdown, the RHR shutdown cooling system is of prime focus of plant operations and since the controls for this system are in the control room the Frequency is considered adequate. This Frequency is consistent with the same type of Surveillance in other LCOs (e.g., ITS 3.5.1).

PLANT SPECIFIC WORDING PREFERENCE OR MINOR EDITORIAL IMPROVEMENT (PA)

PA1 NUREG-1433 Specification 3.4.5, "RCS Pressure Isolation Valve (PIV) Leakage," not incorporated in ITS. Subsequent ITS Specifications and Bases have been renumbered accordingly.

PA2 The Bases have been revised to be consistent with changes made to the Specifications.

PA3 Editorial changes have been made for enhanced clarification, correction, or improvement with no change in intent.

PA4 The Bases have been modified to reflect plant specific terminology.

PA5 The correct LCO number has been included.

PA6 Editorial change made to ensure the allowances in the LCO Bases do not conflict with the description in the Background.

PLANT SPECIFIC DIFFERENCE IN DESIGN OR DESIGN BASIS (DB)

DB1 The Bases have been revised to reflect the JAFNPP specific design. In addition, a clarification regarding support systems necessary for RHR shutdown cooling system operability has been added since the ITS does not include a specific RHR Service Water System Specification for MODE 4.

DB2 The Bases have been modified to reflect JAFNPP specific references.

DIFFERENCE BASED ON APPROVED TRAVELER (TA)

None

DIFFERENCE BASED ON PENDING TRAVELER (TP)

None

DIFFERENCE FOR OTHER REASONS THAN ABOVE (X)

X1 NUREG-1433, Revision 1, Bases reference to "the NRC Policy Statement" has been replaced with 10 CFR 50.36(c)(2)(ii), in accordance with 60 FR 36953 effective August 18, 1995.

JAFNPP

IMPROVED STANDARD TECHNICAL SPECIFICATIONS (ISTS) CONVERSION

ITS: 3.4.8

Residual Heat Removal (RHR) Shutdown Cooling System - Cold Shutdown

RETYPED PROPOSED IMPROVED TECHNICAL SPECIFICATIONS (ITS) AND BASES

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.8 Residual Heat Removal (RHR) Shutdown Cooling System - Cold Shutdown

LCO 3.4.8 Two RHR shutdown cooling subsystems shall be OPERABLE.

-------------------------NOTE-------------------------
One RHR shutdown cooling subsystem may be inoperable for up to 2 hours for the performance of Surveillances.

APPLICABILITY: MODE 4.

ACTIONS

----------------------------------NOTE----------------------------------
Separate Condition entry is allowed for each shutdown cooling subsystem.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or two RHR shutdown cooling subsystems inoperable.	A.1 Verify an alternate method of decay heat removal is available for each inoperable RHR shutdown cooling subsystem.	1 hour <u>AND</u> Once per 24 hours thereafter

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.4.8.1	Verify each RHR shutdown cooling subsystem manual, power operated, and automatic valve in the flow path that is not locked, sealed, or otherwise secured in position, is aligned or can be aligned to its correct position.	31 days

B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.8 Residual Heat Removal (RHR) Shutdown Cooling System—Cold Shutdown

BASES

BACKGROUND

Irradiated fuel in the shutdown reactor core generates heat during the decay of fission products and increases the temperature of the reactor coolant. This decay heat must be removed to maintain the temperature of the reactor coolant ≤ 212°F in preparation for performing refueling operations, or the decay heat must be removed for maintaining the reactor in the Cold Shutdown condition.

The two redundant, manually controlled shutdown cooling subsystems (loops) of the RHR System provide decay heat removal. Each loop consists of two motor driven pumps, a heat exchanger, and associated piping and valves. Both loops have a common suction from the same reactor water recirculation loop. Each pump discharges the reactor coolant, after circulation through the respective heat exchanger, to the reactor via a reactor water recirculation loop. The RHR heat exchangers transfer heat to the RHR Service Water System.

APPLICABLE SAFETY ANALYSES

Decay heat removal by operation of the RHR System in the shutdown cooling mode is not required for mitigation of any event or accident evaluated in the safety analyses (Ref. 1). Decay heat removal is, however, an important safety function that must be accomplished or core damage could result. The RHR Shutdown Cooling System meets Criterion 4 of 10 CFR 50.36(c)(2)(ii) (Ref. 2).

LCO

Two RHR shutdown cooling subsystems are required to be OPERABLE. An OPERABLE RHR shutdown cooling subsystem consists of one OPERABLE RHR pump, one heat exchanger, one or two RHR service water pumps providing water to the heat exchanger, as required for temperature control, and the associated piping and valves. The two subsystems have a common suction source and are allowed to have a common heat exchanger and common discharge piping. Thus, to meet the LCO, both RHR pumps in one loop (and two RHR service water

(continued)

BASES

LCO (continued)

pumps) or one RHR pump in each of the two loops must be OPERABLE. Since the piping and heat exchangers are passive components that are assumed not to fail, they are allowed to be common to both subsystems. In MODE 4, the RHR cross tie valves (10MOV-20 and 10RHR-09) may be opened to allow pumps in one loop to discharge through the opposite recirculation loop to make a complete subsystem. Additionally, each shutdown cooling subsystem is considered OPERABLE if it can be manually aligned (from the control room or locally) in the shutdown cooling mode for removal of decay heat. In MODE 4, one RHR shutdown cooling subsystem can provide the required cooling, but two subsystems are required to be OPERABLE to provide redundancy. Operation of one subsystem can maintain or reduce the reactor coolant temperature as required. However, to ensure adequate core flow to allow for accurate average reactor coolant temperature monitoring, nearly continuous operation is required.

The Note allows one RHR shutdown cooling subsystem to be inoperable for up to 2 hours for the performance of Surveillance tests. These tests may be on the affected RHR System or on some other plant system or component that necessitates placing the RHR shutdown cooling subsystems in an inoperable status during the performance. This is permitted because the core heat generation can be low enough and the heatup rate slow enough to allow some changes to the RHR shutdown cooling subsystems or other operations requiring loss of redundancy.

APPLICABILITY

In MODE 4, the RHR System is required to be OPERABLE so that it may be operated in the shutdown cooling mode to remove decay heat to maintain coolant temperature below 212°F. Otherwise, a recirculation pump is normally in operation to circulate coolant to provide for temperature monitoring.

In MODES 1 and 2, and in MODE 3 with reactor steam dome pressure greater than or equal to the RHR cut in permissive pressure, this LCO is not applicable. Operation of the RHR System in the shutdown cooling mode is not allowed above this pressure because the RCS pressure may exceed the design pressure of the shutdown cooling piping. Decay heat removal at reactor pressures greater than or equal to the RHR cut in permissive pressure is typically accomplished by condensing the steam in the main condenser. Additionally, in MODE 2

(continued)

BASES

APPLICABILITY (continued)

below this pressure, the OPERABILITY requirements for the Emergency Core Cooling Systems (ECCS) (LCO 3.5.1, "ECCS – Operating") do not allow placing the RHR shutdown cooling subsystem into operation.

The requirements for decay heat removal in MODE 3 below the cut in permissive pressure and in MODE 5 are discussed in LCO 3.4.7, "Residual Heat Removal (RHR) Shutdown Cooling System – Hot Shutdown"; LCO 3.9.7, "Residual Heat Removal (RHR) – High Water Level"; and LCO 3.9.8, "Residual Heat Removal (RHR) – Low Water Level."

ACTIONS

A Note has been provided to modify the ACTIONS related to RHR shutdown cooling subsystems. Section 1.3, Completion Times, specifies once a Condition has been entered, subsequent divisions, subsystems, components or variables expressed in the Condition, discovered to be inoperable or not within limits, will not result in separate entry into the Condition. Section 1.3 also specifies Required Actions of the Condition continue to apply for each additional failure, with Completion Times based on initial entry into the Condition. However, the Required Actions for inoperable shutdown cooling subsystems provide appropriate compensatory measures for separate inoperable shutdown cooling subsystems. As such, a Note has been provided that allows separate Condition entry for each inoperable RHR shutdown cooling subsystem.

<u>A.1</u>

With one of the two required RHR shutdown cooling subsystems inoperable, except as permitted by the LCO Note, the remaining subsystem is capable of providing the required decay heat removal. However, the overall reliability is reduced. Therefore, an alternate method of decay heat removal must be provided. With both RHR shutdown cooling subsystems inoperable, an alternate method of decay heat removal must be provided in addition to that provided for the initial RHR shutdown cooling subsystem inoperability. This re-establishes backup decay heat removal capabilities, similar to the requirements of the LCO. The 1 hour Completion Time is based on the decay heat removal function

(continued)

BASES

ACTIONS

A.1 (continued)

and the probability of a loss of the available decay heat removal capabilities. Furthermore, verification of the functional availability of these alternate method(s) must be reconfirmed every 24 hours thereafter. This will provide assurance of continued heat removal capability.

The required cooling capacity of the alternate method should be ensured by verifying (by calculation or demonstration) its capability to maintain or reduce temperature. Decay heat removal by ambient losses can be considered as, or contributing to, the alternate method capability. Alternate methods that can be used include (but are not limited to) the Condensate and Main Steam Systems, Reactor Water Cleanup System (by itself or using feed and bleed in combination with the Control Rod Drive System or Condensate System), or a combination of an RHR pump and safety/relief valve(s).

SURVEILLANCE REQUIREMENTS

SR 3.4.8.1

Verifying the correct alignment for manual, power operated, and automatic valves in the RHR shutdown cooling flow path provides assurance that the proper flow paths will exist for RHR operation. This SR does not apply to valves that are locked, sealed, or otherwise secured in position since these were verified to be in the correct position prior to locking, sealing, or securing. A valve that can be manually (from the control room or locally) aligned is allowed to be in a non-RHR shutdown cooling position provided the valve can be repositioned. This SR does not require any testing or valve manipulation; rather, it involves verification that those valves capable of potentially being mispositioned are in the correct position. This SR does not apply to valves that cannot be inadvertently misaligned, such as check valves.

The 31 day Frequency of this SR was derived from the Inservice Testing Program requirements for performing valve testing at least once every 92 days. The Frequency of 31 days is further justified because the valves are operated under procedural control. This Frequency has been shown to be acceptable through operating experience.

(continued)

BASES (continued)

REFERENCES

1. UFSAR, Chapter 14.
2. 10 CFR 50.36(c)(2)(ii).

JAFNPP

IMPROVED STANDARD TECHNICAL SPECIFICATIONS (ISTS) CONVERSION

ITS: 3.4.9

RCS Pressure and Temperature (P/T) Limits

MARKUP OF CURRENT TECHNICAL SPECIFICATIONS (CTS)

DISCUSSION OF CHANGES (DOCs) TO THE CTS

NO SIGNIFICANT HAZARDS CONSIDERATION (NSHC) FOR LESS RESTRICTIVE CHANGES

MARKUP OF NUREG-1433, REVISION 1, SPECIFICATION

JUSTIFICATION FOR DIFFERENCES (JFDs) FROM NUREG-1433, REVISION 1

MARKUP OF NUREG-1433, REVISION 1, BASES

JUSTIFICATION FOR DIFFERENCES (JFDs) FROM NUREG-1433, REVISION 1, BASES

RETYPED PROPOSED IMPROVED TECHNICAL SPECIFICATIONS (ITS) AND BASES

JAFNPP

IMPROVED STANDARD TECHNICAL SPECIFICATIONS (ISTS) CONVERSION

ITS: 3.4.9

RCS Pressure and Temperature (P/T) Limits

MARKUP OF CURRENT TECHNICAL SPECIFICATIONS (CTS)

specification 3.4.9

A1

3.6 LIMITING CONDITIONS FOR OPERATION

3.6 REACTOR COOLANT SYSTEM

Applicability:

Applies to the operating status of the Reactor Coolant System.

Objective:

To assure the integrity and safe operation of the Reactor Coolant System.

Specification:

RCS A. Pressurization (e) and Thermal (Temperature) Limits A1

[LCO 3.4.9] 1. ~~Reactor Vessel Head Stud Tensioning~~

[SR 3.4.9.6] The reactor vessel head bolting studs shall not be under tension unless the temperatures of the reactor vessel flange and the reactor head flange are at least 90°F.

Add: Applicability: At all times A2

A1

[LCO 3.4.9] [SR 3.4.9.1] 2. ~~In-Service Hydrostatic and Leak Tests~~ A1

During in-service hydrostatic or leak testing the Reactor Coolant System pressure and temperature shall be on or to the right of curve A shown in Figure ~~3.6-1 Part 1 or 2~~ 3.4.9-1 or 3.4.9-2 for the flange and the beltline region, and on or to the right of curve A_{NB} for the non-beltline regions, and on or to the right of curve A_{BH} for the bottom head region. The maximum temperature change during any one hour period shall be:

LA5

4.6 SURVEILLANCE REQUIREMENTS

4.6 REACTOR COOLANT SYSTEM

Applicability:

Applies to the periodic examination and testing requirements for the Reactor Coolant System.

Objective:

To determine the condition of the Reactor Coolant System and the operation of the safety devices related to it.

Specification:

RCS A. Pressurization (e) and Thermal (Temperature) Limits A1

1. ~~Reactor Vessel Head Stud Tensioning~~

[SR 3.4.9.6 SR 3.4.9.7 SR 3.4.9.8] When in the cold condition, the reactor vessel head flange and the reactor vessel flange temperatures shall be recorded A6

Add: SR 3.4.9.8 Note A8

[SR 3.4.9.8] a. Every 12 hours when the reactor vessel head flange is ≤120°F and the studs are tensioned.

Add: SR 3.4.9.7 Note

[SR 3.4.9.7] b. Every 30 minutes when the reactor vessel head flange is ≤100°F and the studs are tensioned.

[SR 3.4.9.6] [SR 3.4.9.6 NOTE] c. Within 30 minutes prior to and every 30 minutes during tensioning of reactor vessel head bolting studs. A6

verified within limits

2. In-Service Hydrostatic and Leak Tests

[SR 3.4.9.1] During hydrostatic and leak testing the Reactor Coolant System pressure and temperature shall be ~~recorded~~ every [SR 3.4.9.1 NOTE] 30 minutes until two consecutive temperature readings are within 5°F of each other. LA1

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RAI 3.4.9-6

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Specification 3.4.9 A1

~~3.6 (cont'd)~~ ~~4.6 (cont'd)~~

[SR 3.4.9.1]
a. ≤20°F when to the left of curve C.

b. ≤100°F when on or to the right of curve C.

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3. ~~Non-Nuclear Heatup and Cooldown~~ A1

[LCO 3.4.9] LA5

During heatup by non-nuclear means (mechanical), cooldown following nuclear shutdown and low power physics tests the Reactor Coolant System pressure and temperature shall be on or to the right of the curve B shown in Figure 3.6-1 Part 1 or 2 3.4.9-1 or 3.4.9-2 for the flange, upper vessel and beltline regions, and on ~~or to the right of~~ curve B_{BH} for the bottom head region. The maximum temperature change during any one hour shall be ≤100°F. [SR 3.4.9.1]

4. ~~Core Critical Operation~~ A1

[LCO 3.4.9] LA5

During all modes of operation with a critical core (except for low power physics tests) the Reactor Coolant System pressure and temperature shall be at or to the right of the curve C shown in Figure 3.6-1 Part 1 or 2 A1. The maximum temperature change during any one hour shall be ≤100°F. [SR 3.4.9.1]

3. ~~Non-Nuclear Heatup and Cooldown~~ A1

[SR 3.4.9.1] [SR 3.4.9.1 NOTE]

During heatup by Non-Nuclear means, cooldown following nuclear shutdown and low power physics tests, the reactor coolant system pressure and temperature shall be recorded A6 every 30 minutes ~~until two consecutive temperature readings are within 5°F of each other.~~ LA1

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RAI 3.4.9-6

AMD 258

4. ~~Core Critical Operation~~ A1 A6 15 M1

During all modes of operation with a critical core (except for low power physics tests) the Reactor Coolant System pressure and temperature shall be ~~recorded~~ within 30 minutes prior to withdrawal of control rods to bring the reactor critical and every 30 minutes during heatup ~~until two consecutive temperature readings are within 5°F of each other.~~ LA1

[SR 3.4.9.1]

[SR 3.4.9.2]

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Specification 3.4.9

A1

Add CONDITION A "NOTE" A3

ITS

3.6 (cont'd)

[ACTION A]

5. With any of the limits of ~~3.6.A.1 through 3.6.A.4 above~~ the LCO exceeded, ~~either~~ in MODES 1, 2, or 3 A4

a. restore the temperature and/or pressure to within the limits within 30 minutes, perform an engineering evaluation to determine the effects of the out-of-limit condition on the structural integrity of the reactor coolant system, LA2 and determine that the reactor coolant system remains acceptable for continued operations in 72 hours M2 or

[ACTION B]

b. be in at least HOT SHUTDOWN within 12 hours and in COLD SHUTDOWN within the following 24 hours.

A1

6. ~~Idle Recirculation Loop Startup~~

Only required to be met in MODES 1, 2, 3 and 4 L1

[SR 3.4.9.3 Note 1]
[SR 3.4.9.5 Note 1]

~~When Reactor Coolant System temperature is > 140°F~~ an idle recirculation loop shall not be started unless:

[SR 3.4.9.3] a. The temperature differential between the reactor coolant system and the reactor vessel bottom head drain line is ≤ 145°F, and

[SR 3.4.9.5] b. When both loops are idle, the temperature difference between the reactor coolant system and the idle loop to be started is ≤ 50°F, or

c. When only one loop is idle, the temperature difference between the idle loop ~~and the operating loop~~ RPV A7 is ≤ 50°F.

add SR 3.4.9.4 L2

Add ACTION C M4

4.6 (cont'd)

5. Not Used

A1

6. ~~Idle Recirculation Loop Startup~~

Within ~~30~~ 15 minutes prior to startup of an idle loop: M3

[SR 3.4.9.3] a. The differential temperature between the reactor coolant system and the reactor vessel bottom head drain line shall be ~~recorded~~, and A6 LA3

[SR 3.4.9.5] b. When both loops are idle, the differential temperature between the reactor coolant system and the idle loop to be started shall be ~~recorded~~; or

[SR 3.4.9.5] c. When only one loop is idle, the temperature differential between the idle loop and the ~~operating loop~~ A7 RPV shall be ~~recorded~~. A6

add SR 3.4.9.4 L2

Add SR 3.4.9.3 "Note"
SR 3.4.9.5 "Note" L1

Specification 3.4.9

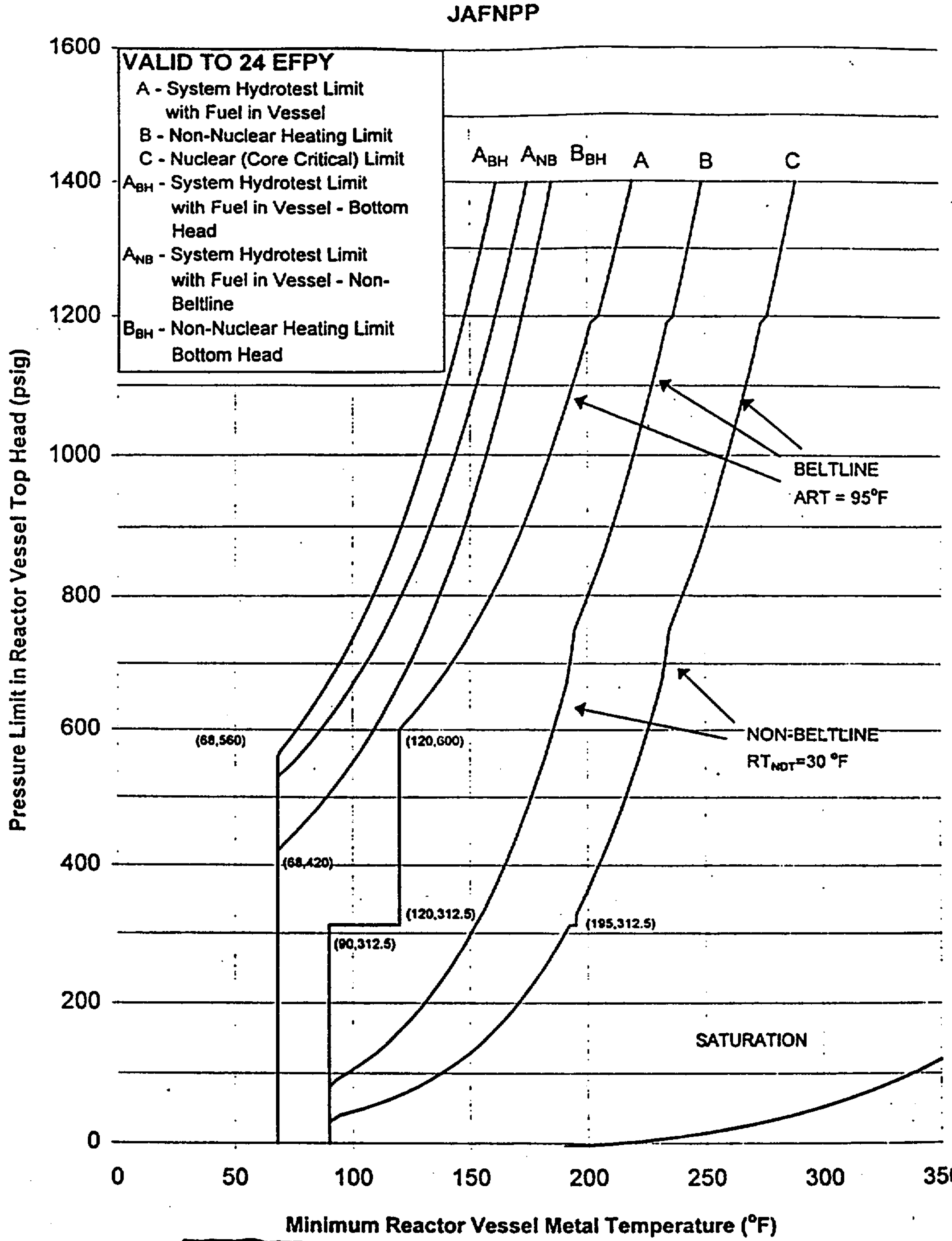

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Figure ~~3.6-1 Part 1~~ 3.4.9-1 (A1) Reactor Vessel Pressure-Temperature Limits Through 24 EFPY

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Specification 3.4.9

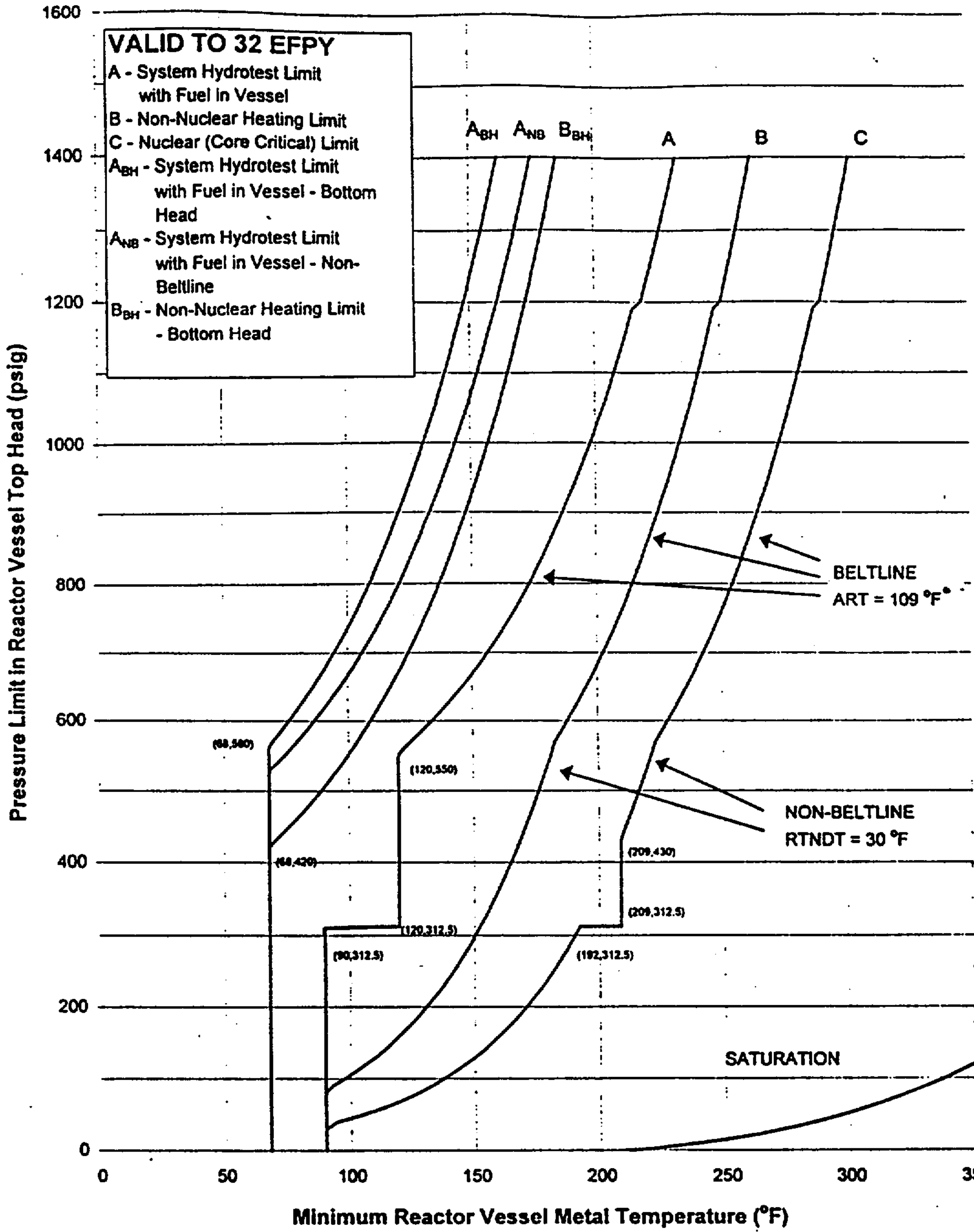

Figure ~~3.6-1 Part 2~~ 3.4.9-2 A1 Reactor Vessel Pressure-Temperature Limits Through 32 EFPY

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IMPROVED STANDARD TECHNICAL SPECIFICATIONS (ISTS) CONVERSION

ITS: 3.4.9

RCS Pressure and Temperature (P/T) Limits

DISCUSSION OF CHANGES (DOCs) TO THE CTS

ADMINISTRATIVE CHANGES

A1 In the conversion of the James A. FitzPatrick Nuclear Power Plant (JAFNPP) 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. Editorial changes, reformatting, and revised numbering are adopted to make ITS consistent with the conventions in NUREG-1433, "Standard Technical Specifications, General Electric Plants, BWR/4", Revision 1 (i.e., Improved Standard Technical Specifications (ISTS)).

A2 CTS 3.6.A does not state any Applicability requirements. ITS 3.4.9 is Applicable "At all times". Because the CTS does not specifically state Applicability requirements, and the limitations imposed apply at all times, it can be implied that the Specification is also Applicable "At all times." Since no technical requirements are altered, this change is administrative and has no adverse impact on safety. RAI 3.4.9-4

A3 CTS 3.6.A.5.a is revised by adding a NOTE (ITS 3.4.9 Condition A Note) which requires that a determination be made whether the RCS is acceptable for continued operation whenever the Condition is entered, regardless of whether compliance with the LCO is restored. This change only provides clarification, because CTS 3.6.A.5.a already contains this requirement. Since no technical requirements are altered, this change is administrative and has no adverse impact on safety.

A4 CTS 3.6.A.5 provides actions appropriate for placing the facility in a condition outside the MODE(S) of Applicability when the Applicability is MODES 1, 2, and 3. Since certain PT limits apply even when not in MODES 1, 2, and 3, Action C was added (refer to DOC M4). To clarify the use and application of applying the appropriate action depending on the MODE of operation, the specific clarification "in MODES 1, 2, and 3" is added. No technical requirements are altered, this change is administrative and has no adverse impact on safety. RAI 3.4.9-4

A5 Not used. AMD 258

ADMINISTRATIVE CHANGES

RAI 3.4.9-7

A6 The requirement to record the results in CTS 4.6.A.1, 4.6.A.2, 4.6.A.3, 4.6.A.4, 4.6.A.6.a, 4.6.A.6.b, and 4.6.A.6.c (ITS SRs 3.4.9.1, 3.4.9.2, 3.4.9.3, 3.4.9.5, 3.4.9.6, 3.4.9.7 and 3.4.9.8) is proposed to be deleted and the requirement will be to verify the associated parameters are within the specified limits. This requirement duplicates the requirements of 10 CFR 50 Appendix B, Section XVII (Quality Assurance Records): maintain records of activities affecting quality, including the results of tests (i.e., Technical Specification Surveillances). Compliance with 10 CFR 50 Appendix B is required by the JAFNPP Operating License. The details of the regulations within the Technical Specifications are repetitious and unnecessary. Therefore, retaining the requirement to perform the associated Surveillances (verifying the specified limits are met) and eliminating the details from Technical Specifications that are found in 10 CFR 50 Appendix B is considered a presentation preference, which is administrative.

A7 Thermal stresses on vessel components are dependent upon the temperature difference between the idle loop coolant and the RPV coolant. ITS SR 3.4.9.5 ensures the temperature difference between the idle loop and the RPV coolant is acceptable. The requirements to monitor the temperature difference between an idle loop and an operating loop (CTS 3/4.6.A.6.c) are unnecessary and are deleted since they are redundant to the loop-to-coolant requirement of ITS SR 3.4.9.5. However, in accordance with procedures and as discussed in the Bases for ITS SR 3.4.9.4, the loop-to-coolant temperature check may use the operating loop temperature as representative of "coolant temperature".

A8 A Note has been added to CTS 4.6.A.1.a and 4.6.A.1.b (Note to ITS SR 3.4.9.8 and 3.4.9.7, respectively) which clarifies that the Surveillances are not required to be performed until 30 minutes after RCS temperature ≤ 100°F or 120°F, respectively. These requirements are consistent with the CTS requirements. The Frequency of the CTS requirements are every 12 hours (CTS 4.6.A.1.a) or 30 minutes (CTS 4.6.A.1.b) when the reactor vessel head flange falls below the prescribed limit. Therefore, the first required Surveillance is 12 hours or 30 minutes after the specified temperature is reached. These requirements are consistent with the proposed Note therefore this change is considered administrative. This change is consistent with NUREG-1433, Revision 1. AMD 267

A9 Not used.

TECHNICAL CHANGES - MORE RESTRICTIVE

M1 CTS 4.6.A.4 requires that RCS pressure and temperature be recorded within 30 minutes prior to withdrawal of control rods to bring the reactor critical. ITS SR 3.4.9.2 requires this verification to be performed within 15 minutes prior to control rod withdrawal for the purpose of achieving criticality. This Frequency is closer to when the control rods will actually be withdrawn and will help ensure the specified are limits are met. Since the time is limited, this change is considered more restrictive but necessary to ensure the specified parameters are within limits prior to control rod withdrawal where the reactor has a potential of becoming critical. This change is consistent with NUREG-1433, Revision 1.

M2 CTS 3.6.A.5 requires that, in the event the RCS pressure and temperature limits are exceeded, it be determined that the RCS remains acceptable for continued operation. There is no Completion Time associated with this requirement. ITS 3.4.9 Required Action A.2 Completion Time requires that this determination be made in 72 hours. This change imposes a time constraint where one does not exist, and is therefore more restrictive but necessary to ensure prompt action is taken to verify the RCPB is acceptable for continuous Operation.

M3 CTS 4.6.A.6 requires that certain RCS differential temperature measurements be recorded within 30 minutes prior to startup of an idle recirculation loop. ITS SRs 3.4.9.3 and 3.4.9.5 require that these differential temperature measurements be verified within 15 minutes prior to startup of an idle recirculation loop. This Frequency is closer to when the pumps will actually be started and therefore will help ensure the specified limits are met prior to pump startup. Since the time is limited, this change is considered more restrictive but necessary to ensure the temperatures are within limits prior to a startup of an idle pump. This change is consistent with NUREG-1433, Revision 1.

M4 CTS 3.6.A is revised by adding a new action (ITS 3.4.9 ACTION C), which requires that action be initiated immediately to restore the parameters to within limits, and a determination be made as to whether the RCS is acceptable for continued operation prior to entering MODE 2 or 3. ITS 3.4.9 ACTION C is Applicable at all times other than in MODES 1, 2, and 3. This change imposes additional requirements and is considered more restrictive but necessary for protection of the RCPB.

TECHNICAL CHANGES - LESS RESTRICTIVE (GENERIC)

LA1 The requirement in CTS 4.6.A.2, 4.6.A.3, and 4.6.A.4 that specifies the criteria for ending the surveillances (performed until two consecutive temperature readings are within 5°F of each other) is proposed to be relocated to the Bases. Requirements of SR 3.4.9.1 provide adequate assurance that heatup and cooldown of the RCS will be monitored and maintained within limits. As a result, the manner in which JAFNPP determines that a heatup or cooldown has been terminated is not necessary for ensuring limits are met. Therefore, the relocated criteria for determining when a heatup or cooldown has terminated is not required to be in the ITS to provide adequate protection of the public health and safety. Changes to the Bases will be controlled by the provisions of the proposed Bases Control Program described in Chapter 5 of the Technical Specifications.

LA2 The details in CTS 3.6.A.5.a to perform an "engineering evaluation" to determine the effects of the out-of-limit condition on the structural integrity of the RCS are proposed to be relocated to the Bases. The requirement in ITS 3.4.9 Required Actions A.2 and C.2 to determine whether the RCS is acceptable for continued operation is adequate to ensure the proper analysis is performed. Therefore, the relocated details are not required to be in the ITS to provide adequate protection of the public health and safety. Changes to the Bases will be controlled by the provisions of the proposed Bases Control Program described in Chapter 5 of the Technical Specifications.

LA3 The method defined in CTS 4.6.A.6 to evaluate the temperature differential using the temperature at the reactor vessel bottom head "drain line" is proposed to be relocated to the Bases. The requirement in ITS SR 3.4.9.3 to verify the difference between the bottom head coolant temperature and the reactor pressure vessel (RPV) coolant temperature is within the limits is adequate. As such, these details are not required to be in the ITS to provide adequate protection of public health and safety. Changes to the Bases will be controlled by the provisions of the proposed Bases Control Program described in Chapter 5 of the Technical Specifications.

LA4 Not used.

LA5 The specific requirements in CTS 3.6.A.2, 3.6.A.3, and 3.6.A.4 that operation be on or to the right of the curves of Figure 3.6-1 Part 1 or 2 are proposed to be relocated to the Bases. These details are not necessary to ensure that P/T limits are met. The requirements to maintain the P/T limits in accordance with the Figures are still maintained in ITS 3.4.9 and SR 3.4.9.1. Therefore, the relocated requirements are not required to be in the ITS to provide adequate

RAI 3.4.9-6 AMD267

TECHNICAL CHANGES - LESS RESTRICTIVE (GENERIC)

LA5 (continued)

protection of the public health and safety. Changes to the Bases will be controlled by the provisions of the proposed Bases Control Program described in Chapter 5 of the Technical Specifications. RAI 3.4.9-6

TECHNICAL CHANGES - LESS RESTRICTIVE (SPECIFIC)

L1 CTS 4.6.A.6 requires verification that the temperature differential between the RCS and the reactor vessel bottom head, and between the RCS and an idle recirculation loop, are within limits prior to startup of the idle recirculation loop. CTS 3.6.A.6 specifies that this is only to be met when Reactor Coolant System temperature is > 140°F. These requirements are modified by a Note which states that these P/T verifications are only required to be met in MODES 1, 2, 3, and 4 during recirculation pump startup (Note to SRs 3.4.9.3 and 3.4.9.5). Since the overall Applicability of the Specification is reduced (Surveillance is no longer required in MODE 5 with temperatures > 140°F), this change is a relaxation of requirements and is less restrictive. This change is acceptable because in MODE 5, the recirculation pumps are rarely placed in operation, and the overall stress on limiting components is lower. Therefore, the differential temperature limits are not required. This change is consistent with NUREG-1433, Revision 1.

L2 CTS 3.6.A.6.a requires the temperature differential between the reactor coolant system and the reactor vessel bottom head drain line be ≤ 145°F during a recirculation pump startup (CTS 3.6.A.6). ITS 3.4.9 provides the option to verify the active recirculation pump flow exceeds 40% of rated pump flow or the active recirculation pump has been operating below 40% rated flow for a period no longer than 30 minutes. ITS SR 3.4.9.4 has been added which provides this allowance. A Note 2 has been added to the requirements of CTS 3.6.A.6.a (ITS SR 3.4.9.3) which provides the option to perform ITS SR 3.4.9.4. Similarly, a Note 2 has been added to proposed ITS SR 3.4.9.4 which provides the allowance to evaluate the temperature differential in SR 3.4.9.3. RAI 3.4-GEN

This change is necessary to avoid an unnecessary plant shutdown to restart an idle recirculation pump when the bottom head drain line temperature indicating channel is inoperable, the drain line is plugged, or if the drain flow is low. The requirement to ensure the differential temperature between the bottom head drain line and the reactor coolant is within limits has been established to assure avoidance of a thermal over stress condition to the Control Rod Drive (CRD) stub tubes and in-core housing welds by sweeping hot water across these relatively cooler RAI 3.4-GEN

TECHNICAL CHANGES - LESS RESTRICTIVE (SPECIFIC)

L2 (continued)

vessel structures and associated components. The temperature in the bottom head region is usually measured by monitoring the temperature of flow being drawn out from the bottom head drain line. In the past, JAFNPP has experienced the problem of the bottom head drain line being plugged with debris. In order to have a good temperature reading, it is necessary to have sufficient flow through the bottom head drain line.

General Electric has determined an alternate method to the verification of the differential temperature between the bottom head coolant temperature and the reactor pressure vessel (RPV) coolant temperature prior to starting a recirculation pump. This alternative is to verify the active recirculation pump flow exceeds 40% of rated pump flow or the active recirculation pump has been operating below 40% rated flow for a period no longer than 30 minutes prior to starting the recirculation pump (GE-NE-208-04-1292, Evaluation of Idle Recirculation loop Restart without Vessel Bottom Temperature Indication for JAFNPP Nuclear Power Plant). The GE alternative is based on an evaluation that collected data of startup testing at a BWR/3 and BWR/4 plant. The results from operating BWR plant provides the basis that if the above restart conditions are met stratification in the lower plenum region will be avoided since there will either be sufficient mixing in the lower plenum or there wasn't time for the condition to develop.

RAI 3.4-GEN

When the active recirculation pump flow exceeds 40% of its rated pump flow under one pump operating condition, the evidence shows, analytically and experimentally, that there is sufficient mixing to prevent the thermal stratification in the lower plenum region. In order, to achieve this flow rate reactor power must be above 25% RTP to clear the feedwater flow interlock at approximately 20% flow. At 25% RTP and 40% recirculation pump flow, a GE steady state hydraulic computer code predicts the core flow for JAFNPP to be at 35% of rated. Essentially all of this flow is predicted to be coming down the jet pump diffusers on the active loop into the lower plenum to provide a good mixing effect.

Test data concerning lower plenum temperature was obtained during natural circulation startup test of a BWR/4 to confirm that this core flow is sufficient to ensure that adequate lower plenum mixing takes place under one-loop operation. A stratified condition existed for very low power levels but it was swept out when core flow reached about 20% of rated (corresponding to natural circulation at 5% RTP). This indicated that 20% core flow was enough to sweep out the stratified cold water. The 35% core flow, as predicted for JAFNPP, is well above the

TECHNICAL CHANGES - LESS RESTRICTIVE (SPECIFIC)

L2 (continued)

20% threshold, and it should clearly avoid stratification. In addition, during a hot restart at a BWR/3 with both recirculation pumps at 20% speed and reactor power at < 1% RTP, it was confirmed that a temperature difference of < 110°F was maintained between the bottom head drain line and the saturation temperature of the vessel. The core flow at this condition was estimated to be 20%. The results at the BWR/3 is also considered to be conservative for the BWR/4 design at JAFNPP. Test data from the BWR/3 plant also indicated that stratification does not occur immediately upon low flow conditions. This test data showed that stratification did not occur until an hour after a main turbine trip and recirculation pump trip. On this basis, it is believed that restoring the 40% recirculation pump flow condition within 30 minutes is an acceptable criteria for avoiding lower plenum stratification and allowing subsequent restart of the second loop.

RAI 3.4-GE2

JAFNPP has reviewed GE report to evaluate whether the proposed criteria for JAFNPP would be acceptable. The original JAFNPP startup test data for a single recirculation pump operation was examined to determine how far away the plant was from stratification at various flow rates. The results indicate that with a recirculation pump flow rate of 27.5% (resulting core flow of ≤ 20 Mlb/hr) and power levels between 25% and 44% of RTP, the resulting temperature difference was ≤ 57°F. In addition, startup test data was reviewed for three tests in which both recirculation pumps were tripped. A maximum differential temperature of 44°F was achieved and this was observed after more than two hours after the pump trip. The results of the JAFNPP review is included in JAF-RPT-RWR-02076, Rev 0 (Verification of Alternative Operation Conditions for Idle Recirculation Loop Restart without Vessel Bottom Temperature Indication). There the proposed alternatives are considered acceptable for JAFNPP.

TECHNICAL CHANGES - RELOCATIONS

None

JAFNPP

IMPROVED STANDARD TECHNICAL SPECIFICATIONS (ISTS) CONVERSION

ITS: 3.4.9

RCS Pressure and Temperature (P/T) Limits

NO SIGNIFICANT HAZARDS CONSIDERATION (NSHC) FOR LESS RESTRICTIVE CHANGES

TECHNICAL CHANGES - LESS RESTRICTIVE (SPECIFIC)

L1 CHANGE

New York Power Authority has evaluated the proposed Technical Specification change and has concluded that it does not involve a significant hazards consideration. Our conclusion is in accordance with the criteria set forth in 10 CFR 50.92. The bases for the conclusion that the proposed change does not involve a significant hazards consideration are discussed below.

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

   The proposed change does not involve any physical alteration of plant systems, structures or components, changes in parameters governing normal plant operation, or methods of operation. The proposed change relaxes the Applicability for verification that the temperature differential between the RCS and the reactor vessel bottom head, and between the RCS and an idle recirculation loop, are within limits prior to startup of the idle recirculation loop when Reactor Coolant System temperature is > 140°F (MODES 1, 2, 3, 4 and 5), to MODES 1, 2, 3, and 4. These differential temperatures are not assumed to initiate any accident. Therefore, this change will not increase the probability of an accident previously evaluated. In MODE 5, the recirculation pumps are rarely placed in operation and the resulting overall stress on limiting components is lower, and therefore, the differential temperature limits are not required. Therefore, this change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

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

   The proposed change does not involve any physical alteration of plant systems, structures or components, changes in parameters governing normal plant operation, or methods of operation. The proposed change relaxes the Applicability for verification of temperature differential prior to startup of an idle recirculation loop. Therefore, the possibility of a new or different kind of accident from any accident previously evaluated is not created.

TECHNICAL CHANGES - LESS RESTRICTIVE (SPECIFIC)

L1 CHANGE

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

   The proposed change relaxes the Applicability for verification of temperature differential prior to startup of an idle recirculation loop. The Specifications will continue to require that RCS pressure and temperature be maintained within analysis limits. Therefore, this change does not involve a significant reduction in a margin of safety.

TECHNICAL CHANGES - LESS RESTRICTIVE (SPECIFIC)

L2 CHANGE

E

The Licensee has evaluated the proposed Technical Specification change and has concluded that it does not involve a significant hazards consideration. Our conclusion is in accordance with the criteria set forth in 10 CFR 50.92. The bases for the conclusion that the proposed change does not involve a significant hazards consideration are discussed below.

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

   This change provides an alternate method to ensure the avoidance of thermal over stress condition to the Control Rod Drive (CRD) stub tubes and in-core housing welds during a recirculation pump startup. The current method is to ensure the temperature differential between the reactor coolant system and the reactor vessel bottom head drain line be ≤ 145°F prior to the recirculation pump startup. ITS 3.4.9 provides the option to verify the active recirculation pump flow exceeds 40% of rated pump flow or the active recirculation pump has been operating below 40% rated flow for a period no longer than 30 minutes. The method of evaluating the temperature difference does not influence any assumptions of a design bases accident. Therefore, this change will not significantly increase the probability of any accident previously evaluated. General Electric has provided an alternative to the verification of the differential temperature between the bottom head coolant temperature and the reactor pressure vessel (RPV) coolant temperature prior to starting a recirculation pump. This alternative is to verify the active recirculation pump flow exceeds 40% of rated pump flow or the active recirculation pump has been operating below 40% rated flow for a period no longer than 30 minutes prior to starting the recirculation pump (GE-NE-208-04-1292, Evaluation of Idle Recirculation loop Restart without Vessel Bottom Temperature Indication for JAFNPP Nuclear Power Plant). The GE alternative is based on an evaluation that collected data of startup testing at various BWR plants. The results from operating BWR plant provides the basis that if the above restart conditions are met stratification in the lower plenum region will be avoided. JAFNPP has reviewed this analysis and has evaluated similar data specific to JAFNPP. JAF-RPT-RWR-02076, Rev 0 confirms that the conclusions reached in the GE study applies to JAFNPP. Therefore the consequences of any previously evaluated accident will be the same using the existing method. Therefore, this change does not involve a significant increase in the consequences of an accident previously evaluated.

   RAI 3.4-GEN

NO SIGNIFICANT HAZARDS CONSIDERATION
ITS: 3.4.9 - RCS PRESSURE AND TEMPERATURE (P/T) LIMITS

TECHNICAL CHANGES - LESS RESTRICTIVE (SPECIFIC)

L2 CHANGE

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

   This change provides an alternate method to ensure the avoidance of thermal over stress condition to the Control Rod Drive (CRD) stub tubes and in-core housing welds during a recirculation pump startup. The proposed change does not involve any physical alteration of plant systems, structures or components, changes in parameters governing normal plant operation, or methods of operation. Therefore, the possibility of a new or different kind of accident from any accident previously evaluated is not created.

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

   This change provides an alternate method to ensure the avoidance of thermal over stress condition to the Control Rod Drive (CRD) stub tubes and in-core housing welds during a recirculation pump startup. The current method is to ensure the temperature differential between the reactor coolant system and the reactor vessel bottom head drain line be $\leq$ 145°F prior to the recirculation pump startup. ITS 3.4.9 provides the option to verify the active recirculation pump flow exceeds 40% of rated pump flow or the active recirculation pump has been operating below 40% rated flow for a period no longer than 30 minutes. General Electric has provided an alternative to the verification of the differential temperature between the bottom head coolant temperature and the reactor pressure vessel (RPV) coolant temperature prior to starting a recirculation pump. This alternative is to verify the active recirculation pump flow exceeds 40% of rated pump flow or the active recirculation pump has been operating below 40% rated flow for a period no longer than 30 minutes prior to starting the recirculation pump (GE-NE-208-04-1292, Evaluation of Idle Recirculation loop Restart without Vessel Bottom Temperature Indication for JAFNPP Nuclear Power Plant). The GE alternative is based on an evaluation that collected data of startup testing at various BWR plants. The results from operating BWR plant provides the basis that if the above restart conditions are met stratification in the lower plenum region will be avoided. JAFNPP has reviewed this analysis and has evaluated similar data specific to JAFNPP. JAF-RPT-RWR-02076, Rev 0 confirms that the conclusions reached in the GE study applies to JAFNPP. The alternative methods accomplishes the same function as the current method in avoidance of thermal over stress condition to the Control Rod Drive (CRD) stub tubes and in-core housing welds during a recirculation pump startup. Therefore, this change does not involve a significant reduction in a margin of safety.

   RAI 3.4-GEN

JAFNPP

IMPROVED STANDARD TECHNICAL SPECIFICATIONS (ISTS) CONVERSION

ITS: 3.4.9

RCS Pressure and Temperature (P/T) Limits

MARKUP OF NUREG-1433, REVISION 1
SPECIFICATION

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.10 RCS Pressure and Temperature (P/T) Limits 9 PA1

[3.6.A.1] [3.6.A.2] [3.6.A.3] [3.6.A.4] [3.6.A.6]

LCO 3.4.10 9 RCS pressure, RCS temperature, RCS heatup and cooldown rates, and the recirculation pump starting temperature requirements shall be maintained within ~~the~~ limits ~~specified in the PTLR~~. CLB1

[Doc A2] APPLICABILITY: At all times.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
[3.6.A.5.a] A. ---------NOTE--------- Required Action A.2 shall be completed if this Condition is entered. ---------------------- Requirements of the LCO not met in MODE~~S~~ 1, 2, ~~and~~ or 3. PA2	A.1 Restore parameter(s) to within limits. AND A.2 Determine RCS is acceptable for continued operation.	30 minutes 72 hours
[3.6.A.5.b] B. Required Action and associated Completion Time of Condition A not met.	B.1 Be in MODE 3. AND B.2 Be in MODE 4.	12 hours 36 hours

RAI 3.4.9-2

(continued)

Typ. All pages

ACTIONS (continued)

CONDITION	REQUIRED ACTION		COMPLETION TIME
[M4] C. ----------NOTE---------- Required Action C.2 shall be completed if this Condition is entered. ------------------------ Requirements of the LCO not met in other than MODES 1, 2, and 3.	C.1	Initiate action to restore parameter(s) to within limits.	Immediately
	AND		
	C.2	Determine RCS is acceptable for operation.	Prior to entering MODE 2 or 3.

RA 3.4.9-2

SURVEILLANCE REQUIREMENTS

	SURVEILLANCE		FREQUENCY
[4.6.A.2] [4.6.A.3] [3.6.A.2] [3.6.A.3] [3.6.A.4] INSERT SR-1	SR 3.4.10.1 9 PA1	--------------------NOTE-------------------- Only required to be performed during RCS heatup and cooldown operations and RCS inservice leak and hydrostatic testing. -- ~~Verify~~ a. RCS pressure, ~~RCS~~ and temperature~~, and RCS heatup and cooldown rates~~ are within the limits specified in ~~the PTLR~~. Figure 3.4.9-1 or Figure 3.4.9-2, as applicable. PA3	30 minutes LB1
[4.6.A.4]	SR 3.4.10.2 9 PA1	Verify RCS pressure and ~~RCS~~ temperature are within the criticality limits specified in ~~the PTLR~~. Figure 3.4.9-1 or Figure 3.4.9-2, as applicable. PA3 CLB1	Once within 15 minutes prior to control rod withdrawal for the purpose of achieving criticality

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(continued)

CLA1 Insert SR-1

b. RCS temperature change averaged over a one hour period is:

1. ≤ 100°F when the RCS pressure and temperature are on or to the right of curve C of Figure 3.4.9-1 or Figure 3.4.9-2, as applicable, during inservice leak and hydrostatic testing;

2. ≤ 20°F when the RCS pressure and temperature are to the left of curve C of Figure 3.4.9-1 or Figure 3.4.9-2, as applicable, during inservice leak and hydrostatic testing; and

3. ≤ 100°F during other heatup and cooldown operations.

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SURVEILLANCE REQUIREMENTS (continued)

	SURVEILLANCE	FREQUENCY
[4.6.A.6.a] [L1] [3.6.A.6.c] Insert SR 3.4.9.4 X1 [L2] SR 3.4.~~10~~ 9.3 PA1	X1 S ------NOTE------ 1. Only required to be met in MODES 1, 2, 3, and 4 ~~[with reactor steam dome pressure ≥ 25 psig]~~ DB1. ------ Verify the difference between the bottom head coolant temperature and the reactor pressure vessel (RPV) coolant temperature is ~~within the limits specified in the PTLR~~ ≤ 145 °F CLB1.	during recirculation pump startup. TA1 Insert Note 2 X1 Once within 15 minutes prior to each startup of a recirculation pump
[4.6.A.6.b] [4.6.A.6.c] [L1] SR 3.4.~~10~~ 9.~~4~~ 5 PA1	X1 ------NOTE------ Only required to be met in MODES 1, 2, 3, and ~~4~~. ------ Verify the difference between the reactor coolant temperature in the recirculation loop to be started and the RPV coolant temperature is ~~within the limits specified in the PTLR~~ ≤ 50 °F CLB1.	during recirculation pump startup. TA1 Once within 15 minutes prior to each startup of a recirculation pump
[4.6.A.1.c] [3.6.A.1] SR 3.4.~~10~~ 9.~~5~~ 6 PA1	X1 ------NOTE------ Only required to be performed when tensioning the reactor vessel head bolting studs. ------ Verify reactor vessel flange and head flange temperatures are ~~within the limits specified in the PTLR~~ ≥ 90 °F CLB1.	30 minutes

(continued)

Kl Insert Note 2

2. Note required to be performed if SR 3.4.9.4 is satisfied.

Kl Insert SR 3.4.9.4

SR 3.4.9.4	------------------NOTES------------------ 1. Only required to be met in MODES 1, 2, 3, and 4 during recirculation pump startup. 2. Not required to be met if SR 3.4.9.3 is satisfied. --- Verify the active recirculation pump flow exceeds 40% of rated pump flow or the active recirculation pump has been operating below 40% rated flow for a period no longer than 30 minutes.	Once within 15 minutes prior to each startup of a recirculation pump

RAI 3.4-Gen

REVISION E

SURVEILLANCE REQUIREMENTS (continued)

	SURVEILLANCE X1	FREQUENCY
[4.6.A.1.b]	SR 3.4.~~10~~ 9 PA1 .~~6~~ 7 --------------------NOTE-------------------- Not required to be performed until 30 minutes after RCS temperature ≤ ~~80~~ 100 °F ~~in MODE 4~~. CLB1 --- Verify reactor vessel flange and head flange temperatures are ~~within the limits specified in the PTLR~~ ≥ 90°F CLB1.	30 minutes with any reactor vessel stud tensioned
[4.6.A.1.a]	SR 3.4.~~10~~ 9 PA1 .~~7~~ 8 X1 --------------------NOTE-------------------- Not required to be performed until 12 hours after RCS temperature ≤ ~~100~~ 120 °F ~~in MODE 4~~. CLB1 --- Verify reactor vessel flange and head flange temperatures are ~~within the limits specified in the PTLR~~ ≥ 90°F CLB1.	12 hours

RAI 3.4.9-5

INSERT Figures 3.4.9-1 and 3.4.9-2 CLB1

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Insert Figure 3.4.9-1

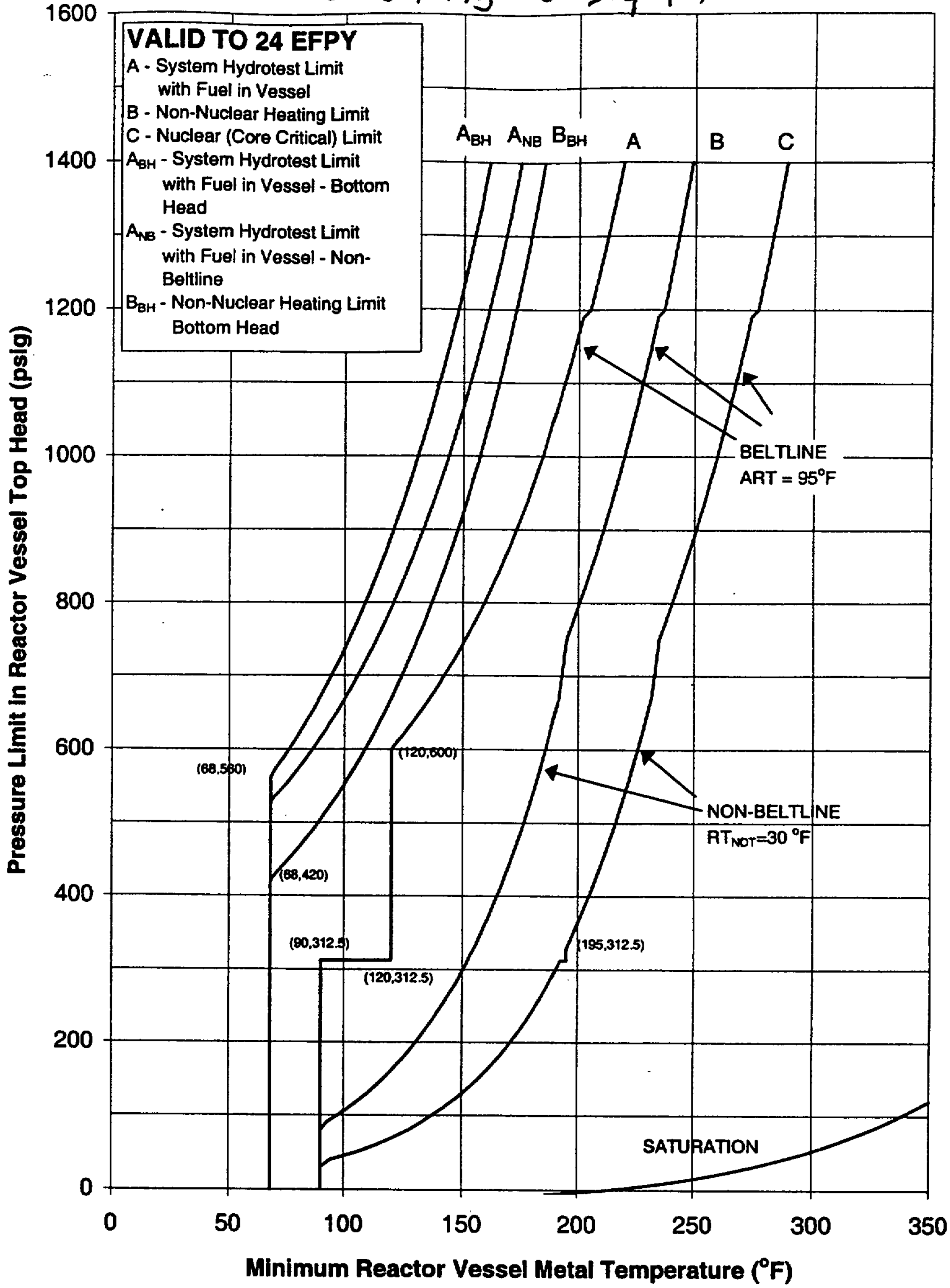

Figure 3.4.9-1 (page 1 of 1)
Reactor Coolant System Pressure and
Temperature Limits through 24 Effective Full Power Years (EFPY)

INSERT PAGE 3.4-26a

Insert Figure 3.4.9-2

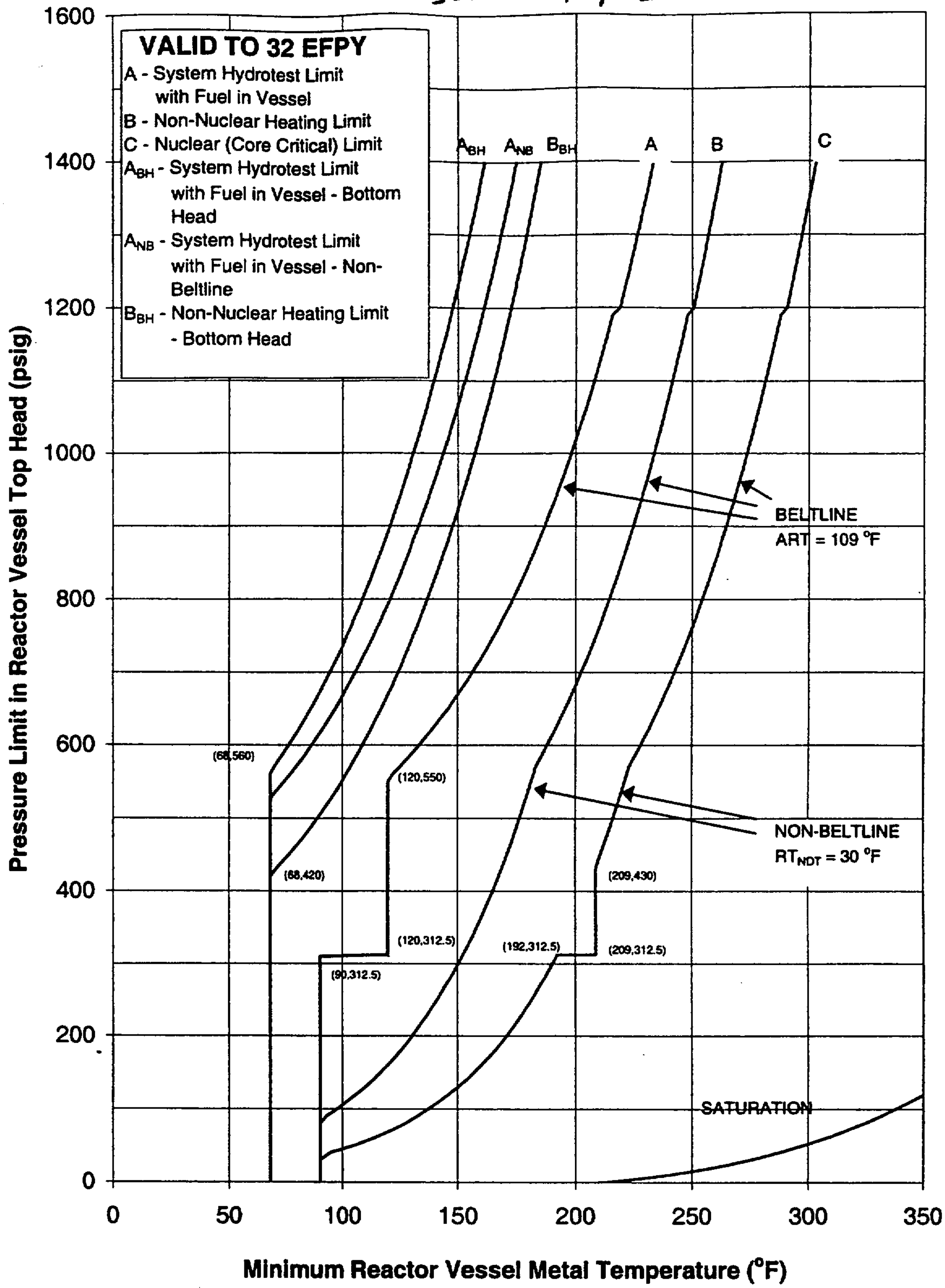

Figure 3.4.9-2 (page 1 of 1)
Reactor Coolant System Pressure and
Temperature Limits through 32 Effective Full Power Years (EFPY)

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Insert Page 3.4-266

JAFNPP

IMPROVED STANDARD TECHNICAL SPECIFICATIONS (ISTS) CONVERSION

ITS: 3.4.9

RCS Pressure and Temperature (P/T) Limits

JUSTIFICATION FOR DIFFERENCES (JFDs)
FROM NUREG-1433, REVISION 1

JUSTIFICATION FOR DIFFERENCES FROM NUREG-1433, REVISION 1
ITS: 3.4.9 - RCS PRESSURE AND TEMPERATURE (P/T) LIMITS

RETENTION OF EXISTING REQUIREMENT (CLB)

CLB1 JAFNPP has not developed the "Reactor Coolant System (RCS) PRESSURE AND TEMPERATURE LIMITS REPORT (PTLR)". References to limits in the PTLR are replaced with current requirements.

PLANT SPECIFIC WORDING PREFERENCE OR MINOR EDITORIAL IMPROVEMENT (PA)

PA1 NUREG-1433 Specification 3.4.5, "RCS Pressure Isolation Valve (PIV) Leakage", is not incorporated in ITS. Subsequent ITS Specifications and Bases have been renumbered accordingly.

PA2 Editorial changes have been made to achieve consistency with the Writer's Guide for the Restructured Technical Specifications.

PA3 Editorial changes have been made with no change in intent.

PLANT SPECIFIC DIFFERENCE IN DESIGN OR DESIGN BASIS (DB)

DB1 The bracketed allowance has been deleted since it does not apply to JAFNPP.

DIFFERENCE BASED ON APPROVED TRAVELER (TA)

TA1 The changes presented in Technical Specification Task Force (TSTF) Technical Specification Change Traveler Number 35, Revision 0, have been incorporated.

DIFFERENCE BASED ON PENDING TRAVELER (TP)

None

DIFFERENCE FOR OTHER REASONS THAN ABOVE (X)

X1 ITS SR 3.4.9.4 has been added to the requirements of ISTS 3.4.10 (ITS 3.4.9) to allow an alternative to the requirements of ITS SR 3.4.9.3. This Surveillance has been added to the CTS in accordance with L1. A Note 2 was added to SR 3.4.9.3 which allows the option to perform SR 3.4.9.4. In addition, subsequent Surveillances have been renumbered, as required.

JAFNPP

IMPROVED STANDARD TECHNICAL SPECIFICATIONS (ISTS) CONVERSION

ITS: 3.4.9

RCS Pressure and Temperature (P/T) Limits

MARKUP OF NUREG-1433, REVISION 1, BASES

B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.~~10~~ 9 PA1 RCS Pressure and Temperature (P/T) Limits

BASES

BACKGROUND

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

This Specification CLB1

~~The PTLR~~ contains P/T limit curves and criticality ~~for~~ heatup, cooldown, ~~and~~ inservice leakage and hydrostatic testing, and ~~data for~~ the maximum rate of change of reactor coolant temperature. ~~The heatup curve provides limits for both heatup and criticality.~~

PA2

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

The LCO establishes operating limits that provide a margin to brittle failure of the reactor vessel and piping of the reactor coolant pressure boundary (RCPB). The vessel is the component most subject to brittle failure. Therefore, the LCO limits apply mainly to the vessel. PA2

PA3 abnormal transients

10 CFR 50, Appendix G (Ref. 1), requires the establishment of P/T limits for material fracture toughness requirements of the RCPB materials. Reference 1 requires an adequate margin to brittle failure during normal operation, ~~anticipated~~ operational ~~occurrences~~, and system inservice leakage and hydrostatic tests. It mandates the use of the ASME Code, Section III, Appendix G (Ref. 2). PA2

INSERT BKGD PA2

The actual determined shift in the RT_{NDT} of the vessel material ~~will be established~~ is PA2 periodically by removing and evaluating the irradiated reactor vessel material specimens, in accordance with ASTM E 185 (Ref. 3) and Appendix H of 10 CFR 50 (Ref. 4). The operating P/T limit curves ~~will be~~ are PA2 adjusted,

(continued)

PA2

Insert BKGD

The nil-ductility transition (NDT) temperature, RT_{NDT}, is defined as the temperature below which ferritic steel breaks in a brittle rather than ductile manner. The RT_{NDT} increases as a function of neutron exposure at integrated neutron exposures greater than approximately 10^{17} nvt with neutron energy in excess of 1 MeV.

BASES

BACKGROUND (continued)

as necessary, based on the evaluation findings and the recommendations of Reference 5.

The P/T limit curves are composite curves established by superimposing limits derived from stress analyses of those portions of the reactor vessel and head that are the most restrictive. At any specific pressure, temperature, and temperature rate of change, one location within the reactor vessel will dictate the most restrictive limit. Across the span of the P/T limit curves, different locations are more restrictive, and, thus, the curves are composites of the most restrictive regions. locations PA 2

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

However, the P/T limit curves reflect the most restrictive of the heatup and cooldown curves. PA 2

The P/T PA3 criticality limits include the Reference 1 requirement that they be at least 40°F above the heatup curve or the cooldown curve and not lower than the minimum permissible temperature for the inservice leakage and hydrostatic testing.

The consequence of violating the LCO limits is that the RCS has been operated under conditions that can result in brittle failure of the RCPB, possibly leading to a nonisolable leak or loss of coolant accident. In the event these limits are exceeded, an evaluation must be performed to determine the effect on the structural integrity of the RCPB components. ASME Code, Section XI, Appendix E (Ref. 6), provides a recommended methodology for evaluating an operating event that causes an excursion outside the limits.

APPLICABLE SAFETY ANALYSES

The P/T limits are not derived from Design Basis Accident (DBA) analyses. They are prescribed during normal operation to avoid encountering pressure, temperature, and temperature rate of change conditions that might cause undetected flaws to propagate and cause nonductile failure of the RCPB, a condition that is unanalyzed. Reference 7 establishes the methodology for determining the P/T limits. Reference 8 approved the curves and limits required by this Specification. DB2 Since the P/T limits are not derived from any DBA, there are no acceptance

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

limits related to the P/T limits. Rather, the P/T limits are acceptance limits themselves since they preclude operation in an unanalyzed condition.

RCS P/T limits satisfy Criterion 2 of ~~the NRC Policy Statement~~ 10 CFR 50.36(c)(2)(ii) (Ref. 9). X1

LCO

The elements of this LCO are: Figure 3.4.9-1 or Figure 3.4.9-2, as applicable — AmD 258

CLB1 INSERT LCO-1

a. RCS pressure, and temperature, ~~and heatup or cooldown rate~~ are within the limits specified in ~~the PILR, during RCS heatup, cooldown, and inservice leak and hydrostatic testing;~~

b. The temperature difference between the reactor vessel bottom head coolant and the reactor pressure vessel (RPV) coolant is ~~within the limits of the PILR~~ ≤ 145°F CLB1 during recirculation pump startup~~, and during increases in THERMAL POWER or loop flow while operating at low THERMAL POWER or loop flow~~; DB1

c. The temperature difference between the reactor coolant in the respective recirculation loop and in the reactor vessel ~~meets the limit of the PILR~~ is ≤ 50°F CLB1 during recirculation pump startup~~, and during increases in THERMAL POWER or loop flow while operating at low THERMAL POWER or loop flow~~; DB1

d. RCS pressure and temperature are within the ~~criticality~~ limits specified in ~~the PILR~~ Figure 3.4.9-1 or Figure 3.4.9-2, as applicable, prior to achieving criticality; and AmD 258

e. The reactor vessel flange and the head flange temperatures are ~~within the limits of the PILR~~ ≥ 90°F CLB1 when tensioning the reactor vessel head bolting studs and when any stud is tensioned. PA2

These limits define allowable operating regions and permit a large number of operating cycles while also providing a wide margin to nonductile failure.

~~The rate of change of temperature limits control~~ The limits on the rate of change of RCS temperature, influenced by RCS flow and RCS stratification, control the thermal gradient through the vessel wall ~~and~~ are used as inputs for calculating the heatup, cooldown, and inservice leakage and hydrostatic testing P/T limit curves. Thus, the LCO for the

PA2 • For this reason, both RCS temperature and RPV metal temperatures

(continued)

CLB1

Insert LCO-1

In addition, RCS temperature change averaged over a one hour period is: ≤ 100°F when the RCS pressure and temperature are on or to the right of curve C of Figure 3.4.9-1 or Figure 3.4.9-2, as applicable, during inservice leak and hydrostatic testing; ≤ 20°F when the RCS pressure and temperature are to the left of curve C of Figure 3.4.9-1 or Figure 3.4.9-2, as applicable, during inservice leak and hydrostatic testing; and ≤ 100°F during other heatup and cooldown operations;

BASES

LCO (continued)

rate of change of temperature restricts stresses caused by thermal gradients and also ensures the validity of the P/T limit curves.

Insert LCO-2 CLB1

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

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

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

c. The existence(s), size(s), and orientation(s) of flaws in the vessel material. PA2

APPLICABILITY

The potential for violating a P/T limit exists at all times. For example, P/T limit violations could result from ambient temperature conditions that result in the reactor vessel metal temperature being less than the minimum allowed temperature for boltup. Therefore, this LCO is applicable even when fuel is not loaded in the core.

RAI 3.4.9-2

ACTIONS

A.1 and A.2

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

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

Besides restoring the P/T limit parameters to ~~operation~~ within limits, an evaluation is required to determine if RCS operation can continue. ~~The~~ This PA2

(continued)

CLB1 Insert LCO-2

P/T limit curves are provided for plant operations through 24 EFPY (Figure 3.4.9-1) and 32 EFPY (Figure 3.4.9-2). Curves A, A_{BH} (bottom head), and A_{NB} (non-beltline) establish the minimum temperature for hydrostatic and leak testing, Curves B and B_{BH} (bottom head) establish limits for plant heatup and cooldown when the reactor is not critical or during low power physics tests, and Curve C establishes the limits when the reactor is critical. In addition, ART is the adjusted reference temperature.

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BASES

ACTIONS

A.1 and A.2 (continued)

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

ASME Code, Section XI, Appendix E (Ref. 6), may be used to support the evaluation. However, its use is restricted to evaluation of the vessel beltline.

PA2

PA2

engineering

The 72 hour Completion Time is reasonable to accomplish the evaluation of a mild violation. More severe violations may require special, event specific stress analyses or inspections. A favorable evaluation must be completed if continued operation is desired.

PA2

A mild violation is one which is technically acceptable because it is bounded by an existing evaluation or one which reasonably can be expected to be found acceptable following evaluation,

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

B.1 and B.2

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

occurrence

PA2

likelihood

Pressure and temperature are reduced by placing the plant in at least MODE 3 within 12 hours and in MODE 4 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.

(continued)

BASES

ACTIONS (continued)

C.1 and C.2

Operation outside the P/T limits in other than MODES 1, 2, and 3 (including defueled conditions) must be corrected so that the RCPB is returned to a condition that has been verified by stress analyses. The Required Action must be initiated without delay and continued until the limits are restored.

Besides restoring the P/T limit parameters to within limits, an evaluation is required to determine if RCS operation is allowed. This evaluation must verify that the RCPB integrity is acceptable and must be completed before approaching criticality or heating up to > 200°F 212 PA4. Several methods may be used, including comparison with pre-analyzed transients, new analyses, or inspection of the components. ASME Code, Section XI, Appendix E (Ref. 6), may be used to support the evaluation; however, its use is restricted to evaluation of the beltline.

INSERT ACTION C PA4

RAZ 3.4.9-06

PA2 The limits of Figures 3.4.9-1 and 3.4.9-2 are met when operation is on or to the right of the applicable curve.

SURVEILLANCE REQUIREMENTS

SR 3.4.~~10~~ 9 .1 PA1

Verification that operation is within ~~PTLR~~ limits RCS pressure and temperature limits as well as within RCS temperature change CLB1 limits is required every 30 minutes when RCS pressure and temperature conditions are undergoing planned changes. This is accomplished by monitoring the bottom head drain, recirculation loop and RPV metal temperatures. PA2 ~~This Frequency is considered reasonable in view of the control room indication available to monitor RCS status. Also, since temperature rate of change limits are specified in hourly increments, 30 minutes permits a reasonable time for assessment and correction of minor deviations.~~

Surveillance for heatup, cooldown, or inservice leakage and hydrostatic testing may be discontinued when the criteria given in the relevant plant procedure for ending the activity are satisfied. CLB2 In general, if two consecutive temperature readings taken ≥ 30 minutes apart are within 5°F of each other, the activity can be considered complete.

This SR ~~has been~~ is modified ~~with~~ by a Note that requires this Surveillance to be performed only during system heatup and cooldown operations and inservice leakage and hydrostatic testing. PA2

INSERT SR 3.4.9.1 PA4

(continued)

PA4

Insert ACTION C

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

PA4

Insert SR 3.4.9.1

Unlike steady-state operation, these intentional operational transients may be characterized by large pressure and temperature changes, and performance of this SR provides assurance that RCS pressure and temperature remain within acceptable regions of the P/T limit curves as well as within RCS temperature change limits.

BASES

SURVEILLANCE REQUIREMENTS (continued)

9 PA1

SR 3.4.10.2

A separate limit is used when the reactor is approaching criticality. Consequently, the RCS pressure and temperature must be verified within the appropriate limits before withdrawing control rods that will make the reactor critical.

Performing the Surveillance within 15 minutes before control rod withdrawal for the purpose of achieving criticality provides adequate assurance that the limits will not be exceeded between the time of the Surveillance and the time of the control rod withdrawal.

SR 3.4.9.3 X2

9 PA1 9 3

X2 SR 3.4.10.3 and SR 3.4.10.4 3

Specified CLB1

Differential temperatures within the ~~applicable~~ PTLR limits ensure that thermal stresses resulting from the startup of an idle recirculation pump will not exceed design allowances. In addition, compliance with these limits ensures that the assumptions of the analysis for the startup of an idle recirculation loop (Ref. 8 10) are satisfied. DB2

Performing the Surveillance within 15 minutes before starting the idle recirculation pump provides adequate assurance that the limits will not be exceeded between the time of the Surveillance and the time of the idle pump start.

X2

INSERT SR 3.4.9.3-1 PA4

An acceptable means of demonstrating compliance with the temperature differential requirement in SR 3.4.10.4 9 3 is to compare the temperatures of the operating recirculation loop and the idle loop. PA1 X2

SR 3.4.9.4, and SR 3.4.9.5 have been PA4 X2

9

SR 3.4.10.3 ~~has been~~ modified by a Note that requires the Surveillance to be performed only in MODES 1, 2, 3, and 4 ~~with reactor steam dome pressure ≥ 25 psig~~. In MODE 5, the overall stress on limiting components is lower. Therefore, ΔT limits are not required.

TA1

during a recirculation pump startup since this is when the stresses occur

INSERT Note 2

X2

9 9 PA1 9

SR 3.4.10.5, SR 3.4.10.6, and SR 3.4.10.7 8 X2

6 7

Limits on the reactor vessel flange and head flange temperatures are generally bounded by the other P/T limits

(continued)

X2 PA4 Insert SR 3.4.9.3-1

Compliance with the temperature differential requirement in SR 3.4.9.3 is demonstrated by comparing the bottom head drain temperature to the reactor vessel steam dome saturation temperature. SR 3.4.9.4 requires the verification that the active recirculation pump flow exceeds 40% of rated pump flow or the active recirculation pump has been operating below 40% rated flow for a period no longer than 30 minutes. As specified in Reference 11 and 12, the alternative verification of SR 3.4.9.4 will ensure the temperature differential of SR 3.4.9.3 is met. RAI 3.4-Gen

Insert Note 2

X2

SR 3.4.9.3 is modified by a second Note, which clarifies that the SR does not have to be performed if SR 3.4.9.4 is satisfied. This is acceptable since References 10 and 11 demonstrate that SR 3.4.9.4 is an acceptable alternative. In addition, SR 3.4.9.4 is modified by a second Note, which clarifies that the SR does not have to be performed if SR 3.4.9.3 is satisfied. This is acceptable since SR 3.4.9.3 directly ensures there is no stratification.

RAI 3.4.9-05

PA1

BASES

SURVEILLANCE REQUIREMENTS

~~SR 3.4.10.6~~, ~~SR 3.4.10.7~~, and ~~SR 3.4.10.8~~ (continued) 9 6, 9 7, 9 8 X2

during system heatup and cooldown. However, operations When ~~approaching MODE 4 from MODE 5 and in MODE 4~~ any reactor vessel stud is tensioned with RCS temperature less than or equal to certain specified values require assurance that these temperatures meet the LCO limits.

The flange temperatures must be verified to be above the limits within 30 minutes before and while tensioning the vessel head bolting studs to ensure that once the head is tensioned the limits are satisfied. When ~~in MODE 4~~ with RCS temperature ≤ ~~80~~ 100°F, 30 minute checks of the flange temperatures are required because of the reduced margin to the limits. When ~~in MODE 4~~ with RCS temperature ≤ ~~100~~ 120°F, monitoring of the flange temperature is required every 12 hours to ensure the temperature is within ~~the~~ specified limits ~~specified in the PTLR~~.

PA2 CLB1 CLB1

The 30 minute Frequency reflects the urgency of maintaining the temperatures within limits, and also limits the time that the temperature limits could be exceeded. The 12 hour Frequency is reasonable based on the rate of temperature change possible at these temperatures.

INSERT SR 3.4.9.6 PA4

REFERENCES

1. 10 CFR 50, Appendix G.

2. ASME, Boiler and Pressure Vessel Code, Section III, Appendix G.

3. ASTM E 185-82, July 1982.

4. 10 CFR 50, Appendix H.

5. Regulatory Guide 1.99, Radiation Embrittlement of Reactor Vessel Materials, Revision 2, May 1988. PA2

6. ASME, Boiler and Pressure Vessel Code, Section XI, Appendix E.

7. NEDO-21778-A, Transient Pressure Rises Affecting Fracture Toughness Requirements for Boiling Water Reactors, December 1978. PA2 PA3

INSERT Ref-1 DB2 X1 X2

~~8~~ 10. ~~U~~FSAR, Section ~~15.1.26~~ 14.5 DB2 DB3

Insert Ref-2

(PA4)

Insert SR 3.4.9.6

SR 3.4.9.6 is modified by a Note which requires the SR to be performed only when tensioning the reactor vessel head bolting studs. SR 3.4.9.7 is modified by a Note which states that the SR is not required to be performed until 30 minutes after RCS temperature is ≤ 100°F in MODE 4. SR 3.4.9.8 is modified by a Note which states that the SR is not required to be performed until 12 hours after RCS temperature is ≤ 120°F in MODE 4. These Notes are necessary to specify when the reactor vessel flange and head flange temperatures are required to be within specified limits.

(DB2)

Insert Ref-1

Amd #258

8. Letter from Guy Vissing (NRC) to James Knubel (NYPA), Issuance of Amendment No. 258 to James A. FitzPatrick Nuclear Power Plant, November 29,1999.

9. 10 CFR 50.36(c)(2)(ii).

(X2)

Insert Ref-2

11. GE-NE-208-04-1292, Evaluation of Idle Recirculation Loop Restart Without Vessel Bottom Temperature Indication for FitzPatrick Nuclear Power Plant, December 1992.

12. JAF-RPT-RWR-02076, Verification of Alternative Operating Conditions for Idle Recirculation Loop Restart without Vessel Bottom Temperature Indication, June 25, 1995.

JAFNPP

IMPROVED STANDARD TECHNICAL SPECIFICATIONS (ISTS) CONVERSION

ITS: 3.4.9

RCS Pressure and Temperature (P/T) Limits

JUSTIFICATION FOR DIFFERENCES (JFDs)
FROM NUREG-1433, REVISION 1, BASES

JUSTIFICATION FOR DIFFERENCES FROM NUREG-1433, REVISION 1
ITS BASES: 3.4.9 - RCS PRESSURE AND TEMPERATURE (P/T) LIMITS

RETENTION OF EXISTING REQUIREMENT (CLB)

CLB1 JAFNPP has not developed the "Reactor Coolant System (RCS) PRESSURE AND TEMPERATURE LIMITS REPORT (PTLR)". References to limits in the PTLR are replaced with current requirements.

CLB2 The details of CTS Surveillance Requirement 4.6.A.3 and 4.6.A.4 allowance, for discontinuing the Surveillance when 2 consecutive readings are within 5°F of each other, is being retained (see LA1).

PLANT SPECIFIC WORDING PREFERENCE OR MINOR EDITORIAL IMPROVEMENT (PA)

PA1 NUREG-1433 Specification 3.4.5, "RCS Pressure Isolation Valve (PIV) Leakage," is not incorporated in ITS. Subsequent ITS Specifications and Bases have been renumbered accordingly.

PA2 Editorial changes have been made for enhanced clarification, correction, or improvement with no change in intent.

PA3 The Bases have been modified to reflect plant specific nomenclature.

PA4 Editorial changes have been made to maintain consistency with Specification and/or other Bases.

PLANT SPECIFIC DIFFERENCE IN DESIGN OR DESIGN BASIS (DB)

DB1 The JAFNPP RCS P/T limits do not include specific requirements for increases in power or flow while operating at low power or low flow. The JAFNPP P/T limits (associated with temperature differences between the reactor vessel bottom head coolant and the reactor pressure vessel coolant, and temperature differences between reactor coolant in the respective recirculation loop and in the reactor vessel) apply only during a recirculation pump startup. Therefore, the Bases are revised to reflect the limitations of the Specifications.

DB2 The Bases have been revised to reflect the plant specific References. Subsequent References have been renumbered, as applicable.

DB3 The brackets have been removed and the plant specific Reference included.

JUSTIFICATION FOR DIFFERENCES FROM NUREG-1433, REVISION 1
ITS BASES: 3.4.9 - RCS PRESSURE AND TEMPERATURE (P/T) LIMITS

DIFFERENCE BASED ON APPROVED TRAVELER (TA)

TA1 The changes presented in Technical Specification Task Force (TSTF) Technical Specification Change Traveler Number 35, Revision 0, have been incorporated.

DIFFERENCE BASED ON PENDING TRAVELER (TP)

None

DIFFERENCE FOR OTHER REASONS THAN ABOVE (X)

X1 NUREG-1433, Revision 1, Bases reference to "the NRC Policy Statement" has been replaced with 10 CFR 50.36(c)(2)(ii), in accordance with 60 FR 36953 effective August 18, 1995.

X2 ITS SR 3.4.9.4 has been added to the requirements of ISTS 3.4.10 (ITS 3.4.9) to allow an alternative to the requirements of ITS SR 3.4.9.3. This Surveillance has been added to the CTS in accordance with L1. A Note 2 was added to SR 3.4.9.3 which allows the option to perform SR 3.4.9.4. In addition, subsequent Surveillances have been renumbered, as required. Modifications have been made to the Bases to reflect the changes made to the Specification and to justify the allowance.

JAFNPP

IMPROVED STANDARD TECHNICAL SPECIFICATIONS (ISTS) CONVERSION

ITS: 3.4.9

RCS Pressure and Temperature (P/T) Limits

RETYPED PROPOSED IMPROVED TECHNICAL SPECIFICATIONS (ITS) AND BASES

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.9 RCS Pressure and Temperature (P/T) Limits

LCO 3.4.9 RCS pressure, RCS temperature, RCS heatup and cooldown rates, and the recirculation pump starting temperature requirements shall be maintained within limits.

APPLICABILITY: At all times.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. ---------NOTE--------- Required Action A.2 shall be completed if this Condition is entered. --------------------- Requirements of the LCO not met in MODE 1, 2, or 3.	A.1 Restore parameter(s) to within limits. AND A.2 Determine RCS is acceptable for continued operation.	30 minutes 72 hours
B. Required Action and associated Completion Time of Condition A not met.	B.1 Be in MODE 3. AND B.2 Be in MODE 4.	12 hours 36 hours

(continued)

RAI 3.4.9-2

ACTIONS (continued)

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

RAI 3.4.9-2

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.4.9.1	-----NOTE----- Only required to be performed during RCS heatup and cooldown operations and RCS inservice leak and hydrostatic testing. -----	
	Verify:	30 minutes
	a. RCS pressure and temperature are within the limits specified in Figure 3.4.9-1 or Figure 3.4.9-2, as applicable.	
	b. RCS temperature change averaged over a one hour period is:	
	1. ≤ 100°F when the RCS pressure and temperature are on or to the right of curve C of Figure 3.4.9-1 or Figure 3.4.9-2, as applicable, during inservice leak and hydrostatic testing;	
	2. ≤ 20°F when the RCS pressure and temperature are to the left of curve C of Figure 3.4.9-1 or Figure 3.4.9-2, as applicable, during inservice leak and hydrostatic testing; and	
	3. ≤ 100°F during other heatup and cooldown operations.	

(continued)

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SURVEILLANCE REQUIREMENTS (continued)

	SURVEILLANCE	FREQUENCY
SR 3.4.9.2	Verify RCS pressure and temperature are within the criticality limits specified in Figure 3.4.9-1 or Figure 3.4.9-2, as applicable.	Once within 15 minutes prior to control rod withdrawal for the purpose of achieving criticality
SR 3.4.9.3	-------------------NOTES------------------- 1. Only required to be met in MODES 1, 2, 3, and 4 during recirculation pump startup. 2. Not required to be performed if SR 3.4.9.4 is satisfied. --- Verify the difference between the bottom head coolant temperature and the reactor pressure vessel (RPV) coolant temperature is ≤ 145°F.	Once within 15 minutes prior to each startup of a recirculation pump
SR 3.4.9.4	-------------------NOTES------------------- 1. Only required to be met in MODES 1, 2, 3, and 4 during recirculation pump startup. 2. Not required to be met if SR 3.4.9.3 is satisfied. --- Verify the active recirculation pump flow exceeds 40% of rated pump flow or the active recirculation pump has been operating below 40% rated flow for a period no longer than 30 minutes.	Once within 15 minutes prior to each startup of a recirculation pump

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RAI 3.4 GEN

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.4.9.2	Verify RCS pressure and temperature are within the criticality limits specified in Figure 3.4.9-1 or Figure 3.4.9-2, as applicable.	Once within 15 minutes prior to control rod withdrawal for the purpose of achieving criticality
SR 3.4.9.3	-------------------NOTES------------------- 1. Only required to be met in MODES 1, 2, 3, and 4 during recirculation pump startup. 2. Not required to be performed if SR 3.4.9.4 is satisfied. -- Verify the difference between the bottom head coolant temperature and the reactor pressure vessel (RPV) coolant temperature is ≤ 145°F.	Once within 15 minutes prior to each startup of a recirculation pump
SR 3.4.9.4	-------------------NOTES------------------- 1. Only required to be met in MODES 1, 2, 3, and 4 during recirculation pump startup. 2. Not required to be met if SR 3.4.9.3 is satisfied. -- Verify the active recirculation drive flow exceeds 40% of rated drive flow or the active loop has been operating below 40% rated flow for a period no longer than 30 minutes.	Once within 15 minutes prior to each startup of a recirculation pump

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.4.9.5	------------------NOTE------------------ Only required to be met in MODES 1, 2, 3, and 4 during recirculation pump startup. -- Verify the difference between the reactor coolant temperature in the recirculation loop to be started and the RPV coolant temperature ≤ 50°F.	Once within 15 minutes prior to each startup of a recirculation pump
SR 3.4.9.6	------------------NOTE------------------ Only required to be performed when tensioning the reactor vessel head bolting studs. -- Verify reactor vessel flange and head flange temperatures are ≥ 90°F.	30 minutes
SR 3.4.9.7	------------------NOTE------------------ Not required to be performed until 30 minutes after RCS temperature ≤ 100°F with any reactor vessel stud tensioned. -- Verify reactor vessel flange and head flange temperatures are ≥ 90°F.	30 minutes
SR 3.4.9.8	------------------NOTE------------------ Not required to be performed until 12 hours after RCS temperature ≤ 120°F with any reactor vessel stud tensioned. -- Verify reactor vessel flange and head flange temperatures are ≥ 90°F.	12 hours

RAI 3.4.9-5

RAI 3.4.9-5

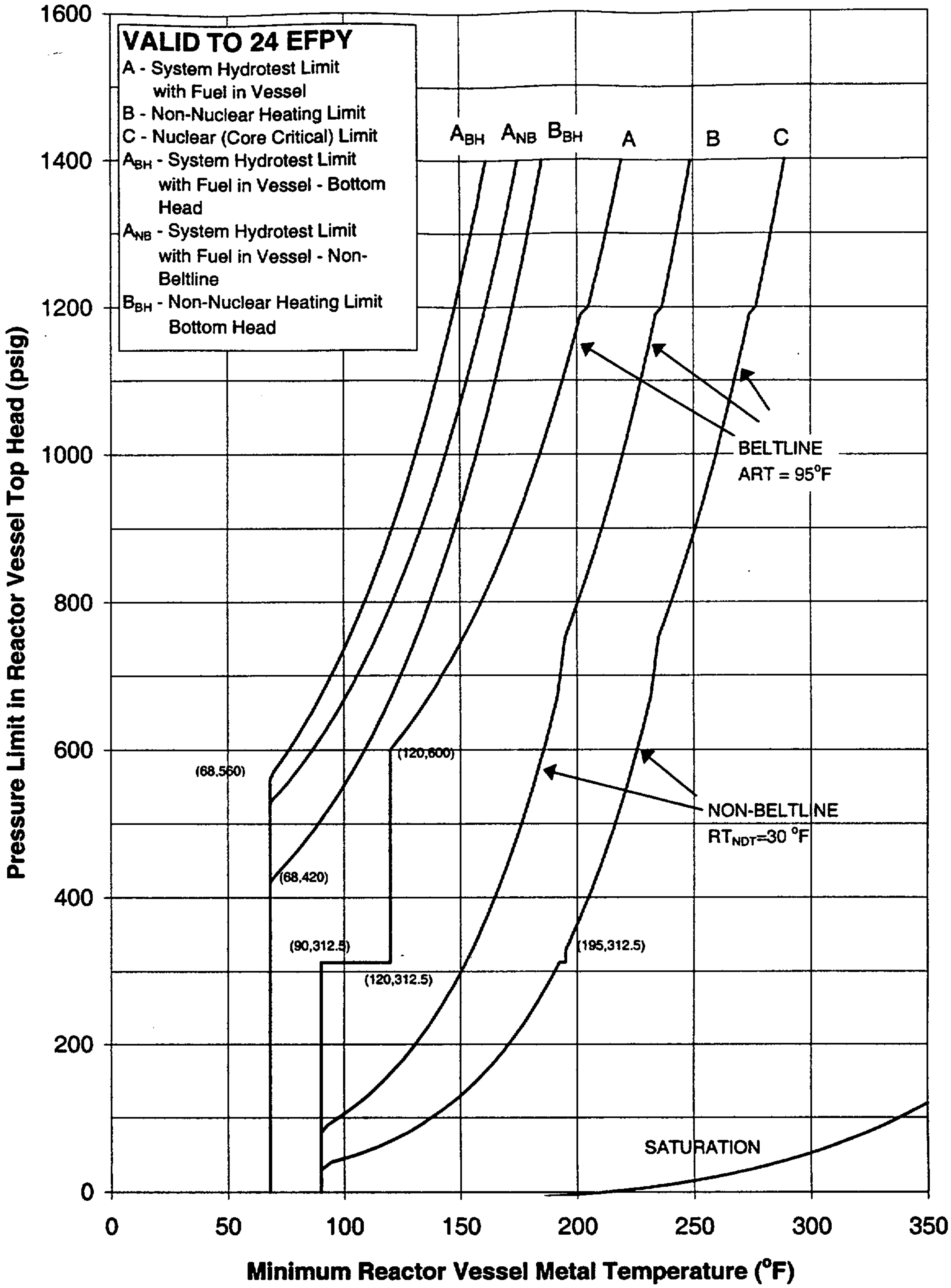

Figure 3.4.9-1 (page 1 of 1)
Reactor Coolant System Pressure and
Temperature Limits through 24 Effective Full Power Years (EFPY)

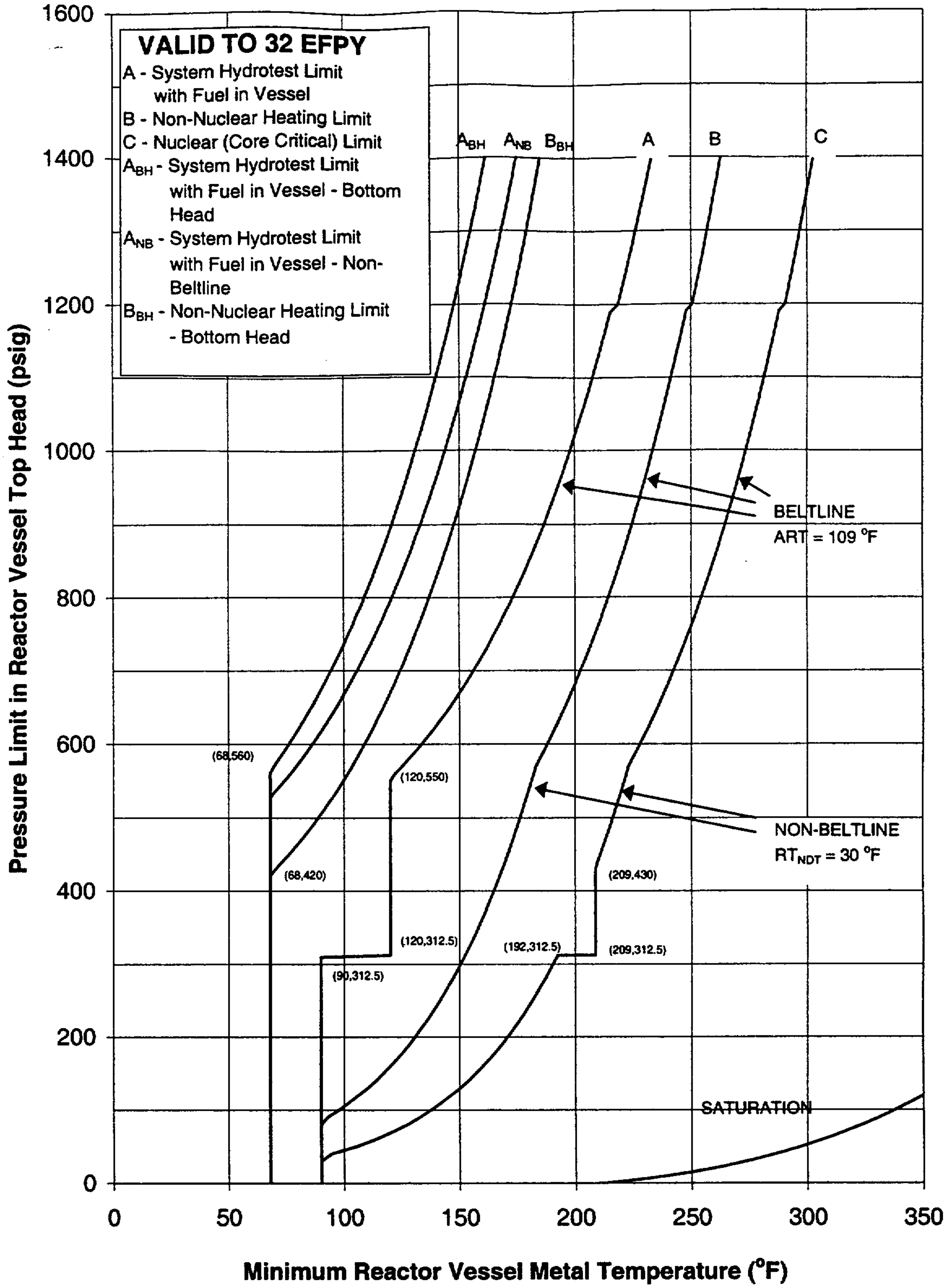

Figure 3.4.9-2 (page 1 of 1)
Reactor Coolant System Pressure and
Temperature Limits through 32 Effective Full Power Years (EFPY)

B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.9 RCS Pressure and Temperature (P/T) Limits

BASES

BACKGROUND

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

This Specification contains P/T limit curves for heatup, cooldown, inservice leakage and hydrostatic testing, and criticality and also limits the maximum rate of change of reactor coolant temperature.

Each P/T limit curve defines an acceptable region for normal operation. The curves are used for operational guidance during heatup or cooldown maneuvering. Pressure and temperature are monitored and compared to the applicable curve to ensure that operation is within the allowable region.

The LCO establishes operating limits that provide a margin to brittle failure of the reactor vessel and piping of the reactor coolant pressure boundary (RCPB). The vessel is the component most subject to brittle failure. Therefore, the LCO limits apply mainly to the vessel.

10 CFR 50, Appendix G (Ref. 1), requires the establishment of P/T limits for material fracture toughness requirements of the RCPB materials. Reference 1 requires an adequate margin to brittle failure during normal operation, abnormal operational transients, and system inservice leakage and hydrostatic tests. It mandates the use of the ASME Code, Section III, Appendix G (Ref. 2).

The nil-ductility transition (NDT) temperature, RT_{NDT}, is defined as the temperature below which ferritic steel breaks in a brittle rather than ductile manner. The RT_{NDT} increases as a function of neutron exposure at integrated neutron exposures greater than approximately 10^{17} nvt with neutron energy in excess of 1 MeV.

(continued)

BASES

BACKGROUND (continued)

The actual shift in the RT_{NDT} of the vessel material is determined periodically by removing and evaluating the irradiated reactor vessel material specimens, in accordance with ASTM E 185 (Ref. 3) and Appendix H of 10 CFR 50 (Ref. 4). The operating P/T limit curves are adjusted, as necessary, based on the evaluation findings and the recommendations of Reference 5.

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

The heatup curve represents a different set of restrictions than the cooldown curve because the directions of the thermal gradients through the vessel wall are reversed. The thermal gradient reversal alters the location of the tensile stress between the outer and inner walls. However, the P/T limit curves reflect the most restrictive of the heatup and cooldown curves.

The P/T criticality limits include the Reference 1 requirement that they be at least 40°F above the heatup curve or the cooldown curve and not lower than the minimum permissible temperature for the inservice leakage and hydrostatic testing.

The consequence of violating the LCO limits is that the RCS has been operated under conditions that can result in brittle failure of the RCPB, possibly leading to a nonisolable leak or loss of coolant accident. In the event these limits are exceeded, an evaluation must be performed to determine the effect on the structural integrity of the RCPB components. ASME Code, Section XI, Appendix E (Ref. 6), provides a recommended methodology for evaluating an operating event that causes an excursion outside the limits.

(continued)

BASES (continued)

APPLICABLE SAFETY ANALYSES

The P/T limits are not derived from Design Basis Accident (DBA) analyses. They are prescribed during normal operation to avoid encountering pressure, temperature, and temperature rate of change conditions that might cause undetected flaws to propagate and cause nonductile failure of the RCPB, a condition that is unanalyzed. Reference 7 establishes the methodology for determining the P/T limits. Reference 8 approved the curves and limits required by this Specification. Since the P/T limits are not derived from any DBA, there are no acceptance limits related to the P/T limits. Rather, the P/T limits are acceptance limits themselves since they preclude operation in an unanalyzed condition.

RCS P/T limits satisfy Criterion 2 of 10 CFR 50.36(c)(2)(ii) (Ref. 9).

LCO

The elements of this LCO are:

a. RCS pressure and temperature are within the limits specified in Figure 3.4.9-1 or Figure 3.4.9-2, as applicable. In addition, RCS temperature change averaged over a one hour period is: ≤ 100°F when the RCS pressure and temperature are on or to the right of curve C of Figure 3.4.9-1 or Figure 3.4.9-2, as applicable, during inservice leak and hydrostatic testing; ≤ 20°F when the RCS pressure and temperature are to the left of curve C of Figure 3.4.9-1 or Figure 3.4.9-2, as applicable, during inservice leak and hydrostatic testing; and ≤ 100°F during other heatup and cooldown operations; AMD258

b. The temperature difference between the reactor vessel bottom head coolant and the reactor pressure vessel (RPV) coolant is ≤ 145°F during recirculation pump startup;

c. The temperature difference between the reactor coolant in the respective recirculation loop and in the reactor vessel is ≤ 50°F during recirculation pump startup;

d. RCS pressure and temperature are within the limits specified in Figure 3.4.9-1 or Figure 3.4.9-2, as applicable, prior to achieving criticality; and AMD258

(continued)

BASES

LCO (continued)

e. The reactor vessel flange and the head flange temperatures are ≥ 90°F when tensioning the reactor vessel head bolting studs and when any stud is tensioned.

These limits define allowable operating regions and permit a large number of operating cycles while also providing a wide margin to nonductile failure.

The limits on the rate of change of RCS temperature, influenced by RCS flow and RCS stratification, control the thermal gradient through the vessel wall. For this reason, both RCS temperature and RPV metal temperatures are used as inputs for calculating the heatup, cooldown, and inservice leakage and hydrostatic testing P/T limit curves. Thus, the LCO for the rate of change of temperature restricts stresses caused by thermal gradients and also ensures the validity of the P/T limit curves.

P/T limit curves are provided for plant operations through 24 EFPY (Figure 3.4.9-1) and 32 EFPY (Figure 3.4.9-2). Curves A, A_{BH} (bottom head), and A_{NB} (non-beltline) establish the minimum temperature for hydrostatic and leak testing. Curves B and B_{BH} (bottom head) establish limits for plant heatup and cooldown when the reactor is not critical or during low power physics tests, and Curve C establishes the limits when the reactor is critical. In addition, ART is the adjusted reference temperature. Amd 258

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

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

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

c. The existence, size, and orientation of flaws in the vessel material.

(continued)

BASES

APPLICABILITY

The potential for violating a P/T limit exists at all times. For example, P/T limit violations could result from ambient temperature conditions that result in the reactor vessel metal temperature being less than the minimum allowed temperature for boltup. Therefore, this LCO is applicable even when fuel is not loaded in the core.

ACTIONS

A.1 and A.2

Operation outside the P/T limits while in MODE 1, 2, or 3 must be corrected so that the RCPB is returned to a condition that has been verified by stress analyses. RAI 3.4.9-2

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

Besides restoring the P/T limit parameters to within limits, an evaluation is required to determine if RCS operation can continue. This evaluation must verify the RCPB integrity remains acceptable and must be completed if continued operation is desired. Several methods may be used, including comparison with pre-analyzed transients in the stress analyses, new analyses, or inspection of the components. ASME Code, Section XI, Appendix E (Ref. 6), may be used to support the evaluation. However, its use is restricted to evaluation of the vessel beltline.

The 72 hour Completion Time is reasonable to accomplish the engineering evaluation of a mild violation. A mild violation is one which is technically acceptable because it is bounded by an existing evaluation or one which reasonably can be expected to be found acceptable following evaluation. More severe violations may require special, event specific stress analyses or inspections. A favorable evaluation must be completed if continued operation is desired.

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

(continued)

BASES

ACTIONS (continued)

B.1 and B.2

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

Pressure and temperature are reduced by placing the plant in at least MODE 3 within 12 hours and in MODE 4 within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

C.1 and C.2

Operation outside the P/T limits in other than MODES 1, 2, and 3 (including defueled conditions) must be corrected so that the RCPB is returned to a condition that has been verified by stress analyses. The Required Action must be initiated without delay and continued until the limits are restored.

Besides restoring the P/T limit parameters to within limits, an evaluation is required to determine if RCS operation is allowed. This evaluation must verify that the RCPB integrity is acceptable and must be completed before approaching criticality or heating up to > 212°F. Several methods may be used, including comparison with pre-analyzed transients, new analyses, or inspection of the components. ASME Code, Section XI, Appendix E (Ref. 6), may be used to support the evaluation; however, its use is restricted to evaluation of the beltline.

Condition C is modified by a Note requiring Required Action C.2 to be completed whenever the Condition is entered. The Note emphasizes the need to perform the evaluation of the effects of the excursion outside the allowable limits. Restoration alone per Required Action C.1 is insufficient

(continued)

BASES

ACTIONS

C.1 and C.2 (continued)

because higher than analyzed stresses may have occurred and may have affected the RCPB integrity.

SURVEILLANCE REQUIREMENTS

SR 3.4.9.1

Verification that operation is within RCS pressure and temperature limits as well as within RCS temperature change limits is required every 30 minutes when RCS pressure and temperature conditions are undergoing planned changes. This is accomplished by monitoring the bottom head drain, recirculation loop, and RPV metal temperatures. This Frequency is considered reasonable in view of the control room indication available to monitor RCS status. Also, since temperature rate of change limits are specified in hourly increments, 30 minutes permits a reasonable time for assessment and correction of minor deviations. The limits of Figures 3.4.9-1 and 3.4.9-2 are met when operation is on or to the right of the applicable curve.

RAI 3.4.9-06

Surveillance for heatup, cooldown, or inservice leakage and hydrostatic testing may be discontinued when the criteria given in the relevant plant procedure for ending the activity are satisfied. In general, if two consecutive temperature readings taken ≥ 30 minutes apart are within 5°F of each other the activity can be considered complete.

This SR is modified by a Note that requires this Surveillance to be performed only during system heatup and cooldown operations and inservice leakage and hydrostatic testing. Unlike steady-state operation, these intentional operational transients may be characterized by large pressure and temperature changes, and performance of this SR provides assurance that RCS pressure and temperature remain within acceptable regions of the P/T limit curves as well as within RCS temperature change limits.

SR 3.4.9.2

A separate limit is used when the reactor is approaching criticality. Consequently, the RCS pressure and temperature must be verified within the appropriate limits before

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.4.9.2 (continued)

withdrawing control rods that will make the reactor critical.

Performing the Surveillance within 15 minutes before control rod withdrawal for the purpose of achieving criticality provides adequate assurance that the limits will not be exceeded between the time of the Surveillance and the time of the control rod withdrawal.

SR 3.4.9.3, SR 3.4.9.4, and SR 3.4.9.5

Differential temperatures within the specified limits ensure that thermal stresses resulting from the startup of an idle recirculation pump will not exceed design allowances. In addition, compliance with these limits ensures that the assumptions of the analysis for the startup of an idle recirculation loop (Ref. 10) are satisfied.

Performing the Surveillance within 15 minutes before starting the idle recirculation pump provides adequate assurance that the limits will not be exceeded between the time of the Surveillance and the time of the idle pump start.

Compliance with the temperature differential requirement in SR 3.4.9.3 is demonstrated by comparing the bottom head drain temperature to the reactor vessel steam dome saturation temperature. SR 3.4.9.4 requires the verification that the active recirculation pump flow exceeds 40% of rated pump flow or the active recirculation pump has been operating below 40% rated flow for a period no longer than 30 minutes. As specified in Reference 11 and 12, the alternative verification of SR 3.4.9.4 will ensure the temperature differential of SR 3.4.9.3 is met.

RAI 3.4 GEN

An acceptable means of demonstrating compliance with the temperature differential requirement in SR 3.4.9.5 is to compare the temperatures of the operating recirculation loop and the idle loop.

SR 3.4.9.3, SR 3.4.9.4 and SR 3.4.9.5 have been modified by a Note that requires the Surveillance to be performed only in MODES 1, 2, 3, and 4 during a recirculation pump startup since this is when the stresses occur. In MODE 5, the

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.4.9.3, SR 3.4.9.4 and SR 3.4.9.5 (continued)

overall stress on limiting components is lower. Therefore, ΔT limits are not required. SR 3.4.9.3 is modified by a second Note, which clarifies that the SR does not have to be performed if SR 3.4.9.4 is satisfied. This is acceptable since References 10 and 11 demonstrate that SR 3.4.9.4 is an acceptable alternative. In addition, SR 3.4.9.4 is modified by a second Note, which clarifies that the SR does not have to be performed if SR 3.4.9.3 is satisfied. This is acceptable since SR 3.4.9.3 directly ensures there is no stratification.

SR 3.4.9.6, SR 3.4.9.7, and SR 3.4.9.8

Limits on the reactor vessel flange and head flange temperatures are generally bounded by the other P/T limits during system heatup and cooldown. However, operations when any reactor vessel stud is tensioned with RCS temperature less than or equal to certain specified values require assurance that these temperatures meet the LCO limits.

The flange temperatures must be verified to be above the limits within 30 minutes before and while tensioning the vessel head bolting studs to ensure that once the head is tensioned the limits are satisfied. When any reactor vessel stud is tensioned with RCS temperature ≤ 100°F, 30 minute checks of the flange temperatures are required because of the reduced margin to the limits. When any reactor vessel stud is tensioned with RCS temperature ≤ 120°F, monitoring of the flange temperature is required every 12 hours to ensure the temperature is within specified limits.

RAI 3.4.9-5

The 30 minute Frequency reflects the urgency of maintaining the temperatures within limits, and also limits the time that the temperature limits could be exceeded. The 12 hour Frequency is reasonable based on the rate of temperature change possible at these temperatures.

SR 3.4.9.6 is modified by a Note which requires the SR to be performed only when tensioning the reactor vessel head bolting studs. SR 3.4.9.7 is modified by a Note which states that the SR is not required to be performed until 30 minutes after RCS temperature is ≤ 100°F in MODE 4. SR 3.4.9.8 is modified by a Note which states that the SR is

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.4.9.6, SR 3.4.9.7, and SR 3.4.9.8 (continued)

not required to be performed until 12 hours after RCS temperature is ≤ 120°F in MODE 4. These Notes are necessary to specify when the reactor vessel flange and head flange temperatures are required to be within specified limits.

REFERENCES

1. 10 CFR 50, Appendix G.
2. ASME, Boiler and Pressure Vessel Code, Section III, Appendix G.
3. ASTM E 185-82, July 1982.
4. 10 CFR 50, Appendix H.
5. Regulatory Guide 1.99, Revision 2, Radiation Embrittlement Of Reactor Vessel Materials, May 1988.
6. ASME, Boiler and Pressure Vessel Code, Section XI, Appendix E.
7. NEDO-21778-A, Transient Pressure Rises Affecting Fracture Toughness Requirements For Boiling Water Reactors, December 1978.
8. Letter from Guy Vissing (NRC) to James Knubel (NYPA), Issuance of Amendment No. 258 to James A. FitzPatrick Nuclear Power Plant, November 29, 1999.
9. 10 CFR 50.36(c)(2)(ii).
10. UFSAR, Section 14.5.
11. GE-NE-208-04-1292, Evaluation of Idle Recirculation Loop Restart Without Vessel Bottom Temperature Indication for FitzPatrick Nuclear Power Plant, December 1992.
12. JAF-RPT-RWR-02076, Verification of Alternate Operating Conditions for Idle Recirculation Loop Restart Without Vessel Bottom Temperature Indication, June 25, 1995.

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IMPROVED STANDARD TECHNICAL SPECIFICATIONS (ISTS) CONVERSION

NUREG: N3.4.5

RCS Pressure Isolation Valve (PIV) Leakage

THIS SPECIFICATION IS DELETED.

THERE ARE NO REQUIREMENTS FOR THIS SPECIFICATION AT JAFNPP; THEREFORE THIS MARKUP PACKAGE CONTAINS ONLY THE FOLLOWING SECTIONS:

MARKUP OF NUREG-1433, REVISION 1, SPECIFICATION

JUSTIFICATION FOR DIFFERENCES (JFDs) FROM NUREG-1433, REVISION 1

MARKUP OF NUREG-1433, REVISION 1, BASES

JUSTIFICATION FOR DIFFERENCES (JFDs) FROM NUREG-1433, REVISION 1, BASES

JAFNPP

IMPROVED STANDARD TECHNICAL SPECIFICATIONS (ISTS) CONVERSION

NUREG: N3.4.5

RCS Pressure Isolation Valve (PIV) Leakage

MARKUP OF NUREG-1433, REVISION 1
SPECIFICATION

CLB1

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.5 RCS Pressure Isolation Valve (PIV) Leakage

LCO 3.4.5 The leakage from each RCS PIV shall be within limit.

APPLICABILITY: MODES 1 and 2.
MODE 3, except valves in the residual heat removal (RHR) shutdown cooling flow path when in, or during the transition to or from, the shutdown cooling mode of operation.

ACTIONS

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

1. Separate Condition entry is allowed for each flow path.

2. Enter applicable Conditions and Required Actions for systems made inoperable by PIVs.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more flow paths with leakage from one or more RCS PIVs not within limit.	------------NOTE------------ Each valve used to satisfy Required Action A.1 and Required Action A.2 must have been verified to meet SR 3.4.5.1 and be in the reactor coolant pressure boundary [or the high pressure portion of the system]. ----------------------------	

(continued)

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. (continued)	A.1	Isolate the high pressure portion of the affected system from the low pressure portion by use of one closed manual, de-activated automatic, or check valve.	4 hours
	<u>AND</u>		
	A.2	Isolate the high pressure portion of the affected system from the low pressure portion by use of a second closed manual, de-activated automatic, or check valve.	72 hours
B. Required Action and associated Completion Time not met.	B.1	Be in MODE 3.	12 hours
	<u>AND</u>		
	B.2	Be in MODE 4.	36 hours

CLB 1

CLB 1

SURVEILLANCE REQUIREMENTS

	SURVEILLANCE	FREQUENCY
SR 3.4.5.1	-----NOTE----- Not required to be performed in MODE 3. ----- Verify equivalent leakage of each RCS PIV is ≤ 0.5 gpm per nominal inch of valve size up to a maximum of 5 gpm, at an RCS pressure ≥ [ ] and ≤ [ ] psig.	[In accordance with the Inservice Testing Program or [18] months]

JAFNPP

IMPROVED STANDARD TECHNICAL SPECIFICATIONS (ISTS) CONVERSION

NUREG: N3.4.5

RCS Pressure Isolation Valve (PIV) Leakage

JUSTIFICATION FOR DIFFERENCES (JFDs) FROM NUREG-1433, REVISION 1

JUSTIFICATION FOR DIFFERENCES
NUREG: 3.4.5 - RCS PRESSURE ISOLATION VALVE (PIV) LEAKAGE

RETENTION OF EXISTING REQUIREMENT (CLB)

CLB1 NUREG-1433 Specification 3.4.5 sets forth Limiting Conditions for Operation and Surveillance Requirements for Reactor Coolant System (RCS) pressure isolation valve (PIV) leakage. PIVs are defined as any two valves in series within the reactor coolant pressure boundary (RCPB) which separate the high pressure RCS from an attached low pressure system. These valves are normally closed during power operation.

The Reactor Safety Study (WASH-1400) identified the potential intersystem loss of coolant accident (Event V) in a PWR as a significant contributor to the risk of core melt. In this scenario, check valves fail in the injection lines of the RHR or low pressure injection systems, allowing high pressure reactor coolant to enter low pressure piping outside containment. Subsequent failure of this low pressure piping would result in loss of reactor coolant outside containment and subsequent core meltdown. Similar scenarios were also determined to be possible in BWRs.

All plants licensed since 1979 have PIVs listed in their Technical Specifications, along with testing intervals, acceptance criteria, and limiting conditions for operation. Certain older plants were required to periodically leak test, on an individual basis, only those PIVs which were listed in an Order dated April 20, 1981 (Event V Order). That Order was sent to 32 operating PWRs and 2 operating BWRs. Other older plants have had no specific requirements imposed to individually leak test any of their PIVs.

A number of events have occurred involving leakage past PIVs, failures of the valves, inadvertent actuation of the valves, or mispositioning of the valves. As a result, the NRC issued Generic Letter 87-06, "Periodic Verification of Leak Tight Integrity of Pressure Isolation Valves," which requested that licensees submit (1) a list of all PIVs in their plant; and (2) a description of the periodic tests or other measures performed to assure the integrity of the valve as an independent barrier of the RCPB, along with the acceptance criteria for leakage, operational limits, and frequency of test performance. NYPA responded to Generic Letter 87-06 by letter JPN-87-034, dated June 11, 1987. All PIVs are tested in accordance with 10 CFR 50, Appendix J, Type B test requirements, except for certain testable check valves, which are cyclically pressure tested in accordance with the JAFNPP Inservice Testing Program.

RETENTION OF EXISTING REQUIREMENT (CLB)

CLB1 (continued)

JAFNPP was licensed prior to 1979, and was not a recipient of the Event V Order to perform periodic leak tests of PIVs. Therefore, the requirements of NUREG-1433 Specification 3.4.5 do not apply to JAFNPP, and are not incorporated in the ITS. Subsequent Specifications are renumbered accordingly.

PLANT SPECIFIC WORDING PREFERENCE OR MINOR EDITORIAL IMPROVEMENT (PA)

None

PLANT SPECIFIC DIFFERENCE IN DESIGN OR DESIGN BASIS (DB)

None

DIFFERENCE BASED ON APPROVED TRAVELER (TA)

None

DIFFERENCE BASED ON PENDING TRAVELER (TP)

None

DIFFERENCE FOR OTHER REASONS THAN ABOVE (X)

None

JAFNPP

IMPROVED STANDARD TECHNICAL SPECIFICATIONS (ISTS) CONVERSION

NUREG: N3.4.5

RCS Pressure Isolation Valve (PIV) Leakage

MARKUP OF NUREG-1433, REVISION 1, BASES

CLB 1

B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.5 RCS Pressure Isolation Valve (PIV) Leakage

BASES

BACKGROUND

The function of RCS PIVs is to separate the high pressure RCS from an attached low pressure system. This protects the RCS pressure boundary described in 10 CFR 50.2, 10 CFR 50.55a(c), and GDC 55 of 10 CFR 50, Appendix A (Refs. 1, 2, and 3). RCS PIVs are defined as any two normally closed valves in series within the reactor coolant pressure boundary (RCPB). PIVs are designed to meet the requirements of Reference 4. During their lives, these valves can produce varying amounts of reactor coolant leakage through either normal operational wear or mechanical deterioration.

The RCS PIV LCO allows RCS high pressure operation when leakage through these valves exists in amounts that do not compromise safety. The PIV leakage limit applies to each individual valve. Leakage through these valves is not included in any allowable LEAKAGE specified in LCO 3.4.4, "RCS Operational LEAKAGE."

Although this specification provides a limit on allowable PIV leakage rate, its main purpose is to prevent overpressure failure of the low pressure portions of connecting systems. The leakage limit is an indication that the PIVs between the RCS and the connecting systems are degraded or degrading. PIV leakage could lead to overpressure of the low pressure piping or components. Failure consequences could be a loss of coolant accident (LOCA) outside of containment, an unanalyzed event that could degrade the ability for low pressure injection.

A study (Ref. 5) evaluated various PIV configurations to determine the probability of intersystem LOCAs. This study concluded that periodic leakage testing of the PIVs can substantially reduce intersystem LOCA probability.

PIVs are provided to isolate the RCS from the following typically connected systems:

a. Residual Heat Removal (RHR) System;

b. Core Spray System;

(continued)

BASES

CLB1

BACKGROUND (continued)

c. High Pressure Coolant Injection System; and

d. Reactor Core Isolation Cooling System.

The PIVs are listed in Reference 6.

APPLICABLE SAFETY ANALYSES

Reference 5 evaluated various PIV configurations, leakage testing of the valves, and operational changes to determine the effect on the probability of intersystem LOCAs. This study concluded that periodic leakage testing of the PIVs can substantially reduce the probability of an intersystem LOCA.

PIV leakage is not considered in any Design Basis Accident analyses. This Specification provides for monitoring the condition of the RCPB to detect PIV degradation that has the potential to cause a LOCA outside of containment. RCS PIV leakage satisfies Criterion 2 of the NRC Policy Statement.

LCO

RCS PIV leakage is leakage into closed systems connected to the RCS. Isolation valve leakage is usually on the order of drops per minute. Leakage that increases significantly suggests that something is operationally wrong and corrective action must be taken. Violation of this LCO could result in continued degradation of a PIV, which could lead to overpressurization of a low pressure system and the loss of the integrity of a fission product barrier.

The LCO PIV leakage limit is 0.5 gpm per nominal inch of valve size with a maximum limit of 5 gpm (Ref. 4).

Reference 7 permits leakage testing at a lower pressure differential than between the specified maximum RCS pressure and the normal pressure of the connected system during RCS operation (the maximum pressure differential). The observed rate may be adjusted to the maximum pressure differential by assuming leakage is directly proportional to the pressure differential to the one-half power.

(continued)

BASES (continued)

APPLICABILITY

In MODES 1, 2, and 3, this LCO applies because the PIV leakage potential is greatest when the RCS is pressurized. In MODE 3, valves in the RHR shutdown cooling flow path are not required to meet the requirements of this LCO when in, or during transition to or from, the RHR shutdown cooling mode of operation.

CLB 1

In MODES 4 and 5, leakage limits are not provided because the lower reactor coolant pressure results in a reduced potential for leakage and for a LOCA outside the containment. Accordingly, the potential for the consequences of reactor coolant leakage is far lower during these MODES.

ACTIONS

The ACTIONS are modified by two Notes. Note 1 has been provided to modify the ACTIONS related to RCS PIV flow paths. Section 1.3, Completion Times, specifies once a Condition has been entered, subsequent divisions, subsystems, components, or variables expressed in the Condition discovered to be inoperable or not within limits will not result in separate entry into the Condition. Section 1.3 also specifies Required Actions of the Condition continue to apply for each additional failure, with Completion Times based on initial entry into the Condition. However, the Required Actions for the Condition of RCS PIV leakage limits exceeded provide appropriate compensatory measures for separate affected RCS PIV flow paths. As such, a Note has been provided that allows separate Condition entry for each affected RCS PIV flow path. Note 2 requires an evaluation of affected systems if a PIV is inoperable. The leakage may have affected system OPERABILITY, or isolation of a leaking flow path with an alternate valve may have degraded the ability of the interconnected system to perform its safety function. As a result, the applicable Conditions and Required Actions for systems made inoperable by PIVs must be entered. This ensures appropriate remedial actions are taken, if necessary, for the affected systems.

A.1 and A.2

If leakage from one or more RCS PIVs is not within limit, the flow path must be isolated by at least one closed manual, deactivated automatic, or check valve within 4 hours.

(continued)

BASES

ACTIONS

A.1 and A.2 (continued)

Required Action A.1 and Required Action A.2 are modified by a Note stating that the valves used for isolation must meet the same leakage requirements as the PIVs and must be on the RCPB [or the high pressure portion of the system].

CLB1

Four hours provides time to reduce leakage in excess of the allowable limit and to isolate the flow path if leakage cannot be reduced while corrective actions to reseat the leaking PIVs are taken. The 4 hours allows time for these actions and restricts the time of operation with leaking valves.

Required Action A.2 specifies that the double isolation barrier of two valves be restored by closing another valve qualified for isolation or restoring one leaking PIV. The 72 hour Completion Time considers the time required to complete the action, the low probability of a second valve failing during this time period, and the low probability of a pressure boundary rupture of the low pressure ECCS piping when overpressurized to reactor pressure (Ref. 7).

B.1 and B.2

If leakage cannot be reduced or the system isolated, the plant must be brought to a MODE in which the LCO does not apply. To achieve this status, the plant must be brought to MODE 3 within 12 hours and MODE 4 within 36 hours. This action may reduce the leakage and also reduces the potential for a LOCA outside the containment. The Completion Times are reasonable, based on operating experience, to achieve the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.4.5.1

Performance of leakage testing on each RCS PIV is required to verify that leakage is below the specified limit and to identify each leaking valve. The leakage limit of 0.5 gpm per inch of nominal valve diameter up to 5 gpm maximum applies to each valve. Leakage testing requires a stable pressure condition. For the two PIVs in series, the leakage

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.4.5.1 (continued)

requirement applies to each valve individually and not to the combined leakage across both valves. If the PIVs are not individually leakage tested, one valve may have failed completely and not be detected if the other valve in series meets the leakage requirement. In this situation, the protection provided by redundant valves would be lost.

CLB1

The 18 month Frequency required by the Inservice Testing Program is within the ASME Code, Section XI, Frequency requirement and is based on the need to perform this Surveillance during an outage and the potential for an unplanned transient if the Surveillance were performed with the reactor at power.

This SR is modified by a Note that states the leakage Surveillance is not required to be performed in MODE 3. Entry into MODE 3 is permitted for leakage testing at high differential pressures with stable conditions not possible in the lower MODES.

REFERENCES

1. 10 CFR 50.2.
2. 10 CFR 50.55a(c).
3. 10 CFR 50, Appendix A, GDC 55.
4. ASME, Boiler and Pressure Vessel Code, Section XI.
5. NUREG-0677, May 1980.
6. FSAR, Section [ ].
7. NEDC-31339, November 1986.

JAFNPP

IMPROVED STANDARD TECHNICAL SPECIFICATIONS (ISTS) CONVERSION

NUREG: N3.4.5

RCS Pressure Isolation Valve (PIV) Leakage

JUSTIFICATION FOR DIFFERENCES (JFDs) FROM NUREG-1433, REVISION 1, BASES

JUSTIFICATION FOR DIFFERENCES
NUREG BASES: 3.4.5 - RCS PRESSURE ISOLATION VALVE (PIV) LEAKAGE

RETENTION OF EXISTING REQUIREMENT (CLB)

CLB1 The Bases for NUREG-1433 Specification 3.4.5 are deleted. NUREG-1433 Specification 3.4.5 does not apply to JAFNPP, and is not incorporated in the ITS. Subsequent Specifications and Bases are renumbered accordingly.

PLANT SPECIFIC WORDING PREFERENCE OR MINOR EDITORIAL IMPROVEMENT (PA)

None

PLANT SPECIFIC DIFFERENCE IN DESIGN OR DESIGN BASIS (DB)

None

DIFFERENCE BASED ON APPROVED TRAVELER (TA)

None

DIFFERENCE BASED ON PENDING TRAVELER (TP)

None

DIFFERENCE FOR OTHER REASONS THAN ABOVE (X)

None

JAFNPP

IMPROVED STANDARD TECHNICAL SPECIFICATIONS (ISTS) CONVERSION

NUREG: N3.4.11

Reactor Steam Dome Pressure

THIS SPECIFICATION IS DELETED.

THERE ARE NO REQUIREMENTS FOR THIS SPECIFICATION AT JAFNPP; THEREFORE THIS MARKUP PACKAGE CONTAINS ONLY THE FOLLOWING SECTIONS:

MARKUP OF NUREG-1433, REVISION 1, SPECIFICATION

JUSTIFICATION FOR DIFFERENCES (JFDs) FROM NUREG-1433, REVISION 1

MARKUP OF NUREG-1433, REVISION 1, BASES

JUSTIFICATION FOR DIFFERENCES (JFDs) FROM NUREG-1433, REVISION 1, BASES

JAFNPP

IMPROVED STANDARD TECHNICAL SPECIFICATIONS (ISTS) CONVERSION

NUREG: N3.4.11

Reactor Steam Dome Pressure

MARKUP OF NUREG-1433, REVISION 1
SPECIFICATION

DB1

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.11 Reactor Steam Dome Pressure

LCO 3.4.11 The reactor steam dome pressure shall be ≤ [1020] psig.

APPLICABILITY: MODES 1 and 2.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Reactor steam dome pressure not within limit.	A.1 Restore reactor steam dome pressure to within limit.	15 minutes
B. Required Action and associated Completion Time not met.	B.1 Be in MODE 3.	12 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
SR 3.4.11.1 Verify reactor steam dome pressure is ≤ [1020] psig.	12 hours

JAFNPP

IMPROVED STANDARD TECHNICAL SPECIFICATIONS (ISTS) CONVERSION

NUREG: N3.4.11

Reactor Steam Dome Pressure

JUSTIFICATION FOR DIFFERENCES (JFDs)
FROM NUREG-1433, REVISION 1

JUSTIFICATION FOR DIFFERENCES FROM NUREG-1433, REVISION 1
NUREG: 3.4.11 - REACTOR STEAM DOME PRESSURE

RETENTION OF EXISTING REQUIREMENT (CLB)

None

PLANT SPECIFIC WORDING PREFERENCE OR MINOR EDITORIAL IMPROVEMENT (PA)

None

PLANT SPECIFIC DIFFERENCE IN DESIGN OR DESIGN BASIS (DB)

DB1 NUREG-1433 Specification 3.4.11, "Reactor Steam Dome Pressure," is not incorporated in the ITS. This NUREG Specification is required to help ensure the vessel overpressure protection analysis can be met by ensuring the initial conditions of the event are preserved. The JAFNPP site specific overpressure protection analysis is analyzed with an initial condition equivalent to the analytical limit (1094 psig) associated with the Reactor Protection System Instrumentation Reactor Pressure - High Function in ITS 3.3.1.1, Reactor Protection System Instrumentation. A CHANNEL CHECK of the associated instrumentation is required to be performed every 12 hours during operations in MODES 1 and 2 which is consistent with the Surveillance Frequency in ISTS SR 3.4.11.1. The CHANNEL CHECK Surveillance will ensure reactor pressure is below the Allowable Value (1080 psig) which is lower than the analytical limit associated with this Function as well as below the initial condition of reactor pressure assumed in the overpressure protection analysis. In fact, since reactor pressure is normally at or below 1040 psig during reactor operations, action will be taken at a much lower pressure to restore reactor pressure or to restore inoperable channels than what is required by ISTS 3.4.11. The requirements of ITS 3.3.1.1, Reactor Protection System Instrumentation, are adequate to ensure that reactor pressure remains below the analytical limit assumed in the vessel overpressure protection analysis and that the Reactor Pressure - High Function channels remain Operable. Therefore, ISTS 3.4.11 is not required to be included in the JAFNPP ITS.

DIFFERENCE BASED ON APPROVED TRAVELER (TA)

None

DIFFERENCE BASED ON PENDING TRAVELER (TP)

None

DIFFERENCE FOR OTHER REASONS THAN ABOVE (X)

None

JAFNPP

IMPROVED STANDARD TECHNICAL SPECIFICATIONS (ISTS) CONVERSION

NUREG: N3.4.11

Reactor Steam Dome Pressure

MARKUP OF NUREG-1433, REVISION 1, BASES

DB1

B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.11 Reactor Steam Dome Pressure

BASES

BACKGROUND	The reactor steam dome pressure is an assumed initial condition of design basis accidents and transients and is also an assumed value in the determination of compliance with reactor pressure vessel overpressure protection criteria.
APPLICABLE SAFETY ANALYSES	The reactor steam dome pressure of ≤ [1020] psig is an initial condition of the vessel overpressure protection analysis of Reference 1. This analysis assumes an initial maximum reactor steam dome pressure and evaluates the response of the pressure relief system, primarily the safety/relief valves, during the limiting pressurization transient. The determination of compliance with the overpressure criteria is dependent on the initial reactor steam dome pressure; therefore, the limit on this pressure ensures that the assumptions of the overpressure protection analysis are conserved. Reference 2 also assumes an initial reactor steam dome pressure for the analysis of design basis accidents and transients used to determine the limits for fuel cladding integrity (see Bases for LCO 3.2.2, "MINIMUM CRITICAL POWER RATIO (MCPR)") and 1% cladding plastic strain (see Bases for LCO 3.2.1, "AVERAGE PLANAR LINEAR HEAT GENERATION RATE (APLHGR)"). Reactor steam dome pressure satisfies the requirements of Criterion 2 of the NRC Policy Statement.
LCO	The specified reactor steam dome pressure limit of ≤ [1020] psig ensures the plant is operated within the assumptions of the transient analyses. Operation above the limit may result in a transient response more severe than analyzed.
APPLICABILITY	In MODES 1 and 2, the reactor steam dome pressure is required to be less than or equal to the limit. In these

(continued)

DB1

BASES

APPLICABILITY (continued)

MODES, the reactor may be generating significant steam and the design basis accidents and transients are bounding.

In MODES 3, 4, and 5, the limit is not applicable because the reactor is shut down. In these MODES, the reactor pressure is well below the required limit, and no anticipated events will challenge the overpressure limits.

ACTIONS

A.1

With the reactor steam dome pressure greater than the limit, prompt action should be taken to reduce pressure to below the limit and return the reactor to operation within the bounds of the analyses. The 15 minute Completion Time is reasonable considering the importance of maintaining the pressure within limits. This Completion Time also ensures that the probability of an accident occurring while pressure is greater than the limit is minimized. If the operator is unable to restore the reactor steam dome pressure to below the limit, then the reactor should be placed in MODE 3 to be operating within the assumptions of the transient analyses.

B.1

If the reactor steam dome pressure cannot be restored to within the limit 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 12 hours. The allowed Completion Time of 12 hours is reasonable, based on operating experience, to reach MODE 3 from full power conditions in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.4.11.1

Verification that reactor steam dome pressure is ≤ [1020] psig ensures that the initial conditions of the design basis accidents and transients are met. Operating experience has shown the 12 hour Frequency to be sufficient for identifying trends and verifying operation within safety analyses assumptions.

(continued)

DB1

BASES (continued)

REFERENCES

1. FSAR, Section [5.2.2.2.4].
2. FSAR, Section [15].

JAFNPP

IMPROVED STANDARD TECHNICAL SPECIFICATIONS (ISTS) CONVERSION

NUREG: N3.4.11

Reactor Steam Dome Pressure

JUSTIFICATION FOR DIFFERENCES (JFDs) FROM NUREG-1433, REVISION 1, BASES

JUSTIFICATION FOR DIFFERENCES FROM NUREG-1433, REVISION 1
NUREG BASES: 3.4.11 - REACTOR STEAM DOME PRESSURE

RETENTION OF EXISTING REQUIREMENT (CLB)

None

PLANT SPECIFIC WORDING PREFERENCE OR MINOR EDITORIAL IMPROVEMENT (PA)

None

PLANT SPECIFIC DIFFERENCE IN DESIGN OR DESIGN BASIS (DB)

DB1 NUREG-1433 Specification 3.4.11, "Reactor Steam Dome Pressure," is not incorporated in the ITS. This NUREG Specification is required to help ensure the vessel overpressure protection analysis can be met by ensuring the initial conditions of the event are preserved. The JAFNPP site specific overpressure protection analysis is analyzed with an initial condition equivalent to the analytical limit (1094 psig) associated with the Reactor Protection System Instrumentation Reactor Pressure - High Function in ITS 3.3.1.1, Reactor Protection System Instrumentation. A CHANNEL CHECK of the associated instrumentation is required to be performed every 12 hours during operations in MODES 1 and 2 which is consistent with the Surveillance Frequency in ISTS SR 3.4.11.1. The CHANNEL CHECK Surveillance will ensure reactor pressure is below the Allowable Value (1080 psig) which is lower than the analytical limit associated with this Function as well as below the initial condition of reactor pressure assumed in the overpressure protection analysis. In fact, since reactor pressure is normally at or below 1040 psig during reactor operations, action will be taken at a much lower pressure to restore reactor pressure or to restore inoperable channels than what is required by ISTS 3.4.11. The requirements of ITS 3.3.1.1, Reactor Protection System Instrumentation, are adequate to ensure that reactor pressure remains below the analytical limit assumed in the vessel overpressure protection analysis and that the Reactor Pressure - High Function channels remain Operable. Therefore, ISTS 3.4.11 is not required to be included in the JAFNPP ITS.

DIFFERENCE BASED ON APPROVED TRAVELER (TA)

None

DIFFERENCE BASED ON PENDING TRAVELER (TP)

None

DIFFERENCE FOR OTHER REASONS THAN ABOVE (X)
None

JAFNPP

IMPROVED STANDARD TECHNICAL SPECIFICATIONS (ISTS) CONVERSION

CTS: 3/4.6.F

Structural Integrity

THIS SPECIFICATION IS Relocated.

MARKUP OF CURRENT TECHNICAL SPECIFICATIONS (CTS)

DISCUSSION OF CHANGES (DOCs) TO THE CTS

NO SIGNIFICANT HAZARDS CONSIDERATION (NSHC) FOR LESS RESTICTIVE CHANGES

JAFNPP

IMPROVED STANDARD TECHNICAL SPECIFICATIONS (ISTS) CONVERSION

CTS: **3/4.6.F**

Structural Integrity

MARKUP OF CURRENT TECHNICAL SPECIFICATIONS (CTS)

Current Specification 3/4.6.F

R1

3.6 (cont'd)

F. Structural Integrity

The structural integrity of the Reactor Coolant System shall be maintained at the level required by the original acceptance standards throughout the life of the Plant.

See ITS: 3.4.2

G. Jet Pumps

Whenever the reactor is in the startup/hot standby or run modes, all jet pumps shall be operable. If it is determined that a jet pump is inoperable, the reactor shall be placed in a cold condition within 24 hours.

4.6 (cont'd)

F. Structural Integrity

1. Nondestructive inspections shall be performed on the ASME Boiler and Pressure Vessel Code Class 1, 2 and 3 components and supports in accordance with the requirements of the weld and support inservice inspection program. This inservice inspection program is based on an NRC approved edition of, and addenda to, Section XI of the ASME Boiler and Pressure Vessel Code which is in effect 12 months or less prior to the beginning of the inspection interval.

2. An augmented inservice inspection program is required for those high stressed circumferential piping joints in the main steam and feedwater lines larger than 4 inches in diameter, where no restraint against pipe whip is provided. The augmented in-service inspection program shall consist of 100 percent inspection of these welds per inspection interval.

3. An Inservice Inspection Program for piping identified in the NRC Generic Letter 88-01 shall be implemented in accordance with NRC staff positions on schedules, methods, personnel, and sample expansion included in this Generic Letter, or in accordance with alternate measures approved by the NRC staff.

G. Jet Pumps

Whenever there is recirculation flow with the reactor in the startup/hot standby or run modes, jet pump operability shall be checked daily by verifying that the following conditions do not occur simultaneously:

Amendment No. ~~96~~, ~~134~~, ~~170~~, ~~180~~, 203

Pa

JAFNPP

IMPROVED STANDARD TECHNICAL SPECIFICATIONS (ISTS) CONVERSION

CTS: 3/4.6.F

Structural Integrity

DISCUSSION OF CHANGES (DOCs) TO THE CTS

ADMINISTRATIVE CHANGES

None

TECHNICAL CHANGES - MORE RESTRICTIVE

None

TECHNICAL CHANGES - LESS RESTRICTIVE (GENERIC)

None

TECHNICAL CHANGES - LESS RESTRICTIVE (SPECIFIC)

None

TECHNICAL CHANGES - RELOCATIONS

R1 The structural integrity inspections are provided to prevent long term component degradation and provide long term maintenance of acceptable structural conditions of the system. The associated inspections are not required to ensure immediate OPERABILITY of the system. Therefore, the requirements specified in CTS 3.4.F did not satisfy the NRC Policy Statement Technical Specification screening criteria as documented in the Application of Selection Criteria to the JAFNPP Technical Specifications and have been relocated to the Technical Requirements Manual controlled in accordance with 10 CFR 50.59.

JAFNPP

IMPROVED STANDARD TECHNICAL SPECIFICATIONS (ISTS) CONVERSION

CTS: 3/4.6.F

Structural Integrity

NO SIGNIFICANT HAZARDS CONSIDERATION (NSHC) FOR LESS RESTICTIVE CHANGES

NO SIGNIFICANT HAZARDS CONSIDERATION
CTS: 3/4.6.F - STRUCTURAL INTEGRITY

TECHNICAL CHANGES - LESS RESTRICTIVE (SPECIFIC)

There are no plant specific less restrictive changes identified for this Specification.

MODIFIED RAI RESPONSES FOR ITS SECTION 3.4

Revision E Changes to Section 3.4 RAI Responses

RAIs
Generic Terminology for jet pump loop flow, jet pump flow, recirculation loop, recirculation pump loops, and recirculation drive flow have been changed, interchanged, etc. Only change the nomenclature that is plant specific. Other changes are generic and have to be changed through the established change process, e.g., the TSTF. This refers to all PA changes.

JAFNPP Response:

1. The Authority agrees that some of the changes made were not necessary to ensure understanding of the terms used while some of the changes are necessary to provide consistent terminology.

2. A summary of the changes needed in ITS 3.4.1 and 3.4.2 to achieve consistent use of terms is provided below:

    a. in ITS 3.4.1, "Insert ACTIONS A and B" - change "Jet pump loop flow..." to "Recirculation loop jet pump flow..." in Condition B to make it consistent with ITS SR 3.4.1.2.a discussed in b below.

    [Revised Response provided with Revision E Package - delete part 2.a since RAI 3.4.1-01 deletes Condition B]

    b. in ITS SR 3.4.1.2.a - retain "...recirculation loop jet pump flow..." as stated in the NUREG.

    c. in ITS SR 3.4.1.2 Bases (first paragraph, last sentence) - retain "...recirculation loop jet pump loop flow..." as stated in the NUREG.

    d. in ITS SR 3.4.2.1.a - retain "Recirculation pump flow..." as stated in the NUREG.

    e. in ITS SR 3.4.2.1.a - change "...jet pump loop flow..." to "...recirculation loop jet pump flow..." to make it consistent with NUREG SR 3.4.1.1 (ITS SR 3.4.1.2).

    f. in NUREG (and ITS) SR 3.4.2.1 Bases (first paragraph, 20th line) - change "...jet pump loop flow.." to "...recirculation loop jet pump flow..." to make consistent with NUREG SR 3.4.1.1 and NUREG SR 3.4.1.1 Bases.

    g. in NUREG (and ITS) SR 3.4.2.1 Bases (first paragraph, 20th line) - change "...recirculation loop flow..." to "...recirculation pump flow..." to make consistent with NUREG (and ITS) SR 3.4.2.1.a.

h. in NUREG (and ITS) SR 3.4.2.1 Bases (second paragraph, first sentence) - change "...(pump flow and loop flow versus..." to "...(recirculation pump flow and recirculation loop jet pump flow versus..." to make consistent with NUREG SR 3.4.1.1 (ITS SR 3.4.1.2) and NUREG (and ITS) SR 3.4.2.1.a, and

i. in NUREG (and ITS) SR 3.4.2.1 Bases (second paragraph, third sentence) - change "...pump flow and loop flow versus..." to "...recirculation pump flow and recirculation loop jet pump flow versus..." to make consistent withNUREG SR 3.4.1.1 (ITS SR 3.4.1.2) and NUREG (and ITS) SR 3.4.2.1.a.

3.4.9-04 CTS 3.6.A.5
DOC A4

CTS 3.6.A.5 indicates that with any of the limits 3.6.1 through 3.6.A.4 exceeded.... 3.6.A.4 specifies "during all Modes of operation. Would this not imply that 3.6.A.5 then should be the same. CTS 3.6.A does not specify Applicability. DOC A2 concluded that because there was not a stated Applicability in CTS 3.6.A, it implies that CTS 3.6.A is applicable at all times. DOC A2 logic conflicts with DOC A4. DOC A4 concludes that because CTS 3.6.C does not include an Applicability statement then the Applicability can be determined from the actions required when the LCO cannot be met. DOC A4 states "Since this Specification requires that, if the Required Actions and Completion Times are not met, the reactor be placed in Cold Shutdown (MODE 4), it can be implied that the Specification is Applicable in MODES 1, 2 and 3." A similar difference in logic exists between DOC L3 of ITS 3.4.6 and DOC A2 of ITS 3.4.9.

Comment: Provide discussion regarding the above apparent conflict in the discussions.

JAFNPP Response:

1. The FitzPatrick ITS conversion has noted in a number of DOCs that Applicability of a particular CTS LCO is implied based on CTS Required Action that stipulates an "end state" that is presumed to place the plant in a Mode or specified condition that is outside the (unstated) Applicability for the particular LCO. This "logic" for determining the Applicability of CTS 3.6.A.5 was (in error) used in ITS 3.4.9, DOC A4 and is (as stated above by the NRC reviewer) in conflict with ITS 3.4.9, DOC A2.

2. The Authority will revise ITS 3.4.9, DOC A2 and DOC A4 as well as ITS 3.4.6, DOC L3 as necessary to eliminate the conflicts.

   [Revised Response provided with Revision E Package]
   The Licensee will revise ITS, DOC A2 and DOC A4. However, there does not appear to be a conflict with ITS 3.4.6, DOC L3; thus it will not be revised.

3.4.9-06 CTS 3.6.A.2, .3, .4
Figure 3.6-1
Bases

CTS 3.6.A.2, 3.6.A.3 and 3.6.A.4 specify being to the right of CTS Figure 3.6-1 curves A, B, and C respectively, which makes clear the safe area for operation. By implication the same applies (being to the right) of the curves on ITS Figure 3.4.9-1. ITS 3.4.9 including ITS Figure 3.4.9-1, which is exactly the same as CTS figure 3.6-1 Part 3, and ITS 3.4.9 Bases do not specify anywhere that the safe area relative to curve A, B, or C is to the right. ITS 3.4.9 simply requires maintaining pressure and temperature within limits.

Comment: State where in the LCO the limits are found. Additionally, provide clarification in ITS 3.4.9 Bases where the safe area relative to ITS Figure 3.4.9-1 curves A, B, and C is located.

JAFNPP Response:

1. As noted in response RAI 3.4.9-01, NUREG-1433, Revision 1, does not state where pressure and temperature limits are found beyond making reference to an external report. In contrast, FitzPatrick limits are incorporated directly into ITS 3.4.9, with specific limits identified or referenced as applicable. Each specific limit is identified in its respective surveillance. (See response RAI 3.4.9-01.)

2. A note will be added to ITS Figure 3.4.9-1 specifying that safe operation is on or to the right of curve A, B, or C, as appropriate.

   [Revised Response provided with Revision E Package]
   The requirement that operation be on or to the right of the Figures will be relocated to the Bases, consistent with recently approved BWR ITS submittals.