United States Nuclear Regulatory Commission
Attachment IX to Serial: RNP-RA/97-0133
(495 Pages)

H. B. ROBINSON STEAM ELECTRIC PLANT, UNIT NO. 2
TECHNICAL SPECIFICATIONS CHANGE REQUEST TO CONVERT TO THE IMPROVED STANDARD TECHNICAL SPECIFICATIONS

SUPPLEMENT 5

SUPPLEMENT 5
CONVERSION PACKAGE SECTION 3.0
PAGE INSERTION INSTRUCTIONS

Remove and insert the following pages into Enclosure 8 to Serial: RNP-RA/96-0141.

	Remove Page	Insert Page
a.	Part 1, "Markup of Current Technical Specifications (CTS)"	
	NA	
b.	Part 2, "Discussion of Changes (DOCs) for CTS Markup"	
	1 through 6	1 through 7
c.	Part 3, " No Significant Hazards Consideration (NSHC), And Basis for Categorical Exclusion from 10 CFR 51.22	
	NA	
d.	Part 4, "Markup of NUREG-4131, Revision 1, Standard Technical Specifications-Westinghouse Plants, (ISTS)"	
	NA	
e.	Part 5, "Justification of Differences (JFDs) to ISTS"	
	NA	
f.	Part 6, "Markup of ISTS Bases"	
	NA	
g.	Part 7, "Justification for Differences (JFDs) to ISTS Bases"	
	NA	
h.	Part 8, "Proposed HBRSEP, Unit No. 2 ITS"	
	NA	
i.	Part 9. "Proposed Bases to HBRSEP, Unit No. 2 ITS Bases"	
	NA	
j.	Part 10. "ISTS Generic Changes"	
	NA	

ADMINISTRATIVE CHANGES

A1 In the conversion of the H. B. Robinson Steam Electric Plant (HBRSEP), Unit 2 Current Technical Specifications (CTS) to the proposed plant specific Improved Technical Specifications (ITS) certain wording preferences or conventions are adopted which do not result in technical changes (either actual or interpretational). Editorial changes, reformatting, and revised numbering are adopted to make ITS consistent with the conventions in the Standard Technical Specifications, Westinghouse Plants, NUREG 1431, Revision 1 (i.e., Improved Standard Technical Specifications (ISTS)).

A2 CTS Specification 3.0 is revised to adopt ISTS Specification LCO 3.0.1 in the ITS. Although this Specification is not explicitly stated in the CTS, current use and application of the CTS is consistent with the intent of ISTS Specification LCO 3.0.1. In addition, each CTS inherently implies that the associated specification requirements must be met when the specification is applicable. ITS LCO 3.0.1 provides clarity with regard to when LCOs must be met, and where any exceptions can be found. ITS LCO 3.0.1 is consistent with NUREG-1431, Revision 1 (including proposed generic change TSTF-6) and does not result in technical changes (either actual or interpretational). As such, this change is administrative, and has no adverse impact on safety.

A3 CTS Specification 3.0 is revised to adopt ISTS Specification LCO 3.0.2 in the ITS. Although this Specification is not explicitly stated in the CTS, current use and application of the CTS is consistent with the intent of ISTS Specification LCO 3.0.2. In addition, each CTS inherently implies that the associated specification actions must be performed when the requirements are not met. ITS LCO 3.0.2 provides clarity with regard to when LCOs must be met, and where any exceptions can be found. ITS LCO 3.0.2 is consistent with NUREG-1431, Revision 1 and the only technical change associated with CTS is the ITS LCO 3.0.5 exception (See Discussion of Change L1). As such, this change is administrative, and has no adverse impact on safety.

A4 CTS Specification 3.0 is revised to adopt ISTS Specification LCO 3.0.3 text in the ITS:

a. The CTS phrase, "Except as otherwise provided for in each specification," is replaced with the ISTS phrase, "Exceptions to this Specification are stated in the individual Specifications," to clarify where exceptions to this LCO can be found.

b. The CTS phrase, "if a Limiting Condition for Operation cannot be satisfied because of circumstances in excess of those addressed in the specification," is replaced with the ISTS phrase, When an LCO is not met and the associated ACTIONS are not met, an associated

ACTION is not provided, or if directed by the associated ACTIONS," to specifically state the circumstances which require compliance with this LCO.

c. The CTS phrase, "unless corrective measures are taken that permit operation under the permissible Limiting Condition for Operation statements for the specified time interval as measured from initial discovery or until the reactor is placed in a condition in which the specification is not applicable," is replaced with the ISTS phrase "Where corrective measures are completed that permit operation in accordance with the LCO or ACTIONS, completion of the actions required by LCO 3.0.3 is not required," to clarify ambiguities regarding the termination of actions related to this LCO.

This change is therefore administrative, and has no adverse impact on safety.

A5 Not used.

A6 ITS LCO 3.0.6 is added to provide guidance regarding the appropriate ACTIONS to be taken when a single inoperability (a support system) also results in the inoperability of one or more related systems (supported system(s)). In the CTS, based on the intent and interpretation provided by the NRC over the years, there has been an ambiguous approach to the combined support/supported inoperability. Some of this history is summarized:

- Guidance provided in the June 13, 1979 NRC memorandum from Brian K. Grimes (Assistant Director for Engineering and Projects) to Samuel E. Bryan (Assistant Director for Field Coordination) would indicate an intent/interpretation consistent with the ITS LCO 3.0.6 - without the necessity of also requiring additional ACTIONS. That is, only the inoperable support system ACTIONS need be taken.

- Guidance provided by the NRC in their April 10, 1980 letter to all Licensees, regarding the definition of OPERABILITY and its impact as a support system on the remainder of the current TS, would indicate a similar philosophy of not taking ACTIONS for the inoperable supported equipment. However, in this case, additional actions (similar to the ITS Safety Function Determination Program actions) were addressed and required.

- Generic Letter 91-18 and a plain-English reading of the existing TS provide an interpretation that inoperability, even as a result of a Technical Specification support system inoperability, requires all associated ACTIONS to be taken.

Considering the history of disagreement and misunderstandings in this area, the ISTS, NUREG-1431, was developed, with Industry input and approval of the NRC, to include ITS LCO 3.0.6, and a new program, Specification 5.5.15, Safety Function Determination Program. Since the function of ITS LCO 3.0.6 is to clarify existing ambiguities and to maintain actions within the realm of previous interpretations, this new provision is deemed to be administrative in nature.

A7 CTS Specification 3.0 is revised to adopt ISTS Specification LCO 3.0.7 in the ITS (ITS LCO 3.0.7). This Specification provides guidance with regard to meeting ITS Specification 3.1.8, "PHYSICS TESTS Exceptions-MODE 2," which allows certain Technical Specification requirements to be changed (made applicable in part or whole, or suspended) to permit performance of a PHYSICS TEST. If the Special Test Exception LCO did not exist, a PHYSICS TEST could not be performed in ITS. ITS LCO 3.0.7 eliminates confusion which would otherwise exist as to which LCOs apply during performance of a PHYSICS TEST. This change is consistent with the intent of the current physics testing exceptions; however, without this specific allowance to change the requirements of another LCO, a conflict of requirements could be incorrectly interpreted to exist. Technical changes associated with physics testing requirements are addressed in ITS 3.1.8 Discussion of Changes. Since this change only adds clarification to interpretation of CTS, this change is considered administrative, and has no adverse impact on safety.

A8 CTS Specification 4.0 is revised to adopt ISTS Specification SR 3.0.1 in the ITS. CTS Specification 4.0 states, "Performance of any surveillance test outlined in these specifications is not required when the system or component is out of service as permitted by the Limiting Conditions for Operation." ITS SR 3.0.1 states, "Surveillances do not have to be performed on inoperable equipment or variables outside specified limits." Although not explicitly stated in CTS 4.0, the complementary requirement that Surveillances must be performed on equipment required to be OPERABLE (in accordance with applicable Technical Specification requirements) is implied, which is consistent with ITS SR 3.0.1.

ISTS Specification SR 3.0.1 also clarifies that failure to meet a Surveillance means failure to meet the LCO, and that such failure can be experienced between performances, as well as during performances of the Surveillance. This is consistent with CTS Surveillance Requirements when applied in conjunction with the CTS definition of OPERABLE. CTS 4.0 states, "If it is discovered that a Surveillance Requirement,... was not performed within its specified frequency, then compliance with the requirement to declare the Technical Specification requirements are not met...." which implies that if a Surveillance Requirement is not performed when required, the LCO is not met. In addition, each CTS inherently implies that when the associated Surveillance Requirements are not met, the associated equipment is inoperable and the appropriate

actions are required to be performed. The requirement to declare the LCO not met, as required by ITS SR 3.0.1, is consistent with current interpretation on the use and application of CTS.

Upon discovery a Surveillance Requirement was not performed within its specified frequency, CTS 4.0 permits an extension up 24 hours or the limits of the applicable Frequency, whichever is less. CTS 4.0 requires, if the SR is not met or not performed during the delay period, the Technical Specification requirements must be immediately declared not met and applicable actions taken. Although not explicitly stated, it is implied, that if the delay period is not invoked the Technical Specification requirements must be immediately declared not met and applicable actions taken. This is consistent with the requirement of ITS SR 3.0.1 and SR 3.0.3.

A9 CTS Specification 4.0 is revised to adopt ISTS Specification SR 3.0.2 in the ITS. The CTS sentence, "Specified intervals may be adjusted plus or minus 25% to accommodate normal test schedules," is modified in ITS SR 3.0.2 to state, "The specified Frequency for each SR is met if the Surveillance is performed within 1.25 times the interval specified in the Frequency, as measured from the previous performance or as measured from the time a specified condition of the Frequency is met," for clarification and to establish what constitutes meeting the specified Frequency of each SR. Although not explicitly stated in CTS, this additional clarification is inherently implied in CTS 4.0 and is consistent with current interpretation of CTS 4.0. Currently, if a surveillance test is performed early, the next surveillance will be scheduled from that performance to ensure the surveillance test is not performed > 25% of the surveillance interval. In addition, the wording of ITS SR 3.0.2 does not preclude performing a surveillance test more frequently. Therefore, the CTS provision to adjust the specified interval by minus 25% is not necessary and is deleted. Also, the ISTS sentence, "Exceptions to this Specification are stated in the individual Specifications," is added to acknowledge the explicit use of exceptions in various Surveillances. This change is therefore administrative, and has no adverse impact on safety.

A10 CTS 4.1.1 explicitly requires calibration, testing and checking of instrument channels be performed as specified in Table 4.1-1. CTS 4.1.2 requires sampling tests be conducted as specified in Table 4.1-2. CTS 4.1.3 requires equipment tests be conducted as specified in Table 4.1-3. These specifications require the performance of the surveillances as specified in the individual tables but are not unique to a particular surveillance requirement. These specifications overlap other similar requirements specified in CTS 4.0 and are not separately retained in the ITS. Performance of SRs are required by ITS SR 3.0.1 consistent with the Applicabilities for the individual specifications. Therefore, elimination of these CTS specifications is considered administrative and

is consistent with ISTS.

A11 The CTS Bases are not retained in the ITS, but are replaced in their entirety. The ITS includes significantly expanded and improved Bases. The Bases do not define or impose any specific requirements but serve to explain, clarify and document the reasons (i.e., Bases) for the associated specification. The Bases are not part of the Technical Specifications required by 10 CFR 50.36. Therefore, this is an administrative change.

A12 CTS 3.3.7 provides Administrative Requirements to notify the NRC when maintenance to restore components or systems will exceed the periods specified. The requirements of this specification were rendered moot when CTS 3.0 was adopted in amendment 67. When an LCO cannot be met because of circumstances in excess of those addressed in the specification, CTS 3.0 requires the unit be placed in Hot Shutdown within 8 hours and Cold Shutdown within an additional 38 hours. Since the requirements of CTS 3.3.7 are obviated by the more restrictive requirements of CTS 3.0, the deletion of CTS 3.3.7 is considered an administrative change and is consistent with ISTS.

TECHNICAL CHANGES - MORE RESTRICTIVE

M1 CTS Specification 3.0 is revised to adopt ISTS Specification LCO 3.0.3 text in the ITS. The CTS requires that, if an LCO cannot be met and there is no specific action required to be taken, the unit be placed in Hot Shutdown within 8 hours and in Cold Shutdown within the next 30 hours. The ITS requires that, if an LCO cannot be met and there is no specific action required to be taken, the unit be placed in Hot Standby (MODE 3) within 7 hours, Hot Shutdown (MODE 4) within 13 hours, and Cold Shutdown (MODE 5) within 37 hours. The ITS MODE 3 specification of 7 hours imposes a more restrictive requirement by one hour. An additional restraint imposed that is not specified in the CTS, is that the unit be in MODE 4 within 13 hours. The time allowed to achieve cold shutdown in the CTS is 38 hours, and the time allowed in the ITS to achieve cold shutdown is 37 hours, resulting in the ITS being more restrictive by one hour. These changes are necessary to establish consistency with other similar shutdown requirements stated in individual specifications and are based on operating experience which indicates the times to place the unit in the specified MODES are reasonable. The time limits specified to reach lower MODES of operation permit the shutdown to proceed in a controlled and orderly manner that is well within the specified maximum cooldown rate and within the capabilities of the unit, assuming that only the minimum required equipment is OPERABLE. This reduces thermal transients on components of the Reactor Coolant System and the potential for a plant upset that could challenge safety systems under conditions to which this Specification applies. This change imposes more

restrictive requirements and is consistent with the ISTS.

M2 CTS Specifications 3.0 and 4.0 are revised to adopt ISTS Specifications LCO 3.0.4 and SR 3.0.4 in the ITS. These Specifications provide guidance related to MODE and operating condition entry when an LCO is not met. They also clarify those MODE changes permitted when required to comply with ACTIONS. The CTS does not preclude entry into a MODE in which compliance with a Specification applicable to that MODE is not met at the time of entry. This change is necessary to establish consistency with the overall approach and philosophy used in developing the ISTS. This change imposes more restrictive requirements and is consistent with the ISTS.

M3 The statement, "For Frequencies specified as "once," the above interval extension does not apply," is added to clarify that the 1.25 times the interval specified in the Frequency does not apply to certain Surveillances. This is because the interval extension concept is based on scheduling flexibility for repetitive performances, and these Surveillances are not repetitive in nature, and essentially have no "interval...as measured from the previous performance." This precludes the ability to extend these performances, and is therefore an additional restriction. The current Specification can be seen to allow the extension to apply to all Surveillances. This change is necessary since the initial performance of the Required Action, whether it is a particular Surveillance or some other remedial action, is considered a single action with a single Completion Time. One reason for not allowing the 25% extension to this Completion Time is that such an action usually verifies that no loss of function has occurred by checking the status of redundant or diverse components or accomplishes the function of the inoperable equipment in an alternative manner. This change imposes more restrictive requirements and is consistent with the ISTS.

TECHNICAL CHANGES - LESS RESTRICTIVE (GENERIC)

LA1 CTS Specification 4.0 is revised to adopt ISTS Specification SR 3.0.2 in the ITS. The CTS sentence, "Prior to returning the system to service, the specified calibration and testing surveillance shall be performed," is replaced with the ISTS sentence, "Surveillances have to be met and performed in accordance with SR 3.0.2, prior to returning equipment to OPERABLE status," and relocated to the Bases for SR 3.0.1. This detail is not required to be in the ITS to provide adequate protection of the health and safety of the public, since it provides details of a clarification nature, which are not pertinent to the actual surveillance requirement, but rather describe acceptable methods of compliance, and more appropriately belong in the Bases. Since these details are not necessary to adequately describe actual surveillance requirements, they

can be relocated to the Bases with no adverse impact on safety. This approach provides an effective level of regulatory control and provides for a more appropriate change control process. Changes to the Bases are controlled in accordance with the provisions of 10 CFR 50.59. The level of safety of facility operation is unaffected by the change because there is no change in the overall operational requirements. Furthermore, NRC and licensee resources associated with processing license amendments to these requirements will be reduced. This change is consistent with NUREG-1431. Therefore, relocation of this detail is acceptable.

TECHNICAL CHANGES - LESS RESTRICTIVE (SPECIFIC)

L1 LCO 3.0.5 is added to provide an exception to LCO 3.0.2 for instances where restoration of inoperable equipment to an OPERABLE status could not be performed while continuing to comply with Required Actions. Many Technical Specification ACTIONS require an inoperable component to be removed from service, such as: maintaining an isolation valve closed or tripping an inoperable instrument channel. To allow the performance of Surveillance Requirements to demonstrate the OPERABILITY of the equipment being returned to service, or to demonstrate the OPERABILITY of other equipment or variables within limits, which otherwise could not be performed without returning the equipment to service, an exception to these Required Actions is necessary. LCO 3.0.5 is necessary to establish an allowance that, although informally utilized in restoration of inoperable equipment, is not formally recognized in the current TS. Without this allowance certain components could not be restored to OPERABLE status and a plant shutdown would ensue. Clearly, it is not the intent or desire that the Technical Specifications preclude the return to service of a suspected OPERABLE component to confirm its OPERABILITY. This allowance is deemed to represent a more stable, safe operation than requiring a plant shutdown to complete the restoration and confirmatory testing.

L2 The statement "If a Completion Time requires periodic performance on a "once per..." basis, the above Frequency extension applies to each performance after the initial performance," is added to allow the 1.25 times the interval specified in the Frequency concept to apply to periodic Required Actions. This provides the consistency in scheduling flexibility for all performances of periodic requirements, whether they are Surveillances or Required Actions. The intent remains to perform the activity, on the average, once during each specified interval.

TECHNICAL CHANGES - LESS RESTRICTIVE (RELOCATION)

None

SUPPLEMENT 5
CONVERSION PACKAGE SECTION 3.1
PAGE INSERTION INSTRUCTIONS

Remove and insert the following pages into Enclosure 9 to Serial: RNP-RA/96-0141.

	Remove Page	Insert Page
a.	Part 1, "Markup of Current Technical Specifications (CTS)"	
	4.9-1, 3.10-2, 3.10-8, 3.1-11, 3.10-1	4.9-1, 3.10-2, 3.10-8, 3.1-11, 3.10-1
b.	Part 2, "Discussion of Changes (DOCs) for CTS Markup"	
	1 & 2	1& 2
	8 through 12	8 through 12
	-	13 though 17
c.	Part 3, " No Significant Hazards Consideration (NSHC), And Basis for Categorical Exclusion from 10 CFR 51.22	
	11 through 14	11 through 14
	-	15
d.	Part 4, "Markup of NUREG-4131, Revision 1, Standard Technical Specifications-Westinghouse Plants, (ISTS)"	
	3.1-23, 3.1-24,	3.1-23, 3.1-24
e.	Part 5, "Justification of Differences (JFDs) to ISTS"	
	1 though 3	1 through 3
f.	Part 6, "Markup of ISTS Bases"	
	B 3.1-43, B 3.1-44, B 3.1-65, B 3.1-66	B 3.1-43, B 3.1-44, B 3.1-65, B 3.1-66
	B 3.1-67	B 3.1-67
	Insert B 3.1.8-1 no page number	B 3.1-67a
g.	Part 7, "Justification for Differences (JFDs) to ISTS Bases"	
	1 & 4	1 & 4
h.	Part 8, "Proposed HBRSEP, Unit No. 2 ITS"	
	3.1-18 & 3.1-19	3.1-18 & 3.1-19
i.	Part 9. "Proposed Bases to HBRSEP, Unit No. 2 ITS Bases"	
	B 3.1-39, B 3.1-40, B 3.1-52, B 3.1-53	B 3.1-39, B 3.1-40, B 3.1-52, B 3.1-53
	B 3.1-54, B 3.1-55	B .1-54, B 3.1-55

SUPPLEMENT 5
CONVERSION PACKAGE SECTION 3.1
PAGE INSERTION INSTRUCTIONS

Remove and insert the following pages into Enclosure 9 to Serial: RNP-RA/96-0141.

	Remove Page	Insert Page
j.	Part 10. "ISTS Generic Changes" TSTF-14, Rev 2 (8 pages)	TSTF-14, Rev. 4 (8 pages)

Specification 3.1.2

ITS

A1

4.9 REACTIVITY ANOMALIES

Applicability

Applies to potential reactivity anomalies.

Objective

To require evaluation of reactivity anomalies within the reactor.

Specification

M2

[SR 3.1.2.1] Following a normalization of the computed boron concentration as a function of burnup, the actual boron concentration of the coolant shall be periodically compared with the predicted value. If the difference between the observed and predicted steady-state concentrations reaches the equivalent of one percent in reactivity, submit a Special Report to the Commission within 30 days.

L7

Basis

To eliminate possible errors in the calculations of the initial reactivity of the core and the reactivity depletion rate, the predicted relation between fuel burn-up and the boron concentration, necessary to maintain adequate control characteristics, must be adjusted (normalized) to accurately reflect actual core conditions. When full power is reached initially, and with the control rod groups in the desired positions, the boron concentration is measured and the predicted curve is adjusted to this point. As power operation proceeds, the measured boron concentration is compared with the predicted concentration and the slope of the curve relating burnup and reactivity is compared with that predicted. This process of normalization should

A7

Add LCO 3.1.2 + Applicability
RAs A.1, A.2 + B.1

M3

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ITS

3.10.1.5 [LCO 3.1.4] Except for physics tests, if a full length control rod is withdrawn as follows: (See - 3.1.8) (All shutdown and) (A1 shall be OPERABLE will all individual indicated rod positions)

- at positions ≥ 200 steps and is > 15 inches out of alignment with its bank position, or
- at positions < 200 steps and is > 7.5 inches out of alignment with the average of its bank position

(A4)

(M7) then within two hours, perform the following: (ONE)

[RAB.1] a. Correct the situation, or

[RAB2.4] [RAB.2.5] b. Determine by measurement the hot channel factors and apply Specification 3.10.2.1 or (within 72 hours perform SR 3.2.1.1 + 3.2.2.1) (and) (M8)

[RAB.2.2] c. Limit power to ≤ 70 percent of rated power (Reduce Thermal)

3.10.1.6 [NOTE TO LCO 3.1.5 AND LCO 3.1.6] Insertion limits do not apply during physics tests or during period exercise of individual rods. However, the shutdown margin indicated in Figure 3.10-2 must be maintained, except during the low power physics test to measure control rod worth and shutdown margin. For this test, the reactor may be critical with all but one full length control rod inserted. (See 3.1.5 3.1.6) (See 3.1.8)

3.10.2 Power Distribution Limits

3.10.2.1 At all times except during low power physics tests, the hot channel factors, $F_Q(Z)$ and $F_{\Delta H}$, defined in the basis, must meet the following limits:

$F_Q(Z) \leq (F_Q^{RTP}/P) \times K(Z)$ for $P > 0.5$

$F_Q(Z) < (F_Q^{RTP}/0.5) \times K(Z)$ for $P \leq 0.5$

$F_{\Delta H} < F_{\Delta H}^{RTP} (1 + PF_{\Delta H} (1-P))$

(See 3.2.1 + 3.2.2)

Add MODES 1 + 2 (M28)

Add RA B.2.1.1, B.2.1.2
RA B.2.3, B2.6
RA C.1, D.1.1, D.1.2
RA D.2
(M9)

ITS

from the fully withdrawn position is — M29

A1

3.10.4 Rod Drop Time — verify the rod

[SR 3.1.4.3] 3.10.4.1 The drop time of each control rod shall be not greater than 1.8 seconds at full flow and operating temperature from the beginning of rod motion to dashpot entry. — ≤ — >540°F — with all reactor coolant pumps operating

3.10.5 Reactor Trip Breakers

3.10.5.1 The reactor shall not be made critical unless the following conditions are met:

a. Two reactor trip breakers are operable.

b. Reactor trip bypass breakers are racked out or removed.

c. Two trains of automatic trip logic are operable.

Decay of stationary gripper coil voltage — M24

3.10.5.2 During power operation, the requirements of 3.10.5.1 may be modified to allow the following components to be inoperable. If the system is not restored to meet the requirements of 3.10.5.1, the reactor shall be placed in the hot shutdown condition utilizing normal operating procedures within the next 8 hours.

a. One reactor trip breaker may be inoperable for up to 12 hours.

b. One train of automatic trip logic may be inoperable for up to 12 hours.

c. One reactor trip bypass breaker may be racked in and closed for up to 12 hours.

See 3.3.1

3.10.5.3 With one of the diverse trip features inoperable (shunt trip attachment/undervoltage trip attachment) on one of the reactor trip breakers, power operation may continue for up to 48 hours. If the

M.26

Add Condition A, associated actions, and completion times for control rod drop time not within limits

Supplement 5

3.1.3 Minimum Conditions for Criticality

3.1.3.1 Except during low power physics tests, the reactor shall not be made critical at any temperature at which the moderator temperature coefficient is outside the limits specified in the CORE OPERATING LIMITS REPORT (COLR). The maximum upper limits shall be less than or equal to:

a) +5.0 pcm/°F at less than 50% of rated power, or

b) 0 pcm/°F at 50% of rated power and above.

See 3.1.3

3.1.3.2 In no case shall the reactor be made critical above and to the left of the criticality limit shown on Figure 3.1-1.

See 3.4.2

3.1.3.3 When the reactor coolant temperature is in a range where the moderator temperature coefficient is outside the limits specified in the COLR, the reactor shall be made subcritical by an amount equal to or greater than the potential reactivity insertion due to depressurization.

See 3.1.3

3.1.3.4 The reactor shall be maintained subcritical by at least 1% until normal water level is established in the pressurizer.

See 3.4.9

Basis

During the early part of fuel cycle, the moderator temperature coefficient may be slightly positive at low power levels. The moderator temperature coefficient at low temperatures or powers will be most positive at the beginning of the fuel cycle, when the boron concentration in the coolant is the greatest. At all times, the moderator temperature coefficient is calculated to be negative in the high power operating range, and after a very brief period of power operation, the coefficient will be negative in all circumstances due to the reduced boron concentration as Xenon and fission products build into the core. The requirement that the reactor is not to be made critical when the moderator temperature coefficient outside the limits specified in the COLR has been imposed to prevent any unexpected power excursion during normal operations as a result of either an increase in moderator temperature or decrease in coolant pressure. This requirement is

A7

M22

Add Applicability: MODE 2 during PHYSICS TESTS

Specification 3.1.8

A1

3.10 REQUIRED SHUTDOWN MARGINS, CONTROL ROD, AND POWER DISTRIBUTION LIMITS

~~Applicability~~

~~Applies to the required shutdown margins, operation of the control rods, and power distribution limits.~~

~~Objective~~

~~To ensure (1) core subcriticality after a reactor trip and during normal shutdown conditions, (2) limited potential reactivity insertions from a hypothetical control rod ejection, and (3) an acceptable core power distribution during power operation.~~

Specification

3.10.1 Full Length Control Rod Insertion Limits

3.10.1.1 (Deleted by Change No. 21 issued 7/6/73)

[LCO 3.1.8]

3.10.1.2 When the reactor is critical, except for physics tests ~~and~~ full length control rod exercises, the shutdown control rods shall be limited in physical insertion as specified in the CORE OPERATING LIMITS REPORT (COLR).

See 3.1.5 3.1.6

[LCO 3.1.8]

3.10.1.3 When the reactor is critical, except for physics tests ~~and~~ full length control rod exercises, the control rods shall be limited in physical insertion as specified in the COLR. Control rod bank insertion beyond the limits specified in the COLR shall be corrected within the time criteria established by the axial power distribution methodology or within one (1) hour, whichever occurs sooner. If bank insertion is not restored to the specified limits (i.e., within one (1) hour or within the time criteria established by the axial power distribution methodology, whichever is sooner) the reactor shall be placed in the hot shutdown condition utilizing normal operating procedures within six (6) hours.

3.10.1.4 At 50 percent of the cycle as defined by burnup, the limits shall be adjusted to the end-of-core values as specified in the COLR.

Add RAs A1, A.2, B.1, C.1, D.1 — M20

Add SR 3.1.8.1
SR 3.1.8.2
SR 3.1.8.3
SR 3.1.8.4 — M21

Add LCO 3.1.8 requirements a, b, and c — M22

Amendment No. ~~141~~, 167

Supplement 5

ADMINISTRATIVE CHANGES

A1 In the conversion of the H.B. Robinson Steam Electric Plant (HBRSEP), Unit No.2 Current Technical Specification (CTS) to the proposed plant specific Improved Technical Specifications (ITS) certain wording preferences or conventions are adopted which do not result in technical changes (either actual or interpretational). Editorial changes, reformatting, and revised numbering are adopted to make ITS consistent with the conventions in the - Standard Technical Specifications, Westinghouse Plants NUREG-1431, Rev 1 (i.e., Improved Standard Technical Specifications (ISTS)).

A2 CTS 3.10.8.1 requires that the shutdown margin (SDM) be within the limits specified in figure 3.10-2 when the reactor is in hot shutdown. CTS 3.10.8.2 requires that the SDM be at least 1 percent $\Delta k/k$ when the reactor is in cold shutdown. ITS 3.1.1 specifies that the SDM be maintained within the limits provided in the Core Operating Limits Report (COLR) with a specified applicability of MODE 2 with k_{eff} <1.0 and MODES 3, 4 and 5. The CTS definition of hot shutdown encompasses ITS MODE 2 with k_{eff} <1.0 and ITS MODES 3 and 4. The CTS definition of cold shutdown is consistent with ITS MODE 5. Therefore, this is an administrative change resulting from combining CTS Specifications 3.10.8.1 and 3.10.8.2.

A3 Not used

A4 In addition to rewording and other editorial changes necessary to the conversion (DOC A1 above) CTS 3.10.5 is reworded in the ITS to clarify current licensing basis requirements. The wording "for bank demand positions" is used to replace the less precise terminology in the introductory phrase for each of the two cases beginning with "at positions . . ." This substitution is necessary to remove ambiguity regarding whether these statements are referring to bank demand or actual rod positions. Additionally, the terminology ". . . with its bank demand position . . ." is used to replace the less precise term " . . . bank position . . ." for the condition of ≥ 200 steps. This substitution is made to clarify that, for this condition, the applicable reference position is bank demand position. For the condition < 200 steps, the terminology ". . . the average of the individual rod positions . . ." is substituted for ". . . the average if its bank position . . ." to clarify that, for this condition, the applicable reference position is the actual average rod position. These are administrative changes made to eliminate confusion and ambiguity regarding application of these requirements.

A5 Not used.

A6 CTS 3.10.6.1 requires a rod misaligned by more than 15 inches from its bank to be declared inoperable and CTS 3.10.6.2 allows one control rod to be inoperable during power operation. ITS 3.1.4 ACTION B allows only one rod to be misaligned. CTS 3.10.1.5 (and ITS 3.1.4) requires for bank demand positions ≥ 200 steps that each rod shall be within 15 inches of its bank demand position and for bank demand positions < 200 steps that each rod shall be within 7.5 inches of the average of the individual rod positions in the bank. If either of these limits are not met, CTS 3.10.1.5 (and ITS 3.1.4 ACTION B) requires action to be taken. Since the change reflects a presentation preference (differentiating between an inoperable rod and a misaligned rod and only allowing one misaligned rod) and is consistent with current plant interpretation, the change is considered to be administrative.

A7 The CTS Bases are not retained in the ITS, but are replaced in their entirety. The ITS includes significantly expanded and improved Bases. The Bases do not define or impose any specific requirements but serve to explain, clarify and document the reasons (i.e., Bases) for the associated specification. The Bases are not part of the Technical Specifications required by 10 CFR 50.36. Therefore, this is an administrative change.

A8 Not used.

M20 CTS required actions comparable to ITS 3.1.8 RAs A.1, A.2, B.1, C.1 and D.1 do not exist. Lacking specified actions, failure to satisfy CTS 3.1.3.1, 3.10.1.2, 3.10.1.3 or 3.10.1.5 requires compliance with CTS 3.0. CTS 3.0 requires hot shutdown in 8 hours and cold shutdown in 30 hours. ITS 3.1.8 RAs A.1, A.2, B.1, C.1 and D.1 mandate actions in shorter times (i.e., either immediately, 15 minutes or 1 hour). When operating in a physics test exception and SDM is not within limits, RA A.1 requires initiation within 15 minutes of boration to restore SDM within limits and RA A.2 requires suspending the physics test exception within 1 hour. These Required Actions are necessary to require prompt restoration of SDM to within limits as well as promptly restore each applicable LCO to within specification. When operating in a physics test exception and THERMAL POWER is not within limits, RA B.1 requires immediately opening the reactor trip breakers. Opening the reactor trip breakers is necessary to prevent operating the reactor beyond its design limits. When operating in a physics test exception and lowest loop average is not within limits, RA C.1 requires restoring lowest loop average temperature within limits within 15 minutes. This action is necessary to prevent the unit from remaining in a unacceptable condition for an extended period of time. These are additional restrictions on plant operation and are consistent with NUREG-1431.

M21 CTS surveillance requirements comparable to ITS SRs 3.1.8.2, 3.1.8.3, and 3.1.8.4 do not exist. SR 3.1.8.2 requires periodic verification that lowest loop average temperature is ≥ 530°F. SR 3.1.8.3 requires periodic verification that SDM is within limits. SR 3.1.8.4 requires periodic verification that THERMAL POWER is ≤ 5% RTP. In addition, SR 3.1.8.1 is added to require the performance of a CHANNEL OPERATIONAL TEST on the power range and intermediate range channels within 7 days prior to initiation of PHYSICS TESTS. CTS does not require performance of a CHANNEL FUNCTIONAL TEST (an ITS CHANNEL OPERATIONAL TEST) within a specified time frame prior to initiation of PHYSICS TESTS. Specifying a time limit for performance of the CHANNEL OPERATIONAL TEST represents an additional restriction on plant operation necessary to ensure the required RPS instrumentation is OPERABLE. These SRs are necessary to periodically confirm unit operation is within the limits of the LCO. These additional SRs are additional restrictions on plant operation and are consistent with NUREG-1431.

M22 Physics tests exceptions included in CTS 3.1.3.1, 3.10.1.2, 3.10.1.3 and 3.10.1.5 do not specify any additional restriction when applying the exception. ITS 3.1.8 imposes additional requirements regarding RCS loop temperatures, THERMAL POWER and SDM requirements. The inclusion of these additional restrictions is necessary to ensure operation is within the bounds of the applicable safety analysis. The adoption of these requirements is an additional restriction on plant operation and is consistent with NUREG-1431.

M23 CTS 3.10.1.6 allows SDM to not be maintained during the low power physics test used to measure control rod worth and SDM. It is not necessary to retain this SDM exception since the measurement technique necessitating the SDM exception is no longer used. While this method is no longer used at HBRSEP Unit No. 2, the CTS still provide the option for its use. The HBRSEP Unit No. 2 ITS does not include this optional allowance. Therefore, this change represents an additional restriction on plant operations through the deletion of an allowed exception to a Limiting Condition for Operation.

M24 CTS specifies measurement of control rod timing ". . . from the beginning of rod motion until dashpot entry." ITS specifies ". . . from the decay of stationary gripper coil voltages." The inclusion of the time from the beginning of stationary gripper coil voltage decay is necessary to ensure timing the complete rod trip sequence. This is an additional restriction on plant operation and is consistent with the NUREG-1431.

M25 CTS 3.10.6.3 establishes the action for one control rod inoperable to include changing the boron concentration to obtain an appropriate SDM but does not specify a time limit. RA A.1.1 require verification that SDM is within limits within one hour. RA A.1.2 requires initiation of boration within one hour to restore SDM to within limits. Requiring either verification of SDM or initiation of action to restore SDM is necessary since available SDM may be significantly reduced. The one hour time limit is necessary to promptly require verification or restoration of SDM to within limits. If any Required Action and Completion Time is not met RA A.2 requires the plant be placed in MODE 3 in 6 hours. This Action is necessary to place the unit in a MODE outside the Applicability of the specification. ITS requirement of placing the unit in a MODE outside the Applicability within 6 hours is more restrictive than CTS 3.0 which allows 8 hours to be outside the MODE of applicability. These changes are more restrictive and are consistent with NUREG 1431.

M26 CTS 3.10.4.1 establishes the requirement for control rod drop times but does not establish a related action if the drop times are not met. In this case Specification 3.0 would be entered. ITS 3.1.4 Condition includes control rod drop times not met and the associated Actions and Completion Times apply. These actions require that either the SDM must be verified to be within limit or the boron concentration must be restored within the limit specified in the COLR, within one hour. Requiring either verification of SDM or initiation of action to restore SDM is necessary since available SDM may be significantly reduced. The one hour time limit is necessary to promptly require verification or restoration of SDM to within limits. This change is necessary to establish consistency with other similar shutdown requirements stated in

other specifications and is based on operating experience which indicates the times to place the unit in the specified MODE is reasonable. The time limit specified to reach MODE 3 permits the shutdown to proceed in a controlled and orderly manner that is within the capabilities of the unit, assuming that only the minimum required equipment is OPERABLE. This reduces the potential for a plant upset that could challenge safety systems under conditions to which this Specification applies. These changes are more restrictive and are consistent with NUREG 1431.

M27 CTS Table 4.1-1 items 9 and 10 (Analog Rod Position and Rod Position Bank Counters) require a CHANNEL CHECK to be performed. The associated remarks of CTS Table 4.1-1 (Remark 2 for item 9 and Remark 1 for item 10) require the CHANNEL CHECK to be performed following rod motion in excess of six inches when the computer is out of service. ITS SR 3.1.7.1 requires this same CHANNEL CHECK to be performed "Once within 4 hours" following > 6 inches of rod motion when the rod position of the SR represents an additional restriction on plant operation necessary to ensure the surveillance is completed within a reasonable time period after rod motion is complete. The four hour time period is consistent with time period provided in ITS SR 3.1.4.1 for verifying individual rod positions are within alignment limits when the rod position deviation monitor is inoperable.

M28 CTS 3.10.1.5 and 3.10.6, which provide requirements for control rod alignment limits and OPERABILITY, do not include explicit operating condition applicability statements. ITS 3.1.4, Rod Group Alignment Limits, adds an explicit applicability of MODES 1 and 2. Control rod OPERABILITY and alignment limits are required in MODES 1 and 2 because these are the MODES in which neutron (or fission) power is generated and the OPERABILITY (i.e., trippability) and alignment of rods have the potential to affect safety of the plant. MODES 1 and 2 are reasonable interpretations of the implicit applicability of CTS 3.10.1.5 and 3.10.6. These requirements are in addition to those in the CTS, and thus, represent a more restrictive change.

M29 CTS 3.10.4.1 establishes the parameters for rod drop time testing but does not specifically address the rod position just prior to testing. ITS SR 3.1.4.3 includes the requirement "from the fully withdrawn position" which is implied by the CTS but not specifically stated. Requiring the affected rod to be rod drop time tested from the fully withdrawn position ensures that safety analysis assumptions related to rod drop times are adequately verified. This requirement is in addition to that in the CTS, and thus, represent a more restrictive change.

TECHNICAL CHANGES - LESS RESTRICTIVE (GENERIC)

LA1 CTS Specifications 3.10.8.1, 3.10.8.2, and Figure 3.10-2 provide required shutdown margin values. CTS Specification 3.10.1.4 requires control rod insertion limits be adjusted to the end-of-core values as provided in the COLR at 50 percent of the cycle. These details are not retained in the ITS and are relocated to licensee controlled documents.

The details associated with the involved Specification are not required to be in the ITS to provide adequate protection of the public health and safety because the ITS still retains the requirement for compliance with the limits, and ITS Section 5.6 specifies the scope of the limits contained in the COLR and mandates NRC approval of the analytical methodology. This approach provides an effective level of regulatory control and provides for a more appropriate change control process. The level of safety of facility operation is unaffected by the change because there is no change in the overall operational requirements. Furthermore, NRC and licensee resources associated with processing license amendments to these requirements will be reduced. Therefore, relocation of these details is acceptable.

LA2 CTS 3.10.1.3 requires that the reactor be placed in hot shutdown, and specifies that this be accomplished, "using normal operating procedures." This detail, specifying the manner in which to achieve hot shutdown, is to be relocated to Bases. Changes to the Bases will be controlled by the provisions of the Bases Control Program described in Chapter 5 of the ITS.

The details associated with the involved Specification are not required to be in the ITS to provide adequate protection of the public health and safety because the ITS still retains the requirement for compliance with the Action. This approach provides an effective level of regulatory control and provides for a more appropriate change control process. The level of safety of facility operation is unaffected by the change because there is no change in the overall operational requirements. Furthermore, NRC and licensee resources associated with processing license amendments to these requirements will be reduced. Therefore, relocation of these details is acceptable.

TECHNICAL CHANGES - LESS RESTRICTIVE (SPECIFIC)

L1 Since there is no action specified for failing to satisfy CTS 3.10.8.1 or 3.10.8.2, the required action is controlled by CTS 3.0. This CTS specification requires the unit be placed in hot shutdown within 8 hours followed by cold shutdown in an additional 30 hours. ITS 3.1.1 RA A.1, specifies initiating boration within 15 minutes to restore the SDM

within limits. Both actions result in the addition of negative reactivity and a return to compliance with the assumptions of the safety analysis. ITS 3.1.1 RA A.1 requires timely restoration of SDM. Timely restoration of SDM is preferred to imposing the increased risk associated with a plant shutdown transient. Additionally, mandating shutdown of the unit may not be the safest course of action while sufficient SDM is not available. The proposed change provides an appropriate specific action for failing to satisfy the LCO instead of applying the generic action mandated by CTS 3.0. This change is consistent with NUREG-1431.

L2 This change involves two separate aspects both of which are analyzed separately here.

With the MTC outside the limits provided in the COLR, CTS 3.1.3.3 requires the reactor be made subcritical by an amount greater than or equal to the potential reactivity insertion due to depressurization. Since no completion time is explicitly stated, this specification implies completion as soon as practical. (Although not directly applicable, CTS 3.0 requires hot shutdown within 8 hours. Without a explicit statement of completion time, the comparable completion time in of 8 hours in CTS 3.0 is considered implicitly binding.) With MTC not within the upper limit, ITS 3.1.3 RA A.1 mandates establishment of administrative withdrawal limits for control banks to maintain MTC within the upper limit with a completion time of 24 hours. Provided ITS 3.1.3 RA A.1 is satisfied, no further action is required. While not explicitly stated, establishment of administrative withdrawal limits for control banks to maintain MTC within the upper limit is not precluded by CTS. However, the completion time of 24 hours to establish administrative control banks withdrawal limits is less restrictive than CTS permits.

With the required action or associated completion time of ITS 3.1.3 RA A.1 not met, ITS 3.1.3 RA B.1 mandates being in MODE 2 with $K_{eff} < 1.0$ within 6 hours. This completion time is in addition to the 24 hours permitted by ITS 3.1.3 RA A.1, and is less restrictive than CTS permits.

A specific Completion Time is added to CTS 3.1.3.3 to restore MTC to within the upper limit (i.e., to "Establish administrative withdrawal limits for control banks to maintain MTC within limits"). Evaluating the MTC measurement and obtaining the necessary input to compute the necessary bank withdrawal limits necessary to restore compliance with the MTC limits may require a time period much longer than 8 hours, the current implied time to place the plant in a non-applicable MODE. The completion time of 24 hours for ITS 3.1.3 Required Action A.1 provides sufficient time for evaluating the MTC measurement and computing the required bank withdrawal limits. Additionally, the 24 hour Completion

Time is based on the low probability of an accident occurring during this period and takes into consideration the fact that as cycle burnup is increased, RCS boron concentration is reduced which causes MTC to become more negative. This change also provides the benefit of not hastily inducing a plant shutdown transient while in a condition where unit response during postulated events may not be as predicted (due to MTC not being within the upper limit).

With MTC outside the limits provided in the COLR, CTS 3.1.3.3 mandates being subcritical by an amount equal to the potential reactivity insertion due to depressurization. With MTC outside the upper limit, ITS 3.1.3 RA B.1 mandates, assuming ITS 3.1.3 RA A.1 and associated completion time not met, being in MODE 2 with $K_{eff} < 1.0$. In this condition, the SDM requirements of ITS LCO 3.1.1 are applicable requiring the SDM be within the limits provided in the COLR. The COLR includes appropriate SDM limits for this condition. Therefore this aspect of the change is administrative in nature.

L3 CTS Table 4.1-3, Item 2 requires verification of each control rods freedom of movement every 14 days during reactor critical operations. ITS SR 3.1.4.2 requires this surveillance to be performed at a 92 day Frequency and excludes control rods that are fully inserted. This is a relaxation of requirements, and is less restrictive. This change is acceptable, however, because the 92 day Frequency takes into consideration other information available to the operator in the control room, and performance of SR 3.1.4.1, which verifies that individual rod positions are within alignment limits every 12 hours and adds to the determination of OPERABILITY of the rods. In addition, not requiring fully inserted rods to be exercised is less restrictive than the CTS which does not have this exception. The intent of the exercise test is to provide assurance that the rod remains trippable (i.e., the rod is not stuck in the withdrawn position); thereby helping to assure that adequate Shutdown Margin is maintained. Not requiring fully inserted rods to be exercised is considered acceptable since with the rod in the fully inserted, it is not possible for the rod to be stuck in the withdrawn position. Therefore, elimination of the requirement to exercise fully inserted rods has no impact on the ability to maintain adequate Shutdown Margin. This change is consistent with NUREG-1431.

L4 For control rod banks inserted in excess of the specified insertion limits, CTS 3.10.3 requires correction within one hour. ITS 3.1.6 RA A.2 permits two hours to restore the banks within limits. However, ITS also requires verification of SDM or initiation of boration to restore SDM within limits within one hour (see related DOC M17). If control rod banks are not within specified insertion limits, SDM may be adversely impacted. Requiring the verification of SDM or the initiation of boration to restore SDM within one hour provides assurance that SDM is

adequate or that action is taken restore SDM. Therefore, the additional one hour time period to restore control banks to within specified insertion limits is considered to be acceptable since the impact of the non-compliance on SDM is minimized. Requiring the verification of SDM or the initiation of boration to restore SDM within one hour in concert with the restoration of control banks to within specified insertion limits within two hour provides some additional time to correct the condition while still restricting operation in this condition to a reasonably short time period. Prompt restoration of the control rod banks to within insertion limits is preferable to a plant shutdown with the associated risk of shutdown transients. This change is consistent with NUREG-1431.

L5 With the MTC outside the limits provided in the COLR, CTS 3.1.3.3 requires the reactor be made subcritical by an amount greater than or equal to the potential reactivity insertion due to depressurization. Since no completion time is explicitly stated, this specification implies completion as soon as practical. (Although not directly applicable, CTS 3.0 requires hot shutdown within 8 hours. Without an explicit statement of completion time, the comparable completion time of 8 hours in CTS 3.0 is considered implicitly binding.) With MTC not within the lower limit, ITS 3.1.3 RA C.1 mandates being in MODE 4 with a completion time of 12 hours. This completion time is more than the implicit completion time for CTS 3.1.3.3. This change allows for a more controlled shutdown which reduces thermal stress on components and also reduces the chances for a plant transient which could challenge safety systems. The additional 4 hours to reach MODE 4 is considered reasonable, based on operating experience, to reach MODE 4 from full power conditions in an orderly manner and without challenging plant systems. The requirement to be in MODE 4 is more restrictive than the CTS 3.1.3.3 requirement to be subcritical by an amount greater than or equal to the potential reactivity insertion due to depressurization. This change is consistent with NUREG-1431.

L6 In the event the rod position indication requirements of CTS Table 4.1-1 items 9 and 10 are not satisfied, the CTS 3.10.1.5 actions associated with a misaligned rod are required to be taken within 2 hours. Rod position indication instruments do not necessarily relate directly to rod OPERABILITY (e.g. rods aligned within limits) or the ability to maintain rods within alignment limits. As such, it is overly restrictive to assume that rods are misaligned when rod position indication is inoperable. Therefore, ITS 3.1.4 is added to require the Analog Rod Position Indication (ARPI) System and the Demand Position Indication System to be OPERABLE in MODES 1 and 2 and provide alternate ACTIONS to determine rod position or reduce power to ≤ 50% RTP in the

event of inoperable rod position indication. Reducing power to ≤ 50% RTP puts the core into a condition where rod position is not significantly affecting core peaking factors. The ACTIONS of ITS 3.1.7 are modified by a Note which allows separate Condition entry for each inoperable rod position indicator per group and each demand position indicator per bank. This Note is acceptable since in conjunction with ITS 1.3, "Completion Times," the ITS 3.1.7 ACTIONS provide appropriate compensatory actions for each inoperable position indicator. In the event one ARPI per group is inoperable for one or more groups, ITS 3.1.7 Required Actions A.1 and A.2 require verification of the position of rods with inoperable position indication by using the movable incore detectors once per 8 hours or require reduction of thermal power within 8 hours. With one or more rods with inoperable position indicators moved in excess of 24 steps in one direction since the last determination of the rod's position, ITS 3.1.7 Required Actions B.1 and B.2 require verification of the position of rods with inoperable position indication by using the movable incore detectors once per 8 hours or require reduction of thermal power within 8 hours. In the event one demand position indicator per bank is inoperable for one or more banks, ITS 3.1.7 Required Actions C.1.1, C.1.2, C.1.3, and C.2 require verification by administrative means that the all ARPIs for the affected banks are OPERABLE once per 8 hours and require verification that the position of each rod in the affected bank(s) is within required limits (the limits of ITS 3.1.4) once per 8 hours; or require reduction of thermal power within 8 hours. If any of these Required Actions are not met within the associated Completion Time, Required Action D.1 requires the plant to be placed in MODE 3 (a non-applicable MODE) within 6 hours. The time periods provided for completing the Required Actions and are considered to be acceptable based on the low probability of having a rod significantly out of position and an event sensitive to that rod position during the time periods.

L7 CTS 4.9 requires submittal of a Special Report within 30 days if the difference between observed and predicted steady-state boron concentration reaches the equivalent of 1 percent Δk/k. This requirement is not retained in the ITS. In addition, a 1 percent Δk/k reactivity anomaly is reasonable equivalent to change of 100 ppm in boron concentration and a 100 ppm boron uncertainty is included in applicable HBRSEP safety analysis. This change deletes the special reporting submittal requirement associated with CTS 4.9. Instead, reporting will be governed by the requirements of 10 CFR 50.73. The HBRSEP Unit No.2 ITS 3.1.2 Required Actions do not require special reporting in this instance. This change is acceptable because the special reporting requirement of CTS 4.9 is not necessary to assure operation in a safe manner (core reactivity is either restored within the required time period or the unit is shutdown) and there is no requirement for the NRC to approve the report. Therefore, this change

has no impact on the safe operation of the plant. Additionally, deletion of the reporting requirement of CTS 4.9 reduces the administrative burden on the plant and allows efforts to be concentrated on restoring core reactivity to within limits.

RELOCATED SPECIFICATIONS

R1 3.10.7 Power Ramp Rate Limits

This Specifications, or Limiting Conditions for Operation (Chapter 3.0), is not retained in the ITS because it has been reviewed against, and determined not to satisfy, the selection criteria for Technical Specifications provided in 10 CFR 50.36. The selection criteria were established to ensure that the Technical Specifications are reserved for those conditions or limitations on plant operation considered necessary to limit the possibility of an abnormal situation or event that could result in an immediate threat to the health and safety of the public. The rationale for relocation of this Specification is provided in the report, "Application of Selection Criteria to the H. B. Robinson Steam Electric Plant Unit No. 2 Technical Specifications."

This Limiting Conditions for Operation, is relocated to licensee controlled documents. Relocation of the specific requirements for systems or variables contained in these Specifications to licensee documents will have no impact on the operability or maintenance of those systems or variables. The licensee will initially continue to meet the requirements contained in the relocated Specifications. The licensee is allowed to make changes to these requirements in accordance with the provisions of 10 CFR 50.59. Such changes can be made without prior NRC approval, if the change does not involve an unreviewed safety question, as defined in 10 CFR 50.59. These controls are considered adequate for assuring that structures, systems, and components in the relocated Specifications are maintained operable, and variables are maintained within limits. This change is consistent with the NRC Final Policy Statement on Technical Specification Improvements.

L6 Change

Carolina Power & Light Company 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 change eliminates the requirement to consider rods to be misaligned when rod position indication is inoperable by providing an LCO and associated ACTIONS for rod position indication. This change does not result in any hardware changes. The rod position indication instruments are not initiators of any analyzed event. The role of this instrumentation is in providing the operators information to allow them to determine rod positions and thereby ensure compliance with rod alignment and insertion limits. The requested change does not allow continuous operation in this condition without implementing an alternate method of determining rod position or reducing power to a level where rod position does not significantly affect core peaking factors. Additionally, the consequences of an event occuring with the proposed actions are no more severe than the consequences of an event occurring within the allowed outage time of the current actions. Therefore, this change will 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 introduce a new mode of plant operation and does not involve physical modification to the plant. The proposed change continues to assure that rod position can be determined or requires a power reduction to a level where rod position does not significantly affect core peaking factors. Therefore it does not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

   The change eliminates the requirement to consider rods to be misaligned when rod position indication is inoperable by providing an LCO and associated ACTIONS for rod position indication. The proposed change is acceptable based on the small probability of having a rod significantly out of position and an event sensitive to that rod position during the time period allowed to either implement an alternate method of determining rod position or reducing power to a level where rod position does not significantly affect core peaking factors. Rod position indication instruments do not necessarily relate directly to rod OPERABILITY or the ability to maintain rods within alignment limits. As such, it is overly

restrictive to assume that rods are inoperable (e.g., misaligned) when rod position indication is inoperable. Providing the proposed Actions will minimize the potential for plant transients that can occur during a power reduction or shutdown by providing additional time for (and the option of) implementation of an alternate means of determining rod position when rod position indication is inoperable. In addition, if the alternate method of determining rod position is not implemented within the time frame established in the Required Actions of ITS 3.1.7, ITS 3.1.7 Required Action D.1 would require a shutdown to MODE 3. Requiring a shutdown to MODE 3 places the unit in a non-applicable MODE. As such, any reduction in a margin of safety resulting from the proposed change will be offset by the potential benefit gained by avoiding an unnecessary plant power reduction or shutdown transient when alternate means exist to determine rod position. Therefore, this change does not involve a significant reduction in a margin of safety.

L7 Change

Carolina Power & Light Company 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 deletes the special reporting requirement when the difference between the observed and predicted steady-state concentrations reaches the equivalent of one percent in reactivity. Instead, reporting will be governed by the requirements of 10 CFR 50.73. Reporting requirements are not assumed to be initiators of any analyzed event and do not impact assumptions of any design basis accident. Additionally, the submittal of a special report is not required or assumed for the mitigation of any accident. Therefore, this proposed change does not involve an increase in the probability or consequences of an accident previously evaluated.

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

   This change does not introduce a new mode of plant operation and does not involve a physical modification to the plant. This change deletes a special reporting requirement. Therefore, it does not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

   This change is acceptable because the special reporting requirement deleted by this change is not necessary to assure operation in a safe manner and there is no requirement for the NRC to approve the report.

Therefore, this change has no impact on the safe operation of the plant. Additionally, the report will still be required per 10 CFR 50.73 if results of the core reactivity re-evaluation (required by ITS 3.1.2 Required Action A.1) indicate plant operation in an unanalyzed condition. Deletion of the above reporting requirement also reduces the administrative burden on the plant and allows efforts to be concentrated on restoring core reactivity within limits. Therefore, the deletion of the special reporting requirement does not involve a significant reduction in a margin of safety.

RELOCATED CHANGES
("R" Labeled Comments/Discussions)

Relocating Requirements which do not meet the Technical Specification criteria to documents with an established control program allows the Technical Specifications to be reserved only for those conditions or limitations upon reactor operation which are necessary to adequately limit the possibility of an abnormal situation or event giving rise to an immediate threat to the public health and safety, thereby focusing the scope of Technical Specifications.

Therefore, requirements which do not meet the Technical Specification criteria in the NRC Final Policy Statement on Technical Specification Improvement for Nuclear Power Reactors (58FR 39132, dated 7/22/93) have been relocated to licensee controlled documents. This policy statement addresses the scope and purpose of Technical Specifications. In doing so, it establishes a specific set of objective criteria for determining which regulatory requirements and operating restrictions should be included in Technical Specifications. These criteria are as follows:

Criterion 1: Installed instrumentation that is used to detect and indicate in the control room, a significant abnormal degradation of the reactor coolant pressure boundary;

Criterion 2: A process variable that is an initial condition of a design basis accident (DBA) or transient analyses that either assumes the failure of or presents a challenge to the integrity of a fission product barrier;

Criterion 3: A structure, system or component that is part of the primary success path and which functions or actuates to mitigate a design basis accident or transient that either assumes the failure of or presents a challenge to the integrity of a fission barrier;

Criterion 4: A structure, system or component which operating experience or probabilistic safety assessment has shown to be significant to public health and safety.

The application of these criteria is provided in the "Application of Screening Criteria to the HBRSEP Unit No. 2 Technical Specifications." Requirements which met the criteria have been included in the proposed improved Technical Specifications. Carolina Power & Light (CP&L) proposes to remove the requirements which do not meet the criteria from the Technical Specifications and relocate the requirements to a suitable owner controlled document. The requirements in the relocated Specifications are not affected by this Technical Specification change. CP&L will initially continue to perform the required operation and maintenance to assure that the requirements are satisfied. Relocating specific requirements for systems or variables has no impact on the system's operability or the variable's maintenance, as applicable.

Licensee controlled programs will be utilized as the control mechanism for the relocated Specifications as they will be placed in plant procedures or other licensee controlled documents. CP&L is allowed to make changes to these requirements, without prior NRC approval, if the change does not involve an unreviewed safety question. These controls are considered adequate for assuring structures, systems and components in the relocated Specifications are maintained operable and variables in the relocated Specifications are maintained within limits.

Carolina Power & Light Company has evaluated each of the proposed Technical Specification changes identified as "Relocated" and has concluded that they do 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 proposed changes do 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 relocates requirements and surveillances for structures, systems, components or variables which did not meet the criteria for inclusion in Technical Specifications as identified in the "Application of Selection Criteria to the HBRSEP Unit No. 2 Technical Specifications." The affected structures, systems, components or variables are not assumed to be initiators of analyzed events and are not assumed to mitigate accident or transient events. The requirements and surveillances for these affected structures, systems, components or variables will be relocated from the Technical Specifications to an appropriate administratively controlled document under licensee control. Therefore, this change does not involve an increase in the probability or consequences of an accident previously evaluated.

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

   The proposed change does not necessitate a physical alteration of the plant (no new or different type of equipment will be installed) or change in parameters governing normal plant operation. The proposed change will not impose any different requirements and adequate control of information will be maintained. Thus, this change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

   The proposed change will not reduce a margin of safety because it has no impact on any safety analysis assumptions. In addition, the affected requirement will be relocated to an owner controlled document for which future changes will be evaluated pursuant to the requirements of licensee controlled programs. Therefore, this change does not involve a reduction in a margin of safety.

CTS

3.1 REACTIVITY CONTROL SYSTEMS

3.1.10 (8) PHYSICS TESTS Exceptions – MODE 2

LCO 3.1.10 During the performance of PHYSICS TESTS, the requirements of

[3.1.3.1] LCO 3.1.4 (3) "Moderator Temperature Coefficient (MTC)";
[3.10.1.5] LCO 3.1.5 (4) "Rod Group Alignment Limits";
[3.10.1.2] LCO 3.1.6 (5) "Shutdown Bank Insertion Limits";
[3.10.1.3] LCO 3.1.7 (6) "Control Bank Insertion Limits"; and
[3.1.3.1] LCO 3.4.2. "RCS Minimum Temperature for Criticality"

may be suspended, provided:

[M22] a. RCS lowest loop average temperature is ≥ ~~[531]~~ (530) °F; and

[3.10.1.6] b. SDM is ~~≥ 1.6% Δk/k~~ within limits provided in the COLR; and (2)

c. THERMAL POWER is < 5% RTP. (TSTF-14)

APPLICABILITY: MODE 2 during PHYSICS TESTS.

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
[M20] A. SDM not within limit.	A.1	Initiate boration to restore SDM to within limit.	15 minutes
	AND		
	A.2	Suspend PHYSICS TESTS exceptions.	1 hour
[M20] B. THERMAL POWER not within limit.	B.1	Open reactor trip breakers.	Immediately

(continued)

ACTIONS (continued)

	CONDITION	REQUIRED ACTION	COMPLETION TIME
[M20]	C. RCS lowest loop average temperature not within limit.	C.1 Restore RCS lowest loop average temperature to within limit.	15 minutes
[M20]	D. Required Action and associated Completion Time of Condition C not met.	D.1 Be in MODE 3.	15 minutes

SURVEILLANCE REQUIREMENTS

	SURVEILLANCE		FREQUENCY
	SR 3.1.~~10~~ 8.1	Perform a CHANNEL OPERATIONAL TEST on power range and intermediate range channels per ~~[~~SR 3.3.1.7, SR 3.3.1.8, and Table 3.3.1-1~~]~~.	Within ~~12 hours~~ 7 days prior to initiation of PHYSICS TESTS (13)
[M21]	SR 3.1.~~10~~ 8.2	Verify the RCS lowest loop average temperature is ≥ ~~[531]~~ 530 °F.	30 minutes
[M21]	SR 3.1.~~10~~ 8.4	Verify SDM is ~~≥ 1.6% Δk/k~~ within limits provided in COLR.	24 hours (2)
[M21]	SR 3.1.8.2	VERIFY THERMAL POWER is ≤ 5% RTP.	30 minutes

TSTF-14

1 ISTS Specification 3.1.2 is not included as a separate specification in the ITS. Since the specific shutdown margin requirements for various plant conditions are relocated to the Core Operating Limits Report (COLR), there is no need for separate specifications for different MODES of Applicability. Consequently, shutdown margin requirements applicable to MODE 5 are included in ITS Specification 3.1.1. This eliminates the need for Specification 3.1.2. Subsequent Specifications are renumbered accordingly.

2 Specific numerical values for SDM located throughout the Technical Specifications have been relocated to the COLR. SDM is a cycle-specific variable similar to Moderator Temperature Coefficient, Rod insertion Limits, Axial Flux Difference, Heat Flux Hot Channel Factor, and Nuclear Enthalpy Rise Hot Channel Factor, which are currently contained in the COLR. In addition, there is an NRC-approved methodology for calculating SDM. Relocating SDM to the COLR provides core design and operational flexibility that can be used for improved fuel management and to solve plant specific issues. If the SDM is retained in the COLR the core design can be finalized after shutdown, when the actual end of cycle burnup is known. This can save redesign efforts if the actual burnup differs from the projected value. Currently, reload design efforts and resolution of plant specific issues are somewhat restricted, since a change in the SDM requires a License Amendment.

3 ISTS Figure 3.1.4-1 is not used in the ITS. The maximum upper limit for MTC consists of two values specified in ITS 3.1.3, obviating any need for the Figure. The MTC values provided in ITS 3.1.3 are consistent with the HBRSEP Unit No. 2 current licensing basis approved in Amendment 87 dated November 7, 1984 and Amendment 121 dated January 9, 1989. These values are also reflected in UFSAR Section 15.0.5 and Table 15.0.5-1.

4 ITS Specification 3.1.5, "Rod Group Alignment Limits," consists of two separate requirements: 1) shutdown and control for rod OPERABILITY (defined in the Bases as the ability to insert on an RPS trip), and 2) indicated position of each rod within 12 steps of its group demand position (i.e., correctly positioned). These requirements have been separated in the LCO and Actions to ensure the appropriate actions are taken for each condition. Condition A wording is broadened from "untrippable" to "inoperable" such that the condition encompasses all rod inoperability conditions. Without this change, it is ambiguous with regard to a rod with a slow drop time but one that is still trippable.

5 ITS 3.1.4 rod group alignment limits are modified to be consistent with current licensing basis.

6 ITS 3.1.4, Required Action B.2.2, which requires reducing THERMAL POWER to ≤ 75 percent RTP, is modified to specify reducing THERMAL POWER to

≤70% RTP, consistent with current licensing basis.

7 ITS SR 3.1.4.3 is modified to reflect a minimum T_{avg} of 540°F for verification of rod drop times, consistent with current licensing basis.

8 The word, "more," is changed to the word, "both," because plant design includes two shutdown banks.

9 ITS Specification 3.1.7, Required Action B.1, requires verification of rod position using the movable incore detectors for rods with inoperable position indications that have moved in excess of 24 steps since the last determination of rod position. The bracketed Completion Time of 4 hours is modified to 6 hours. Since the CTS does not include a comparable requirement, there is no current licensing basis for this value. A Completion Time of 6 hours is considered to be a reasonable time in which to perform the required flux mapping and data analysis. A Completion Time of 6 hours still provides sufficient time to complete alternate Required Action B.2, reduction of THERMAL POWER to ≤ 50% RTP within 8 hours.

10 ISTS Specification 3.1.8, Required Action C.1.2, is modified to provide two actions (ITS 3.1.7 Required Action C.1.2 and C.1.3) to address bank positions < 200 steps and bank positions ≥ 200 steps. This change is necessary to address the two different acceptance criteria associated with bank positions provided in ITS Specification 3.1.4 (for bank demand positions ≥ 200 steps, each rod shall be within 15 inches of its bank demand position; and for bank demand position < 200 steps, each rod shall be within 7.5 inches of the average of the individual rod positions in the bank) and the current licensing basis approved in HBRSEP Unit 2 Amendment No. 48.

11 Not used.

12 ISTS Specification 3.1.9, "PHYSICS TEST Exceptions - MODE 1," is not adopted in the ITS. These physics tests are not performed during post-refueling startup testing. ISTS Specification 3.1.11, "SDM Test Exceptions," is not adopted in the ITS. The use of other rod worth measurement techniques will maintain the shutdown margin during the entire measurement process and still provide the necessary physics data verification. Since the N-1 measurement technique is no longer used, the SDM test exception is not necessary. Subsequent Specifications are renumbered accordingly.

13 ISTS SR 3.1.10.1 (ITS SR 3.1.8.1) is revised to require performance of the required CHANNEL OPERATIONAL TESTS within "7 days" prior to initiation of PHYSICS TESTS instead of within "12 hours" prior to initiation of PHYSICS TESTS. The current licensing basis reflected in the CTS does not currently require performance of a CHANNEL FUNCTIONAL

TEST (an ITS CHANNEL OPERATIONAL TEST) on power range and intermediate range channels within a specified time period prior to initiating PHYSICS TESTS. CTS Table 4.1-1 requires performance of the CHANNEL FUNCTIONAL TEST of the intermediate range channels prior to startup and performance of the CHANNEL FUNCTIONAL TEST of the power range channels bi-weekly. However, current plant practice is perform CHANNEL FUNCTIONAL TESTs of these channels within 7 days prior to initiation of PHYSICS TESTS. This 7 day Frequency has been determined to be sufficient for verification that the power range and intermediate range monitors are properly functioning.

14 ISTS SR 3.1.8.1 requires verification that each [D]RPI agrees within [12] steps of the group demand position for the [full indicated range] of rod travel once per [18 months]. This Surveillance Requirement is not included in the HBRSEP Unit No. 2 ITS. Instead, ITS 3.1.7 (Rod Position Indication) includes SRs 3.1.7.1, 3.1.7.2, 3.1.7.3, and 3.1.7.4. SR 3.1.7.1 requires the performance of a CHANNEL CHECK by comparing analog rod position indication to bank demand position indication. SRs 3.1.7.2 and 3.1.7.3 require a test to be performed to verify the rod position indications read within the required acceptance criteria after moving each full length RCCA bank ≥ 19 steps and returning the banks to their original positions. SR 3.1.7.4 requires the performance of a CHANNEL CALIBRATION of the Analog Rod Position Indication System. These SRs are provided consistent with current plant practice and licensing basis reflected in CTS Table 4.1-1 (items 9 and 10) and approved in HBRSEP Unit 2 Amendment No. 48. Amendment No. 48 approved revised control rod position indication systems misalignment limits and requires the following:

> For bank demand positions ≥ 200 steps, each rod shall be within 15 inches of its bank demand position; and
>
> For bank demand positions < 200 steps, each rod shall be within 7.5 inches of the average of the individual rod positions in the bank.

As such, comparisons between the analog rod position indication and the bank demand position indication are only required for bank positions ≥ 200 steps (ITS SR 3.1.7.1 and ITS SR 3.1.7.3) and the acceptance criteria for the monthly tests vary depending on whether the bank positions are ≥ 200 steps (ITS SR 3.1.7.3) or < 200 steps (ITS SR 3.1.7.2).

BASES

ACTIONS

A.1.1, A.1.2, A.2, B.1.1, B.1.2, and B.2 (continued)

Similarly, if the control banks are found to be out of sequence or in the wrong overlap configuration, they must be restored to meet the limits.

Operation beyond the LCO limits is allowed for a short time period in order to take conservative action because the simultaneous occurrence of either a LOCA, loss of flow accident, ejected rod accident, or other accident during this short time period, together with an inadequate power distribution or reactivity capability, has an acceptably low probability.

The allowed Completion Time of 2 hours for restoring the banks to within the insertion, sequence, and overlaps limits provides an acceptable time for evaluating and repairing minor problems without allowing the plant to remain in an unacceptable condition for an extended period of time.

C.1

If the Required Actions ~~A.1 and A.2 or B.1 and B.2~~ cannot be completed within the associated Completion Times, the plant must be brought to MODE 3 (utilizing normal operating procedures), where the LCO is not applicable. The allowed Completion Time of 6 hours is reasonable, based on operating experience, for reaching the required MODE from full power conditions in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.1.7.1

This Surveillance is required to ensure that the reactor does not achieve criticality with the control banks below their insertion limits.

The estimated critical position (ECP) depends upon a number of factors, one of which is xenon concentration. If the ECP was calculated long before criticality, xenon concentration could change to make the ECP substantially in error. Conversely, determining the ECP immediately before criticality could be an unnecessary burden. There are a number of unit parameters requiring operator attention at

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.1.~~7~~ 6.1 (continued)

that point. Performing the ECP calculation within 4 hours prior to criticality avoids a large error from changes in xenon concentration, but allows the operator some flexibility to schedule the ECP calculation with other startup activities.

SR 3.1.~~7~~ 6.2

With an OPERABLE bank insertion limit monitor, verification of the control bank insertion limits at a Frequency of 12 hours is sufficient to ensure OPERABILITY of the bank insertion limit monitor and to detect control banks that may be approaching the insertion limits since, normally, very little rod motion occurs in 12 hours. If the insertion limit monitor becomes inoperable, verification of the control bank position at a Frequency of 4 hours is sufficient to detect control banks that may be approaching the insertion limits.

SR 3.1.~~7~~ 6.3

When control banks are maintained within their insertion limits as checked by SR 3.1.~~7~~ 6.2 above, it is unlikely that their sequence and overlap will not be in accordance with requirements provided in the COLR. A Frequency of 12 hours is consistent with the insertion limit check above in SR 3.1.~~7~~ 6.2.

REFERENCES

1. ~~10 CFR 50, Appendix A, GDC 10, GDC 26, GDC 28.~~
2. 10 CFR 50.46.
3. ~~4.~~ FSAR, Chapter ~~[15]~~ 15.
4. ~~FSAR, Chapter [15].~~
5. ~~FSAR, Chapter [15].~~

UFSAR, Sections 3.1.2.14, 3.1.2.27, 3.1.2.28, 3.1.2.29, 3.1.2.30, 3.1.2.31, and 3.1.2.32.

Supplement 5

BASES

LCO (continued)

limits is permitted for the purpose of performing PHYSICS TESTS and poses no threat to fuel integrity, provided the SRs are met.

The requirements of LCO 3.1.4, LCO 3.1.5, LCO 3.1.6, LCO 3.1.7, and LCO 3.4.2 may be suspended during the performance of PHYSICS TESTS provided:

a. RCS lowest loop average temperature is ≥ 531 °F; and

b. SDM is ≥ 1.6% Δk/k.

c. THERMAL POWER is ≤ 5% RTP.

APPLICABILITY

This LCO is applicable in MODE 2 when performing low power PHYSICS TESTS. The applicable PHYSICS TESTS are performed in MODE 2 at HZP. Other PHYSICS TESTS are performed in MODE 1 and are addressed in LCO 3.1.9, "PHYSICS TESTS Exceptions – MODE 1."

ACTIONS

A.1 and A.2

If the SDM requirement is not met, boration must be initiated promptly. A Completion Time of 15 minutes is adequate for an operator to correctly align and start the required systems and components. The operator should begin boration with the best source available for the plant conditions. Boration will be continued until SDM is within limit.

Suspension of PHYSICS TESTS exceptions requires restoration of each of the applicable LCOs to within specification.

B.1

When THERMAL POWER is > 5% RTP, the only acceptable action is to open the reactor trip breakers (RTBs) to prevent operation of the reactor beyond its design limits. Immediately opening the RTBs will shut down the reactor and prevent operation of the reactor outside of its design limits.

(continued)

BASES

ACTIONS (continued)

C.1

When the RCS lowest T_{avg} is < 531°F, the appropriate action is to restore T_{avg} to within its specified limit. The allowed Completion Time of 15 minutes provides time for restoring T_{avg} to within limits without allowing the plant to remain in an unacceptable condition for an extended period of time. Operation with the reactor critical and with temperature below 531°F could violate the assumptions for accidents analyzed in the safety analyses.

D.1

If the Required Actions cannot be completed within the associated Completion Time, the plant must be brought to a MODE in which the requirement does not apply. To achieve this status, the plant must be brought to at least MODE 3 within an additional 15 minutes. The Completion Time of 15 additional minutes is reasonable, based on operating experience, for reaching MODE 3 in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.1.10.1

The power range and intermediate range neutron detectors must be verified to be OPERABLE in MODE 2 by LCO 3.3.1, "Reactor Trip System (RTS) Instrumentation." A CHANNEL OPERATIONAL TEST is performed on each power range and intermediate range channel within 12 hours prior to initiation of the PHYSICS TESTS. This will ensure that the RTS is properly aligned to provide the required degree of core protection during the performance of the PHYSICS TESTS. The 12 hour time limit is sufficient to ensure that the instrumentation is OPERABLE shortly before initiating PHYSICS TESTS.

SR 3.1.10.2

Verification that the RCS lowest loop T_{avg} is ≥ 531°F will ensure that the unit is not operating in a condition that could invalidate the safety analyses. Verification of the RCS temperature at a Frequency of 30 minutes during the

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.1.10.2 (continued)

performance of the PHYSICS TESTS will ensure that the initial conditions of the safety analyses are not violated.

INSERT B 3.1.8-1 TSTF-14

SR 3.1.10.3 TSTF-14

The SDM is verified by performing a reactivity balance calculation, considering the following reactivity effects:

a. RCS boron concentration;

b. Control bank position;

c. RCS average temperature;

d. Fuel burnup based on gross thermal energy generation;

e. Xenon concentration;

f. Samarium concentration; and

g. Isothermal temperature coefficient (ITC).

Using the ITC accounts for Doppler reactivity in this calculation because the reactor is subcritical, and the fuel temperature will be changing at the same rate as the RCS.

The Frequency of 24 hours is based on the generally slow change in required boron concentration and on the low probability of an accident occurring without the required SDM.

REFERENCES

1. 10 CFR 50, Appendix B, Section XI.

2. 10 CFR 50.59.

3. Regulatory Guide 1.68, Revision 2, August, 1978.

4. ANSI/ANS-19.6.1-1985, December 13, 1985.

39

(continued)

ITS INSERT B3.1.8-1

SR 3.1.8.3

Verification that the THERMAL POWER is ≤ 5% RTP will ensure that the plant is not operating in a condition that could invalidate the safety analyses. Verification of THERMAL POWER at a frequency of 30 minutes during the performance of the PHYSICS TEST will ensure that the initial conditions of the safety analyses are not violated.

1 In the conversion of the HBRSEP current Technical Specifications (CTS) to the proposed plant specific Improved Technical Specifications (ITS), certain wording preferences or conventions are adopted which do not result in technical changes (either actual or interpretational). Editorial changes or clarifications which involve the insertion of plant specific terms, parameters, or descriptions are used to preserve consistency with the CTS and licensing basis.

2 HBRSEP was designed and licensed to the proposed Appendix A to 10 CFR 50, which was published in the Federal Register on July 11, 1967 (FR 32FR10213). Appendix A to 10 CFR 50 effective in 1971 and subsequently amended, is somewhat different from the proposed 1967 criteria. UFSAR section 3.1 includes an evaluation of HBRSEP with respect to the proposed 1967 criteria. The ISTS statement concerning the GDC criteria is modified in the ITS to reference the current licensing basis description in the UFSAR.

3 ISTS Specification 3.1.2 is not included as a separate specification in the ITS. Since the specific shutdown margin requirements for various plant conditions are relocated to the Core Operating Limits Report (COLR), there is no need for separate specifications for different MODES of Applicability. Consequently, shutdown margin requirements applicable to MODE 5 are included in ITS Specification 3.1.1. This eliminates the need for Specification 3.1.2. ISTS Specifications 3.1.9 and 3.1.11 are also not adopted in the ITS. Subsequent Specifications are renumbered accordingly.

4 The phrase, "... and the fuel and moderator temperatures are changed to the nominal hot zero power value ...," is added to clarify the assumptions used in determining the shutdown margin requirements during operation.

5 The term, "Control Rod System," is replaced with the phrase, "two independent reactivity control systems," to clarify that power maneuvers require both the control rod system and the Chemical and Volume Control System in concert (i.e., for boron concentration changes) to maintain the core flux shape within the axial and radial differential limitations.

6 The terms, "soluble boron system," and "boration system," are replaced with the plant specific terminology, "Chemical and Volume Control System," or "CVCS."

7 The phrase, "... Rod Cluster Control Assemblies and ...," is added because the worth of the control rod banks provide an essential portion of the shutdown margin.

8 The inserted phrase is relocated from ISTS page B 3.1-8, to address SDM in MODE 5.

repeated in the Bases.

35 Demand position indication is not calibrated. The counters are reset to zero when rods are fully inserted prior to startup.

36 The referenced analysis does not include explicit consideration of the effects on core peaking factors of rod position versus power level, and is not retained in the ITS.

37 The Bases for ITS 3.1.7 are revised to reflect changes made to the associated Specification and the current licensing basis approved in HBRSEP Unit No. 2 Amendment No. 48..

38 Not used.

39 HBRSEP is not committed to either Regulatory Guide 1.68 or ANSI/ANS-19.6.1.

40 The word, "more," is changed to the word, "both," because plant design includes two shutdown banks.

41 The boron exchange methodology is the method used at HBRSEP to perform integral and differential rod worth measurements. This method is used to determine the reactivity of individual rod banks, as well as the reactivity of the predicted "worst case" stuck rod.

42 The "average slope method" is used at HBRSEP for measuring isothermal temperature coefficient (ITC).

43 Not used.

44 The referenced reports are not applicable to HBRSEP.

45 Not used.

46 Not used.

47 Bases are modified for consistency with the scope and content of the associated Specification. This change is based on the need to perform the surveillance following plant evolutions that could cause disturbance of the instruments.

3.1 REACTIVITY CONTROL SYSTEMS

3.1.8 PHYSICS TESTS Exceptions – MODE 2

LCO 3.1.8 During the performance of PHYSICS TESTS, the requirements of

LCO 3.1.3, "Moderator Temperature Coefficient (MTC)";
LCO 3.1.4, "Rod Group Alignment Limits";
LCO 3.1.5, "Shutdown Bank Insertion Limits";
LCO 3.1.6, "Control Bank Insertion Limits"; and
LCO 3.4.2, "RCS Minimum Temperature for Criticality"

may be suspended, provided:

a. RCS lowest loop average temperature is ≥ 530°F;

b. SDM is within the limits provided in the COLR; and,

c. THERMAL POWER is ≤ 5% RTP

APPLICABILITY: MODE 2 during PHYSICS TESTS.

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. SDM not within limit.	A.1	Initiate boration to restore SDM to within limit.	15 minutes
	AND		
	A.2	Suspend PHYSICS TESTS exceptions.	1 hour
B. THERMAL POWER not within limit.	B.1	Open reactor trip breakers.	Immediately

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
C. RCS lowest loop average temperature not within limit.	C.1 Restore RCS lowest loop average temperature to within limit.	15 minutes
D. Required Action and associated Completion Time of Condition C not met.	D.1 Be in MODE 3.	15 minutes

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.1.8.1	Perform a CHANNEL OPERATIONAL TEST on power range and intermediate range channels per SR 3.3.1.7, SR 3.3.1.8, and Table 3.3.1-1.	Within 7 days prior to initiation of PHYSICS TESTS
SR 3.1.8.2	Verify the RCS lowest loop average temperature is ≥ 530°F.	30 minutes
SR 3.1.8.3	Verify THERMAL POWER is ≤ 5% RTP.	30 minutes
SR 3.1.8.4	Verify SDM is within the limits provided in the COLR.	24 hours

BASES

ACTIONS

A.1.1, A.1.2, A.2, B.1.1, B.1.2, and B.2 (continued)

Similarly, if the control banks are found to be out of sequence or in the wrong overlap configuration, they must be restored to meet the limits.

Operation beyond the LCO limits is allowed for a short time period in order to take conservative action because the simultaneous occurrence of either a LOCA, loss of flow accident, ejected rod accident, or other accident during this short time period, together with an inadequate power distribution or reactivity capability, has an acceptably low probability.

The allowed Completion Time of 2 hours for restoring the banks to within the insertion, sequence, and overlaps limits provides an acceptable time for evaluating and repairing minor problems without allowing the plant to remain in an unacceptable condition for an extended period of time.

C.1

If the Required Actions cannot be completed within the associated Completion Times, the plant must be brought to MODE 3 (utilizing normal operating procedures), where the LCO is not applicable. The allowed Completion Time of 6 hours is reasonable, based on operating experience, for reaching the required MODE from full power conditions in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.1.6.1

This Surveillance is required to ensure that the reactor does not achieve criticality with the control banks below their insertion limits.

The estimated critical position (ECP) depends upon a number of factors, one of which is xenon concentration. If the ECP was calculated long before criticality, xenon concentration could change to make the ECP substantially in error. Conversely, determining the ECP immediately before criticality could be an unnecessary burden. There are a number of unit parameters requiring operator attention at

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.1.6.1 (continued)

that point. Performing the ECP calculation within 4 hours prior to criticality avoids a large error from changes in xenon concentration, but allows the operator some flexibility to schedule the ECP calculation with other startup activities.

SR 3.1.6.2

With an OPERABLE bank insertion limit monitor, verification of the control bank insertion limits at a Frequency of 12 hours is sufficient to ensure OPERABILITY of the bank insertion limit monitor and to detect control banks that may be approaching the insertion limits since, normally, very little rod motion occurs in 12 hours. If the insertion limit monitor becomes inoperable, verification of the control bank position at a Frequency of 4 hours is sufficient to detect control banks that may be approaching the insertion limits.

SR 3.1.6.3

When control banks are maintained within their insertion limits as checked by SR 3.1.6.2 above, it is unlikely that their sequence and overlap will not be in accordance with requirements provided in the COLR. A Frequency of 12 hours is consistent with the insertion limit check above in SR 3.1.6.2.

REFERENCES

1. UFSAR, Sections 3.1.2.14, 3.1.2.27, 3.1.2.28, 3.1.2.29, 3.1.2.30, 3.1.2.31, and 3.1.2.32.

2. 10 CFR 50.46.

3. UFSAR, Chapter 15.

BASES

LCO (continued)

limits is permitted for the purpose of performing PHYSICS TESTS and poses no threat to fuel integrity, provided the SRs are met.

The requirements of LCO 3.1.3, LCO 3.1.4, LCO 3.1.5, LCO 3.1.6, and LCO 3.4.2 may be suspended during the performance of PHYSICS TESTS provided:

a. RCS lowest loop average temperature is ≥ 530 °F;

b. SDM is within the limits provided in the COLR; and

c. THERMAL POWER is ≤ 5% RTP.

APPLICABILITY

This LCO is applicable in MODE 2 when performing low power PHYSICS TESTS. The applicable PHYSICS TESTS are performed in MODE 2 at HZP.

ACTIONS

A.1 and A.2

If the SDM requirement is not met, boration must be initiated promptly. A Completion Time of 15 minutes is adequate for an operator to correctly align and start the required systems and components. The operator should begin boration with the best source available for the plant conditions. Boration will be continued until SDM is within limit.

Suspension of PHYSICS TESTS exceptions requires restoration of each of the applicable LCOs to within specification.

B.1

When THERMAL POWER is > 5% RTP, the only acceptable action is to open the reactor trip breakers (RTBs) to prevent operation of the reactor beyond its design limits. Immediately opening the RTBs will shut down the reactor and prevent operation of the reactor outside of its design limits.

(continued)

BASES

ACTIONS (continued)

C.1

When the RCS lowest T_{avg} is < 530°F, the appropriate action is to restore T_{avg} to within its specified limit. The allowed Completion Time of 15 minutes provides time for restoring T_{avg} to within limits without allowing the plant to remain in an unacceptable condition for an extended period of time. Operation with the reactor critical and with temperature below 530°F could violate the assumptions for accidents analyzed in the safety analyses.

D.1

If the Required Actions cannot be completed within the associated Completion Time, the plant must be brought to a MODE in which the requirement does not apply. To achieve this status, the plant must be brought to at least MODE 3 within an additional 15 minutes. The Completion Time of 15 additional minutes is reasonable, based on operating experience, for reaching MODE 3 in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.1.8.1

The power range and intermediate range neutron detectors must be verified to by OPERABLE in MODE 2 by LCO 3.3.1, "Reactor Protection System (RPS) Instrumentation." A CHANNEL OPERATIONAL TEST is performed on each power range and intermediate range channel within 7 days prior to initiation of the PHYSICS TESTS. This will ensure that the RPS is properly aligned to provide the required degree of core protection during the performance of PHYSICS TESTS. The 7 day time limit is sufficient to ensure that the instrumentation is OPERABLE before initiating PHYSICS TESTS.

SR 3.1.8.2

Verification that the RCS lowest loop T_{avg} is ≥ 530°F will ensure that the unit is not operating in a condition that could invalidate the safety analyses. Verification of the RCS temperature at a Frequency of 30 minutes during the

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.1.8.2 (continued)

performance of the PHYSICS TESTS will ensure that the initial conditions of the safety analyses are not violated.

SR 3.1.8.3

Verification that the THERMAL POWER is ≤ 5% RTP will ensure that the plant is not operating in a condition that could invalidate the safety analyses. Verification of THERMAL POWER at a frequency of 30 minutes during the performance of the PHYSICS TEST will ensure that the initial conditions of the safety analyses are not violated.

SR 3.1.8.4

The SDM is verified by performing a reactivity balance calculation, considering the following reactivity effects:

a. RCS boron concentration;

b. Control bank position;

c. RCS average temperature;

d. Fuel burnup based on gross thermal energy generation;

e. Xenon concentration;

f. Samarium concentration; and

g. Isothermal temperature coefficient (ITC).

Using the ITC accounts for Doppler reactivity in this calculation because the reactor is subcritical, and the fuel temperature will be changing at the same rate as the RCS.

The Frequency of 24 hours is based on the generally slow change in required boron concentration and on the low probability of an accident occurring without the required SDM.

(continued)

BASES (continued)

REFERENCES

1. 10 CFR 50, Appendix B, Section XI.
2. 10 CFR 50.59.

Industry/TSTF Standard Technical Specification Change Traveler

Add an LCO item and SR to Mode 2 Physics Tests Exceptions to verify that Thermal Power <= 5% RTP.

Classification: Not Classified

NUREGs Affected: ☐ 1430 ☑ 1431 ☐ 1432 ☐ 1433 ☐ 1434

Description:

Add an LCO requirement and SR to Mode 2 Physics Tests Exceptions 3.1.10 to verify that Thermal Power <= 5% RTP. Deleted references in the Bases to Physics Tests to tests performed in Mode 1 as this Tech Spec only applies to tests performed in Mode 2. Deleted the reference to Mode 2 in the Applicability.

Justification:

This LCO requirements and SR were added to verify that Thermal Power is within the defined power level for Mode 2 during performance of Physics Tests, since there is an action that addresses Thermal Power not within limit and no corresponding LCO or surveillance.
The Bases references to Physics Tests performed in Mode 1 were unnecessary as this specification refers only to tests performed in Mode 2.
The explicit reference to Mode 2 in the Applicability is unnecessary as the LCO limits the use of the Test Exception to power levels less than 5% (the upper limit of Mode 2).

Affected Technical Specifications

LCO 3.1.10	Physics Test Exceptions - Mode 2	
LCO 3.1.10 Bases	Physics Test Exceptions - Mode 2	
SR 3.1.10.3	Physics Test Exceptions - Mode 2	Change Description: Renumber to 3.1.10.4
SR 3.1.10.3	Physics Test Exceptions - Mode 2	Change Description: Inserted
SR 3.1.10.3 Bases	Physics Test Exceptions - Mode 2	Change Description: Renumber to 3.1.10.4
SR 3.1.10.3 Bases	Physics Test Exceptions - Mode 2	Change Description: Inserted

WOG Review Information

WOG-4.6

Originating Plant: Date Provided to OG: 11-Mar-95 Needed By:

Owners Group History:

WOG-04, C.6

Owners Group Resolution: Approved Date: 11-Aug-95

TSTF Review Information

TSTF Received Date: 05-Sep-95 Date Distributed to OGs for Review: 05-Sep-95

OG Review Completed: ☑ BWOG ☑ WOG ☑ CEOG ☑ BWROG

TSTF History:

TSTF Resolution: Approved Date: 05-Sep-95 **TSTF- 14**

NRC Review Information

NRC Received Date: 03-Oct-95 NRC Reviewer: R. Tjader Reviewer Phone #:

Reviewer Comments:

10/4/95 - R. Tjader approved change, pkg to TSB mgmt.
11/17/95 - C. Grimes approved change.
1/20/96 changes processed. Completion of pkg. waiting on completion of TSTF-12.
6/12/96 - -Reviewer completed review. Reviewer's comment: Change is a matter of preference and editorial. Adding "c. Thermal Power <= 5% RTP," to LCO and Bases adds clarity and should be approved. Removing "Mode 2" from Applicability and Bases does not enhance clarity. Except for a few refueling LCOs, all other LCOs refer to a Mode. Prefer "Mode 2" be retained in the Applicability section.
Note: TSTF-14, R. 2 was submitted by TSB reviewer on 6/12/96 for his review.
6/11/96 - C. Grimes comment: TSTF-14, R. 1 was approved.
9/18/96 - NRC requested revision to retain Mode 2 in the applicability. TSTF agreed and will prepare revision.
10/15/96 - New revision forwarded to the TSTF for review.
3/13/97 - NRC approves TSTF-14, Rev. 3.
3/18/97 - NRC informed by TSTF that editorial change to TSTF-14, Rev. 3 was needed. Rev. 4 forthcoming.

Final Resolution: NRC Requests Changes: TSTF Will Revise Final Resolution Date:

Revision History

TSTF Revision 1 Revision Date: 08-Jan-96 Proposed by: TSTF

Revision Description:
Remarked the pages to use TSTF number instead of OG number.
The Tech Spec markup contains other changes not discussed in the Discussion or Justification. The TSTF package was WOG-4, C.6 only, but changes WOG-4, C.1 and C.4 were included in the TSTF package. These were removed.

Distributed to TSTF: Resolution: Approved Date: 08-Jan-96 Rev to NRC: 1/8/96

TSTF Revision 2 Revision Date: 15-Jan-96 Proposed by: TSTF

Revision Description:
Added a LCO requirement in addition to the surveillance.

Distributed to TSTF: Resolution: Approved Date: 28-May-96 Rev to NRC: 5/28/96

TSTF Revision 3 Revision Date: 18-Sep-96 Proposed by:

Revision Description:
Reviewer completed review. Reviewer's comment: Change is a matter of preference and editorial. Adding "c. Thermal Power <= 5% RTP," to LCO and Bases adds clarity and should be approved. Removing "Mode 2" from Applicability and Bases does not enhance clarity. Except for a few refueling LCOs, all other LCOs refer to a Mode. Prefer "Mode 2" be retained in the Applicability section.

Distributed to TSTF: 11/20/96 Resolution: Approved Date: 19-Dec-96 Rev to NRC: 1/17/97

TSTF Revision 4 Revision Date: 23-Mar-97 Proposed by: TSTF

Revision Description:
Insert 1 to the Bases contained brackets around the Surveillance Frequency even though the Frequency was not backeted in the SR. This revision corrects this by eliminating the brackets in the insert.

Distributed to TSTF: 3/23/97 Resolution: Approved Date: 23-Mar-97 Rev to NRC:

Incorporation Into the NUREGs

File to BBS/LAN Date:

File to TSTF Date:

File Rev Incoporated:

File Rev Incorporated Date

3.1 REACTIVITY CONTROL SYSTEMS

3.1.10 PHYSICS TESTS Exceptions—MODE 2

LCO 3.1.10 During the performance of PHYSICS TESTS, the requirements of

LCO 3.1.4, "Moderator Temperature Coefficient (MTC)";
LCO 3.1.5, "Rod Group Alignment Limits";
LCO 3.1.6, "Shutdown Bank Insertion Limits";
LCO 3.1.7, "Control Bank Insertion Limits"; and
LCO 3.4.2, "RCS Minimum Temperature for Criticality"

may be suspended, provided:

a. RCS lowest loop average temperature is ≥ [531]°F; ~~and~~

b. SDM is ≥ [1.6]% $\Delta k/k$]; and

c. THERMAL POWER is ≤ 5% RTP.

APPLICABILITY: MODE 2 during PHYSICS TESTS.

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. SDM not within limit.	A.1	Initiate boration to restore SDM to within limit.	15 minutes
	AND		
	A.2	Suspend PHYSICS TESTS exceptions.	1 hour
B. THERMAL POWER not within limit.	B.1	Open reactor trip breakers.	Immediately

(continued)

TSTF-14, Rev. 4

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
C. RCS lowest loop average temperature not within limit.	C.1 Restore RCS lowest loop average temperature to within limit.	15 minutes
D. Required Action and associated Completion Time of Condition C not met.	D.1 Be in MODE 3.	15 minutes

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.1.10.1	Perform a CHANNEL OPERATIONAL TEST on power range and intermediate range channels per [SR 3.3.1.7, SR 3.3.1.8, and Table 3.3.1-1].	Within 12 hours prior to initiation of PHYSICS TESTS
SR 3.1.10.2	Verify the RCS lowest loop average temperature is ≥ [531]°F.	30 minutes
SR 3.1.10.~~2~~ 4	Verify SDM is ≥ 1.6% Δk/k.	24 hours
SR 3.1.10.3	Verify THERMAL POWER is ≤ 5% RTP.	30 minutes

TSTF-14, Rev. 4

BASES

LCO (continued)

limits is permitted for the purpose of performing PHYSICS TESTS and poses no threat to fuel integrity, provided the SRs are met.

The requirements of LCO 3.1.4, LCO 3.1.5, LCO 3.1.6, LCO 3.1.7, and LCO 3.4.2 may be suspended during the performance of PHYSICS TESTS provided:

a. RCS lowest loop average temperature is ≥ [531] °F; ~~and~~

b. SDM is ≥ [1.6]% Δk/k; and

c. THERMAL POWER is ≤ 5% RTP.

APPLICABILITY

This LCO is applicable in MODE 2 when performing low power PHYSICS TESTS. The applicable PHYSICS TESTS are performed in MODE 2 at HZP. Other PHYSICS TESTS are performed in MODE 1 and are addressed in LCO 3.1.9, "PHYSICS TESTS Exceptions—MODE 1."

ACTIONS

A.1 and A.2

If the SDM requirement is not met, boration must be initiated promptly. A Completion Time of 15 minutes is adequate for an operator to correctly align and start the required systems and components. The operator should begin boration with the best source available for the plant conditions. Boration will be continued until SDM is within limit.

Suspension of PHYSICS TESTS exceptions requires restoration of each of the applicable LCOs to within specification.

B.1

When THERMAL POWER is > 5% RTP, the only acceptable action is to open the reactor trip breakers (RTBs) to prevent operation of the reactor beyond its design limits. Immediately opening the RTBs will shut down the reactor and prevent operation of the reactor outside of its design limits.

(continued)

TSTF-14, Rev. 4

BASES

SURVEILLANCE REQUIREMENTS

SR 3.1.10.2 (continued)

performance of the PHYSICS TESTS will ensure that the initial conditions of the safety analyses are not violated.

INSERT 1

SR 3.1.10.~~3~~4

The SDM is verified by performing a reactivity balance calculation, considering the following reactivity effects:

a. RCS boron concentration;

b. Control bank position;

c. RCS average temperature;

d. Fuel burnup based on gross thermal energy generation;

e. Xenon concentration;

f. Samarium concentration; and

g. Isothermal temperature coefficient (ITC).

Using the ITC accounts for Doppler reactivity in this calculation because the reactor is subcritical, and the fuel temperature will be changing at the same rate as the RCS.

The Frequency of 24 hours is based on the generally slow change in required boron concentration and on the low probability of an accident occurring without the required SDM.

REFERENCES

1. 10 CFR 50, Appendix B, Section XI.

2. 10 CFR 50.59.

3. Regulatory Guide 1.68, Revision 2, August, 1978.

4. ANSI/ANS-19.6.1-1985, December 13, 1985.

(continued)

TSTF-14, Rev 4

INSERT 1

SR 3.1.~~8~~10.3

Verification that the THERMAL POWER is ≤ 5% RTP will ensure that the plant is not operating in a condition that could invalidate the safety analyses. Verification of the THERMAL POWER at a Frequency of 30 minutes during the performance of the PHYSICS TESTS will ensure that the initial conditions of the safety analyses are not violated.

SUPPLEMENT 5
CONVERSION PACKAGE SECTION 3.2
PAGE INSERTION INSTRUCTIONS

Remove and insert the following pages into Enclosure 10 to Serial: RNP-RA/96-0141.

	Remove Page	Insert Page
a.	Part 1, "Markup of Current Technical Specifications (CTS)"	
	3.10-2, 3.10-3 (ITS 3.2.1) 3.10-3 (ITS 3.2.2)	3.10-2, 3.10-3 (ITS 3.2.1) 3.10-3 (ITS 3.2.2)
	3.10-5a, 3.10-7a	3.10-5a,3.10-7a
b.	Part 2, "Discussion of Changes (DOCs) for CTS Markup"	
	1 through 24	1 through 28
c.	Part 3, " No Significant Hazards Consideration (NSHC), And Basis for Categorical Exclusion from 10 CFR 51.22	
	NA	
d.	Part 4, "Markup of NUREG-4131, Revision 1, Standard Technical Specifications-Westinghouse Plants, (ISTS)"	
	NA	
e.	Part 5, "Justification of Differences (JFDs) to ISTS"	
	1 through 3	1 through 3
f.	Part 6, "Markup of ISTS Bases"	
	B 3.2-14	B 3.2-14
	Insert B. 3.2.1-1 (no page number)	B 3.2-14a
	Insert B. 3.2.1-1 (no page number)	B 3.2-14b
	B 3.2-18	B 3.2-18
	Insert IB 3. 2. 3-1 (no page number)	B 3.2-31a
g.	Part 7, "Justification for Differences (JFDs) to ISTS Bases"	
	1	1
h.	Part 8, "Proposed HBRSEP, Unit No. 2 ITS"	
	NA	
i.	Part 9. "Proposed Bases to HBRSEP, Unit No. 2 ITS Bases"	
	B 3.2-4, B 3.2-5, B 3.2-6, B 3.2-7, B 3.2-8	B 3.2-4, B 3.2-5, B 3.2-6, B 3.2-7, B 3.2-8
	B 3.2-20	B 3.2-20
j.	Part 10. "ISTS Generic Changes"	
	NA	

ITS

A1

3.10.1.5 Except for physics tests, if a full length control rod is withdrawn as follows:

- at positions ≥ 200 steps and is > 15 inches out of alignment with its bank position, or

- at positions < 200 steps and is > 7.5 inches out of alignment with the average of its bank position

then within two hours, perform the following:

a. Correct the situation, or

b. Determine by measurement the hot channel factors and apply Specification 3.10.2.1, or

c. Limit power to 70 percent of rated power

3.10.1.6 Insertion limits do not apply during physics tests or during period exercise of individual rods. However, the shutdown margin indicated in Figure 3.10-2 must be maintained, except during the low power physics test to measure control rod worth and shutdown margin. For this test, the reactor may be critical with all but one full length control rod inserted.

See 3.1.4 3.1.5 3.1.6 3.1.8

L1

3.10.2 Power Distribution Limits

MODE 1

[applicability]

3.10.2.1 At all times except during low power physics tests, the hot channel factors, $F_Q(Z)$ and $F_{\Delta H}$, defined in the Basis, must meet the following limits:

M1

[LCO 3.2.1]

$F_Q(Z) \leq (F_Q^{RTP}/P) \times K(Z)$ for $P > 0.5$

$F_Q(Z) < (F_Q^{RTP}/0.5) \times K(Z)$ for $P \leq 0.5$

limits specified in the COLR.

LA1

A23

as approximated by $F_Q^V(Z)$

LA1

$F_{\Delta H} < F_{\Delta H}^{RTP} (1 + PF_{\Delta H} (1-P))$

See 3.2.2

LA1

See 3.2.2

Add RA A.2.2
RA A.2.4
RA B.1

M2

Supplement 5

ITS

A1

where P is the fraction of rated power (2300 Mwt) at which the core is operating. $F_Q(Z)$ is the measured $F_Q^N(Z)$ multiplied by the measurement uncertainty factor $F_Q^N = 1.05$ and the engineering factor $F_Q^E = 1.03$. $F_{\Delta H}$ is the measured $F_{\Delta H}^N$ multiplied by a 1.04 measurement uncertainty factor. K(Z) is the normalized $F_Q(Z)$ as a function of core height specified in the CORE OPERATING LIMITS REPORT (COLR). F_Q^{RTP} is the F_Q limit at RATED THERMAL POWER (RTP). $F_{\Delta H}^{RTP}$ is the $F_{\Delta H}$ limit at RATED THERMAL POWER. $PF_{\Delta H}$ is the Power Factor Multiplier for $F_{\Delta H}^{RTP}$. F_Q^{RTP}, $F_{\Delta H}^{RTP}$, and $PF_{\Delta H}$ are specified in the COLR.

LA1 See 3.2.2 LA1 See 3.2.2 LA1

3.10.2.1.1 Following ~~initial loading~~ refueling and prior to exceeding 75% RTP, or ~~upon~~ achieving equilibrium conditions after exceeding by 10% or more of rated power, the power $F_Q(Z)$ was last determined, and at least once per ~~effective full power month~~ 31 EFPDs, power distribution maps using the movable detector system shall be made to confirm that the ~~hot channel factor~~ $F_Q(Z)$ limits of Specification 3.10.2.1 are satisfied and to establish the target axial flux difference as a function of power level (called the target flux difference).*

[SR 3.2.1.1]

within 12 hours of

M3 A3

If either measured hot channel factor exceeds the specified limit, the reactor power shall be reduced so as not to exceed a fraction equal to the ratio of the $F_Q(Z)$ or $F_{\Delta H}$ limit to the measured value, whichever is less, and the high neutron flux trip setpoint shall be reduced by the same ratio.

See 3.2.3 A4

If subsequent incore mapping cannot, within a ~~24~~ 72 hour period, demonstrate that the hot channel factors are met, the overpower ΔT and overtemperature ΔT trip setpoints shall be similarly reduced.

[RA A.2.3]

See 3.2.2 L2 A2 M28

During power escalation at the beginning of each cycle, the design target may be used until a power level for extended operation has been achieved.

See 3.2.3

Supplement 5

Specification 3.2.2

ITS

where P is the fraction of rated power (2300 Mwt) at which the core is operating. $F_Q(Z)$ is the measured $F_Q^N(Z)$ multiplied by the measurement uncertainty factor $F_Q^N = 1.05$ and the engineering factor $F_Q^E = 1.03$. $F_{\Delta H}$ is the measured $F_{\Delta H}^N$ multiplied by a 1.04 measurement uncertainty factor. K(Z) is the normalized $F_Q(Z)$ as a function of core height specified in the CORE OPERATING LIMITS REPORT (COLR). F_Q^{RTP} is the F_Q limit at RATED THERMAL POWER (RTP). $F_{\Delta H}^{RTP}$ is the $F_{\Delta H}$ limit at RATED THERMAL POWER. $PF_{\Delta H}$ is the Power Factor Multiplier for $F_{\Delta H}^{RTP}$. F_Q^{RTP}, $F_{\Delta H}^{RTP}$, and $PF_{\Delta H}$ are specified in the COLR.

LA2 A1 See 3.2.1 LA2 See 3.2.1 LA2 See 3.2.1 A2

3.10.2.1.1 31 EFPDs refueling prior to exceeding 75% RTP

Following initial loading, or upon achieving equilibrium conditions after exceeding by 10% or more of rated power, the power $F_Q(Z)$ was last determined, and at least once per effective full power month, power distribution maps using the movable detector system, shall be made to confirm that the hot channel factor limits of Specification 3.10.2.1 are satisfied and to establish the target axial flux difference as a function of power level (called the target flux difference).*

[SR 3.2.2.] LA5 M8 $F_{\Delta H}$ L4 A3

[RA A.1.1] [RA A.1.2.1] [RA A.1.2.2] If either measured hot channel factor ($F_{\Delta H}$) exceeds the specified limit, the reactor power shall be reduced so as not to exceed a fraction equal to the ratio of the $F_Q(Z)$ or $F_{\Delta H}$ limit to the measured value, whichever is less, and the high neutron flux trip setpoint shall be reduced by the same ratio (to < 55% RTP).

See 3.2.3 A3 by 50% M10 M9 See 3.2.1 M10

If subsequent incore mapping cannot, within a 24-hour period, demonstrate that the hot channel factors are met, the overpower ΔT and overtemperature ΔT trip setpoints shall be similarly reduced.

then restore $F_{\Delta H}^N$ to within limits within 4 hours OR

Add Note to Condition A

	RA	A.2
	RA	A.3
Note to	RA	A.3
	RA	B.1

L5 M9 M11

During power escalation at the beginning of each cycle, the design target may be used until a power level for extended operation has been achieved.

See 3.2.3

Supplements

Amendment No. 87, 118, 119, 141, 151

ITS

Specification 3.2.2

A1

3.10.2.2.3 With successive measurements indicating the enthalpy rise hot channel factor, $F^N_{\Delta H}$, to be increasing with exposure, the total peaking factor, $F_Q(Z)$, shall be further increased by two percent over that specified in Specifications 3.10.2.2, 3.10.2.2.1, and

[SR 3.2.2.1 NOTE]

$F^V_Q(z)$

A2

Supplement 5

Amendment No. 141

For each OPERABLE ex core channel — A24 — A1

b. at power levels less than 90 percent of rated power or 0.9 x APL (whichever is less) provided the indicated axial flux difference does not exceed the limits specified in the COLR. — L7

3.10.2.10 Alarms shall normally be used to indicate non-conformance with the flux difference requirement of 3.10.2.6 or the flux difference-time requirement of 3.10.2.7.a. If the alarms are temporarily out of service, the axial flux difference shall be logged and conformance with the limits assessed, every hour for the first 24 hours, and half-hourly thereafter. — A15

[SR 3.2.3.2 Frequency Note] [SR 3.2.3.2]

Insert 3.2.3.4

3.10.2.11 The axial flux difference target band about the target axial flux difference shall be determined in conjunction with the measurement of $F_Q(Z)$ as specified in 3.10.2.1.1. The allowable values of the target band are specified in the COLR. Redefinition of the target band from more restrictive to less restrictive ranges between determinations of the target axial flux difference is allowed when appropriate redefinitions of APL are made. Redefinition of the target band from less restrictive to more restrictive ranges is allowed only in conjunction with the determination of a new target axial flux difference. — M20 — L8 — A16

[LCO 3.2.3.a]

3.10.3 Quadrant Power Tilt Limits

3.10.3.1 Except for physics tests and during power increases below 50 percent of rated power, whenever the indicated quadrant power tilt ratio exceeds 1.02, the tilt condition shall be eliminated within two hours or the following actions shall be taken:

a. Restrict core power level and reset the power range high flux setpoint to be less than two percent of rated values for every percent of indicated power tilt ratio exceeding 1.0, and — LA3

See 3.2.4

Add Note to SR 3.2.3.2 — A15

Supplement 5

ADMINISTRATIVE CHANGES

A1 In the conversion of the H. B. Robinson Steam Electric Plant (HBRSEP), Unit 2 Current Technical Specifications (CTS) to the proposed plant specific Improved Technical Specifications (ITS) certain wording preferences or conventions are adopted which do not result in technical changes (either actual or interpretational). Editorial changes, reformatting, and revised numbering are adopted to make ITS consistent with the conventions in the Standard Technical Specifications, Westinghouse Plants, NUREG-1431, Revision 1 (i.e., Improved Standard Technical Specifications (ISTS)). These changes are administrative, and have no adverse impact on safety.

A2 CTS Specification 3.10.2.1.1 includes the term, "effective full power month," which is changed to 31 Effective Full Power Days (EFPDs) in the ITS to be consistent with NUREG-1431. Both the CTS and ITS terms are equivalent. This change is administrative, and has no adverse impact on safety.

A3 CTS Specification 3.10.2.1.1 applies to both hot channel factors $F_Q(Z)$ and $F_{\Delta H}$. CTS Specification 3.10.2.1.1 is retained in ITS as two Limiting Conditions for Operations (LCOs), which are, ITS Specification 3.2.1, "Heat Flux Hot Channel Factor ($F_Q(Z)$)" and ITS Specification 3.2.2, "Nuclear Enthalpy Rise Hot Channel Factor ($F^N_{\Delta H}$)." As such the term "hot channel factors" and $F_Q(Z)$ in CTS Specification 3.10.2.1.1 is retained as $F^V_Q(Z)$ in ITS Specification 3.2.1 and as $F^N_{\Delta H}$ in ITS Specification 3.2.2. This change is administrative, and has no adverse impact on safety.

A4 CTS Specification 3.10.2.1.1, second paragraph, which contains a required action for the condition where the measured F_Q exceeds the specified limits, is not retained in the ITS. This required action contains a method for reducing power that is less restrictive than CTS Specification 3.10.2.2.1.b, which provides an alternative method that is more conservative than CTS Specification 3.10.2.1.1, second paragraph. CTS Specification 3.10.2.2.1.b requires that the reactor power be reduced by 1% for every 1% that $F_Q(Z)$ exceeds its limits rather than limiting reactor power to the fraction expressed in CTS Specification 3.10.2.1.1 as $F_Q(Z)_{limit}/F^V_Q(Z)_{actual}$. The CTS Specification 3.10.2.2.1.b method of determining the reduced power limitation becomes more conservative than CTS Specification 3.10.2.1.1 as the deviation between the F_Q limits and the measured F_Q increases. CTS Specification 3.10.2.2.1.b is also consistent with NUREG-1431, and is adopted in the ITS. Therefore, this change is administrative, and has no adverse impact on safety.

A5 CTS Specification 3.10.2.2, first sentence, is redundant to, and refers to, CTS Specification 3.10.2.1, and is not retained in ITS. This change is administrative, and has no adverse impact on safety.

ADMINISTRATIVE CHANGES
(continued)

A6 CTS Specifications 3.10.2.2.1.b requires that reactor power be reduced by the expression:

[[max. over Z of ($F_Q(Z)$ x V(Z)) / (($F_Q^{RTP}(Z)/P$) x K(Z))] -1] X 100%

when $F_Q^V(Z)$ exceeds the limit. In the bases to ITS, the expression $F_Q(Z)$ x V(Z) is defined as $F_Q^V(Z)$. In the CORE OPERATING LIMITS REPORT (COLR), the limits for $F_Q^V(Z)$ are defined as ($F_Q^{RTP}(Z)/P$) x K(Z). The above expression then reduces to a mathematical equivalent to converting the fraction that $F_Q^V(Z)$ exceeds the limit into a percent RATED THERMAL POWER (RTP). This change is administrative, and has no adverse impact on safety.

A7 CTS Specification 3.10.2.2.1.b, which requires that reactor power be reduced when the measured F_Q exceeds the F_Q limits is retained in ITS Specification 3.2.1 as Required Action A.1.2. CTS Specification 3.10.2.2.1.b also states that the action applies to the ". . . middle axial 80% of the core." This requirement is not retained in ITS, because the axial offset methodology only applies to the middle 80% of the core, as described in the Bases to ITS Specification 3.2.1. This change is therefore administrative, and has no adverse impact on safety.

A8 The CTS Bases are not retained in the ITS, but are replaced in their entirety. The ITS includes significantly expanded and improved Bases. The Bases do not define or impose any specific requirements but serve to explain, clarify and document the reasons (i.e., Bases) for the associated Specification. The Bases are not part of the Technical Specifications required by 10 CFR 50.36. This change is administrative, and has no adverse impact on safety.

A9 CTS Specification 3.10.2.6, which contains the Required Action to return the AXIAL FLUX DIFFERENCE (AFD) to the target band immediately if the AFD is outside of the target band, is modified in the ITS 3.2.3 Required Action A.1 to require a Completion Time of 15 minutes to restore AFD to within the target band. ITS Section 1.3, "Completion Times," states that if "Immediately" is used as a Completion Time, the Required Action should be pursued without delay and in a controlled manner. The Completion time of 15 minutes for accomplishing ITS 3.2.3 Required Action A.1 is a reasonable interpretation of the CTS Completion Time of "immediately." The completion time is reasonable because xenon distributions change little in this relatively short time. Therefore, this change to CTS Specification 3.10.2.6 is administrative, and has no adverse impact on safety.

ADMINISTRATIVE CHANGES
(continued)

A10 CTS Specification 3.10.2.7.a, which contains the Required Action to immediately reduce reactor power to < 50% rated power if cumulative time exceeds one (1) hour if the AFD is outside of the target band, is modified in the ITS 3.2.3 Required Action C.1 to require a Completion Time of 30 minutes to reduce THERMAL POWER to < 50% RTP. ITS Section 1.3, "Completion Times," states that if "Immediately" is used as a Completion Time, the Required Action should be pursued without delay and in a controlled manner. The CTS has no interpretation of "immediately" equivalent to the ITS Section 1.3. The ITS Completion time of 30 minutes for accomplishing Required Action C.1 is a reasonable interpretation of the CTS Completion Time of "immediately," when considering the operating experience associated with reduction in THERMAL POWER from 100% RTP to less than 50% RTP. Therefore, this change to CTS Specification 3.10.2.7.a is administrative, and has no adverse impact on safety.

A11 CTS Specifications 3.10.2.7.b contains requirements that restrict an increase in reactor power above rated power levels in which the particular specifications for AFD apply unless the specifications are met. This requirement duplicates that of CTS 3.10.2.5 which is retained as ITS LCO 3.2.3.a and therefore is not retained in ITS. This change is administrative, and has no adverse impact on safety.

A12 CTS Specification 3.10.2.8.b, which requires the accumulation of penalty deviation time for AFD outside of the target band at power levels less than or equal to 50% reactor power, provides that penalty deviation time be accumulated at one half of the rate that penalty deviation time is accumulated when the reactor is greater than 50% rated power. This requirement is retained in ITS Note to LCO 3.2.3.c, but is completely rewritten for clarity and states that penalty deviation time ". . . shall be accumulated on the basis of a 0.5 minute penalty deviation for each 1 minute of power operation with AFD outside of the target band." CTS Specification 3.10.2.8.b states that operation above 50% reactor power is allowed when ". . . the indicated axial flux difference not being outside its target band for more than two hours (cumulative) out of the preceding 24-hour period. One-half of the time the indicated AFD is out of its target band up to 50 percent of rated power is to be counted as contributing to the one-hour cumulative maximum the flux difference may deviate from its target band at a power level less than or equal to 90 percent of rated power. . ." The CTS statement is identical in meaning to the ITS Note to LCO 3.2.3.c. When applying the Note to LCO 3.2.3.c to the stated CTS requirement, cumulative penalty hours will add up to a total of one (1) hour for each two (2) hours below 50% RTP. Therefore, this change is administrative, and has no adverse impact on safety.

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A13 CTS Specification 3.10.2.7.a, which allows the indicated AFD to deviate from its target band for a maximum of one hour (cumulative) in any 24-hour period, is modified in the ITS to add a Note to LCO 3.2.3.b that clarifies the requirement for cumulative penalty time. This change clarifies that cumulative penalty time is accumulated in increments of one minute. Since this change provides clarification only and does not add requirements, this change is administrative, and has no adverse impact on safety.

A14 CTS Specification 3.10.2.7.a, which requires that power be reduced to no greater than 50% rated power, when cumulative penalty time exceeds one hour, is revised in ITS 3.2.3 Required Action C.1 to include a Note to Condition C that Required Action C.1 must be completed whenever Condition C is entered. The CTS does not include ITS LCO 3.0.2, which permits exiting from a Required Action if the LCO is met or no longer applicable prior to the expiration of the specified Completion Time. Hence, the CTS also requires that the required action be completed whenever the Specification requirement is entered. Since the addition of this note only provides clarification with regard to ITS LCO 3.0.2, this change adds no requirements, is administrative, and has no adverse impact on safety.

A15 CTS Specification 3.10.2.10, which requires that alarms shall be normally used to indicate non-conformance with AFD requirements, and if the AFD monitor alarms are out of service, the AFD be logged. This surveillance is retained in ITS and is modified by a Note in ITS Surveillance Requirement (SR) 3.2.3.2 to clarify that logged values should be assumed to exist during the preceding time interval, and by a Note to the SR 3.2.3.2 Frequency that the SR is only required to be performed if the AFD monitor alarm is inoperable. The Note to SR 3.2.3.2 clarifies that the LCO is satisfied for time periods that AFD alarms are operable and not in alarm. This Note is equivalent in meaning to the CTS requirement that alarms ". . . shall normally be used. . ." The Note to SR 3.2.3.2 Frequency clarifies that the LCO is satisfied for the same time periods that AFD alarms are operable and not in alarm, without performance of the SR. This Note is equivalent in meaning to the CTS requirement to log the AFD when the alarms are out of service. Because this change adds clarification and does not add or relax requirements, this change is administrative, and has no adverse impact on safety.

A16 CTS Specification 3.10.2.11, first sentence, which requires that the AFD be determined in conjunction with the measurement of F_Q is not separately retained in the ITS. The first sentence duplicates the requirements of CTS Specification 3.10.2.3 and CTS Specification 3.10.2.3 is retained in the ITS as SR 3.2.3.3. Therefore, the deletion

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of this duplicate requirement is administrative, and has no adverse impact on safety.

A17 CTS Specification 3.10.2.2.2, which defines the Allowable Power Level (APL) as a function of $F_Q(Z)$ and $F_Q^V(Z)$, is retained in the ITS as a note to LCO 3.2.3. APL reduces the allowable AFD target as a function of RTP, and therefore is required to ensure that the deviation from target flux difference is within the acceptable target band. The expression $[F_Q(Z) \times V(Z)]$ is simplified to the equivalent variable expression $F_Q^V(Z)$, which is also defined in the ITS bases. Because this change does not add or reduce requirements, this change is administrative, and has no adverse impact on safety.

A18 CTS Specification 3.10.2.1.1, which requires that power distribution maps using the moveable detector be made to confirm the target AFD is retained in the ITS and restated to "Determine by measurement the target flux difference of each OPERABLE excore channel." The ITS requirement is identical in meaning to the CTS Specification, with the exception that the CTS is silent with respect to whether the AFD is required or not for an inoperable excore channel. Since the target flux difference cannot be determined for inoperable excore channels, this change is administrative, and has no adverse impact on safety.

A19 CTS Specification 3.10.3.1.a, which requires that core power and power range high flux setpoint be reduced when the QUADRANT POWER TILT RATIO (QPTR) is in excess of the limit, is retained in the ITS with the term "rated values" clarified to be "rated thermal power values" to clarify that it is a reduction in rated thermal power that is required when the QPTR limit is exceeded. This is an administrative change, and has no adverse impact on safety.

A20 CTS Specification 1.8, which states that three inservice excore detectors "are" used to determine quadrant power tilt when one is out of service, is revised in ITS SR 3.2.4.1, Note 1, to state that the three remaining power range channels "can be" used for calculating the QPTR. The CTS contains no specific SR Applicability section, and consequently, CTS requirements for surveillance of QPTR when one excore detector is inoperable uses the descriptive verb "are" when describing the surveillance requirement. The ITS includes SR 3.0.1 which, in combination with SR 3.2.4.1 and Note 1, and the ITS definition of QPTR, prohibits determination of QPTR utilizing excore detectors unless three or four excore detectors are OPERABLE. Therefore, the change from "are" to "can be" is administrative, and has no adverse impact on safety.

A21 The footnote to CTS Specifications 3.10.2.3 and 3.10.2.6, providing a reference for Allowable Power Level (APL) is not retained in ITS. ITS

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LCO 3.2.3.b, Note 2 adequately defines APL for LCO 3.2.3. This is an administrative change, and has no adverse impact on safety.

A22 CTS Specification 3.10.2.2.2 is revised to add descriptive information for Allowable Power Level (APL) and is retained in ITS LCO 3.2.3.b, Note 2. This is an administrative change, and has no adverse impact on safety.

A23 CTS Specification 3.30.2.1 which identifies the hot channel factor $F_q(Z)$ is modified in ITS LCO 3.2.1 to state that $F_q(Z)$ is approximated by $F_Q^V(Z)$. As stated in the Bases to LCO 3.2.1, $F_Q^V(Z)$ is a function of the measured hot channel factor times a constant, and V(Z), which is a cycle specific function which varies with core height typically from approximately 1.05 to approximately 1.11. Since V(Z) is always ≥ 1.0, $F_Q^V(Z)$ is always $\leq F_q(Z)$ and maintaining $F_Q^V(Z)$ within limits assures that $F_q(Z)$ is always within limits. The function V(Z) is cycle specific and is contained in the COLR. Since the approximate relationship between $F_q(Z)$ and $F_Q^V(Z)$ is also specified in accordance with the COLR and is in accordance with the PDC-3 Axial Offset Control Methodology, this change is considered administrative, and has no adverse impact on safety.

A24 CTS specification 3.10.2.10 requires that if the AFD alarms are temporarily out of service, the AFD shall be logged and conformance with the limits assessed. ITS SR 3.2.3.2 clarifies that the AFD be logged for each OPERABLE excore channel. Since the ITS definition of AFD applies to the difference in flux signals between the top and bottom halves of a two section excore neutron detector, and this definition is consistent with plant interpretation of AFD as it is applied in the PDC-3 Axial Offset Control Methodology, this change to the CTS is administrative, and has no impact on safety.

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M1 CTS Specification 3.10.2.1, which excludes applicability for maintaining F_Q within limits during physics testing, is not retained in ITS. ISTS Specification 3.2.1, "Heat Flux Hot Channel Factor ($F_Q(Z)$) (F_Q Methodology)," does not allow a physics test exception to F_Q limits. The F_Q limits are applicable at all times when the reactor is at power. There are no physics tests performed that require an exception to the F_Q limits. Additionally, since limits on $F_Q(Z)$ ensure that the value of the initial total peaking factor assumed in the accident analyses remains valid, a physics test exception to the F_Q limits is inappropriate. Since this change imposes new requirements, it is more restrictive and has no adverse impact on safety.

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M2 The CTS is revised to adopt the Required Actions A.1, A.4 and B.1 from ISTS Specification 3.2.1, "Heat Flux Hot Channel Factor ($F_Q(Z)$) (F_Q Methodology)," as ITS 3.2.1 Required Actions A.2.1, A.2.4 and B.1 in the ITS to ensure that appropriate additional actions are taken when ($F_Q(Z)$ is not within the required limits. Reducing THERMAL POWER by ≥ 1% RTP for each 1% by which $F_Q^V(Z)$ exceeds its limit, in accordance with Required Action A.2.1, maintains an acceptable absolute power density. The Completion Time of 30 minutes provides an acceptable time to reduce power in an orderly manner and without allowing the plant to remain in an unacceptable condition for an extended period of time. Verification that $F_Q^V(Z)$ has been restored to within its limit, by performing SR 3.2.1.1 prior to increasing THERMAL POWER above the limit imposed by Required Action A.2.1, as required by Required Action A.2.4, ensures that core conditions during operation at higher power levels are consistent with safety analyses assumptions. If Required Actions of Condition A are not met within their associated Completion Times, Required Action B.1 requires the plant be placed in a mode or condition in which the LCO requirements are not applicable. The allowed Completion Time of Required Action B.1 is reasonable based on operating experience regarding the amount of time it takes to reach MODE 2 from full power operation in an orderly manner and without challenging plant systems.

Since this change imposes new requirements, it is more restrictive and has no adverse impact on safety.

M3 CTS Specification 3.10.2.1.1, which requires that power distribution maps using the movable detector system be made to confirm that the F_Q limits are satisfied following initial loading or upon achieving equilibrium conditions after exceeding by 10% or more of RTP, is retained in ITS as a general Note to the Surveillance Requirements, and in ITS SR 3.2.1.1 has the Frequency changed to refueling interval and prior to exceeding 75% rated power. An additional restriction is imposed in the ITS to perform SR 3.2.1.1 within 12 hours of achieving equilibrium conditions after exceeding by 10% or more of rated power. Performing the Surveillance in MODE 1 prior to exceeding 75% RTP ensures that the $F_Q(Z)$ limit is met when RTP is achieved, because peaking factors are generally decreased as power level is increased. Verifying $F_Q(Z)$ at power levels ≥ 10% RTP above the THERMAL POWER of its last verification, 12 hours after achieving equilibrium conditions, ensures that $F_Q(Z)$ is within its limit at higher power levels. This change imposes more restrictive requirements, and has no adverse impact on safety.

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M4 CTS Specification 3.10.2.2.1.a, which requires that the reactor core be placed in an equilibrium condition where the Heat Flux Hot Channel Factor is satisfied and reestablish the target axial flux difference, is retained and restated in ITS 3.2.3 Required Action A.1. The CTS Required Action, as restated in ITS Required Action A.1, allows the option of reducing the target axial flux band to the ±3% band in order to obtain a lower V(Z) penalty. By restricting operation to the ±3% band rather than the ±5% band, V(Z) is reduced by approximately 3% resulting in a lower $F_Q^V(Z)$. With a lower V(Z) penalty, $F_Q^V(Z)$ may return to within limits without a power reduction. The option provided by Required Action A.1 is consistent with the PDC-3 axial offset control methodology used by Siemens Power Corporation for calculating cycle specific hot channel factor limits (Ref. 1). Because Required Action A.1 imposes the same requirement as CTS 3.10.2.2.1.a, this aspect of the change is administrative.

Also, ITS 3.2.1 Required Action A.1 is revised to include a completion time of 15 minutes for achieving the more restrictive target flux band, rather than reestablish the existing target flux band without a required completion time as allowed in the CTS. The Completion Time of 15 minutes provides an acceptable time to reevaluate $F_Q^V(Z)$ within the more restrictive target band to determine if $F_Q^V(Z)$ remains within limits. Since this change imposes the new requirement of a completion time to achieve a more restrictive target band, this change is more restrictive and has no adverse impact on safety.

M5 CTS Specification 3.10.2.2.1.b, which requires that reactor power be reduced if F_Q is not within limits, is retained in ITS Specification 3.2.1 Required Action A.2.1, with the additional requirement of a Completion Time of 30 minutes. The CTS does not impose a required Completion Time. The Completion Time of 30 minutes provides an acceptable time to reduce power in an orderly manner and without allowing the plant to remain in an unacceptable condition for an extended period of time. Since this change imposes the new requirement of a completion time to reduce THERMAL POWER, it is more restrictive and has no adverse impact on safety.

M6 CTS Specifications 3.10.2.2.2 defines an APL that permits operation slightly above reduced power levels that are required when hot channel factors, AFD, and QPTR are outside the required limits. Increasing power to the APL requires that the Axial Power Distribution Monitoring System (APDMS) be initiated. This provision in CTS is not retained in ITS. As a result, the ITS Required Actions for reducing power in response to exceeding power distribution limits will be followed without any provision for increasing power to above the ITS Required Action

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THERMAL POWER levels. Consequently, this change is more restrictive, and has no adverse impact on safety.

In conjunction with this more restrictive change, CTS Specification 4.11, which contains the surveillance requirements for the APDMS, is not retained in ITS. The APDMS is only required to be initiated to support THERMAL POWER levels above those contained in the CTS required actions.

Therefore, these changes are more restrictive, and have no adverse impact on safety.

M7 CTS Specification 3.10.2.1, which excludes applicability for maintaining $F_{\Delta H}$ within limits during physics testing, is not retained in ITS. ISTS Specification 3.2.2, "Nuclear Enthalpy Rise Hot Channel Factor ($F^N_{\Delta H}$)," does not allow a physics test exception to $F_{\Delta H}$ limits. The $F_{\Delta H}$ limits are applicable at all times when the reactor is at power. There are no physics tests performed that require an exception to the $F_{\Delta H}$ limits. Additionally, since the limits on $F^N_{\Delta H}$ ensure that the DNB design basis is met, a physics test exception to the $F^N_{\Delta H}$ limits is inappropriate. Since this change imposes new requirements, it is more restrictive and has no adverse impact on safety.

M8 CTS Specification 3.10.2.1.1, which requires that the enthalpy rise hot channel factor, $F_{\Delta H}$, be determined following initial core loading, has the Frequency changed in ITS SR 3.2.2.1 to refueling interval and prior to exceeding 75% RTP. As a result, a limit on THERMAL POWER is imposed for the initial performance of SR 3.2.2.1 following a refueling outage before THERMAL POWER reaches a level where a deviation between measured and predicted $F_{\Delta H}$ could significantly affect the assumptions in the safety analyses. This requirement ensures that $F^N_{\Delta H}$ limits are met at the beginning of each fuel cycle. This change imposes more restrictive requirements, and has no adverse impact on safety.

M9 CTS Specification 3.10.2.1.1, second paragraph, requires that the reactor power be reduced in the event that $F_{\Delta H}$ is not within limits. CTS Specification 3.10.2.1.1, second paragraph, is retained in ITS 3.2.2 as Required Actions A.1.1 and A.1.2. No completion time is required in the CTS for the required action. A Completion Time of 4 hours is imposed in the ITS for Required Action A.1.1 and A.1.2.1 A Completion Time of 72 hours is imposed for Required Action A.1.2.2. The allowed Completion Times provide an acceptable time to restore $F^N_{\Delta H}$ to within its limits without allowing the plant to remain in an unacceptable condition for an extended period of time. Since this change imposes new requirements, it is more restrictive and has no adverse impact on safety.

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M10 CTS Specification 3.10.2.1.1, second paragraph, includes the requirement that reactor power be limited to the fraction of RTP equal to $F_{\Delta H}^{limit}/F_{\Delta H}^{actual}$, and that the high neutron flux setpoint be reduced by the same ratio. ITS 3.2.2 Required Action A.1.2 requires that THERMAL POWER be reduced to less than 50% RTP, and that the Power Range Neutron Flux high setpoint be reduced to ≤ 55% RTP. The ITS requirement to reduce to below 50% RTP is more restrictive for values of $F_{\Delta H}$ in excess of the limits up to twice the required limits. Since the Surveillance Frequency is sufficiently short that any $F_{\Delta H}$ measurement in excess of limits is reasonably assured to be less than twice the required limits, this change is considered more restrictive, and has no adverse impact on safety. The more restrictive requirement was taken in ITS 3.2.2 Required Action A.1.2 to be consistent with ISTS. In addition, no completion time is required in the CTS for the setpoint reduction. A completion time of 72 hours is imposed in the ITS for Required Action A.1.2.2. The allowed Completion Time of 72 hours to reset the trip setpoints per Required Action A.1.2.2 recognizes that, once power is reduced, the safety analysis assumptions are satisfied and there is no urgent need to reduce the trip setpoints. This is a sensitive operation that may inadvertently trip the Reactor Protection System. Since this change imposes new requirements, it is more restrictive and has no adverse impact on safety.

M11 The CTS is revised in the ITS to adopt a Note to Condition A, Required Actions A.1.2.1, A.2, A.3 and Note, and B.1, from ISTS 3.2.2, "Nuclear Enthalpy Rise Hot Channel Factor ($F_{\Delta H}^{N}$)," when $F_{\Delta H}^{N}$ is not within limits. Reducing RTP to < 50% RTP increases the DNB margin and does not likely cause the DNBR limit to be violated in steady state operation. The reduction in trip setpoints ensures that continuing operation remains at an acceptable low power level with adequate DNBR margin. The allowed Completion Time of 4 hours for Required Action A.1.2.1 is consistent with those allowed for in Required Action A.1.1 and provides an acceptable time to reach the required power level from full power operation without allowing the plant to remain in an unacceptable condition for an extended period of time. The Completion Times of 4 hours for Required Actions A.1.1 and A.1.2.1 are not additive. Required Action A.2 requires that an incore flux map (SR 3.2.2.1) be obtained and the measured value of $F_{\Delta H}^{N}$ verified not exceed the allowed limit at the lower power level. The unit is provided 20 additional hours to perform this task over and above the 4 hours allowed by either Action A.1.1 or Action A.1.2.1. The Completion Time of 24 hours is acceptable because of the increase in the DNB margin, which is obtained at lower power levels, and the low probability of having a DNB limiting event within this 24 hour period. Additionally, operating experience has indicated that this Completion Time is sufficient to obtain the incore flux map, perform the required calculations, and evaluate $F_{\Delta H}^{N}$.

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Verification in accordance with Required Action A.3 that $F^N_{\Delta H}$ is within its specified limits after an out of limit occurrence ensures that the cause that led to the $F^N_{\Delta H}$ exceeding its limit is corrected, and that subsequent operation proceeds within the LCO limit. This Action demonstrates that the $F^N_{\Delta H}$ limit is within the LCO limits prior to exceeding 50% RTP, again prior to exceeding 75% RTP, and within 24 hours after THERMAL POWER is ≥ 95% RTP.

When Required Actions A.1.1 through A.3 cannot be completed within their required Completion Times, Required Action requires that the plant be placed in a mode in which the LCO requirements are not applicable. This is done by placing the plant in at least MODE 2 within 6 hours. The allowed Completion Time of 6 hours is reasonable, based on operating experience regarding the time required to reach MODE 2 from full power conditions in an orderly manner and without challenging plant systems.

Since this change imposes new requirements, it is more restrictive and has no adverse impact on safety.

M12 CTS Specification 3.10.2.2.2, which imposes additional requirements upon $F_Q(Z)$ (i.e., increases the measured $F_Q(Z)$ in the direction of the limit) if the enthalpy rise hot channel factor is increasing, is retained in ITS Surveillance Requirement 3.2.2.1 as a Note, and is further revised to ensure that $F_Q(Z)$ is reverified to be within the required F_Q limits. While the CTS requirement to remain within F_Q limits remains unchanged, the additional requirement to reverify that $F_Q(Z)$ is within the F_Q limits adds new requirements. Reverifying that $F_Q(Z)$ is within the F_Q limits prevents $F^V_Q(Z)$ from exceeding its limit for any significant period of time during the surveillance interval. Therefore, this change has no adverse impact on safety.

M13 CTS Specification 3.10.2.1.1, which requires that the target AFD be established following initial loading, includes a footnote that allows the "design target value" to be used during power escalation until extended operation is achieved and the target values can be determined from actual core parameters. The footnote to CTS Specification 3.10.2.1.1 does not include a specific Completion Time upon which the target flux difference must be established based on actual core parameters. The ITS requires that the target flux difference be initially determined within 31 EFPDs of refueling. Since the target flux difference varies slowly with core burnup, the Frequency of 31 EFPD after each refueling establishes an initial measurement of the target flux difference based upon actual core parameters before measured values vary excessively with design prediction. This change imposes more restrictive requirements, and has no adverse impact on safety.

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M14 CTS Specification 3.10.2.5, which excludes applicability for maintaining AFD within the target band during physics testing, is not retained in ITS. ITS Specification 3.2.3, "Axial Flux Difference (AFD) (PDC-3 Axial Offset Control Methodology)," does not allow a physics test exception for AFD Applicability. The AFD must be maintained as specified by LCO 3.2.3 at all times when the reactor is at power, and a physic test exception is inappropriate. Since this change imposes new requirements, it is more restrictive and has no adverse impact on safety.

M15 CTS Specification 3.10.2.5, which requires that the AFD be maintained within its target band, is revised in ITS LCO 3.2.3.b to also require that the AFD be within the acceptable operation limits. LCO 3.2.3 intends that the unit is operated with the AFD within the target band about the target flux difference. However, during rapid THERMAL POWER reductions, control bank motion may cause the AFD to deviate outside of the target band at reduced THERMAL POWER levels. This deviation does not affect the xenon distribution sufficiently to change the envelope of peaking factors that may be reached on a subsequent return to RTP with the AFD within the target band, provided the time duration of the deviation is limited. The cumulative penalty time assures that the time duration of the deviation is limited. Violating the LCO acceptable operation limits for AFD could produce unacceptable consequences if a Condition 2, 3, or 4 event occurs.

Similarly, CTS Specification 3.10.2.7.a, which requires actions to be taken if AFD is outside of its target band, is revised in ITS 3.2.3 Condition C.1 to apply Required Action C.1 when the AFD is outside of acceptable operation limits. If the indicated AFD is outside the target band and outside the acceptable operation limits provided in the COLR, the peaking factors assumed in accident analysis may be exceeded with the existing xenon condition. The Completion Time of 30 minutes allows for a prompt, yet orderly, reduction in power. Since this change adds requirements, this change is more restrictive, and has no adverse impact on safety.

M16 CTS Specification 3.10.2.6, which requires actions to be taken to maintain the AFD within the target band for rated power greater than 90% of rated power or 0.9 APL (whichever is less), and CTS Specification 3.10.2.7, which defines actions that result in accumulation of penalty deviation time when the reactor power is ≥ 50% rated power, and less than 90% rated power or 0.9 APL (whichever is less), are revised in ITS 3.2.3 Applicability to MODE 1 with THERMAL POWER > 15% RTP. Since this change imposes applicability for THERMAL POWER < 50%, this change is more restrictive. This change is being made to ensure that the distributions of xenon are consistent with safety analysis assumptions. Therefore, this change has no adverse impact on safety.

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The CTS requirement to maintain AFD to within the target band at a THERMAL POWER > 90% RTP or 0.9 APL (whichever is less), is revised in Required Action A.1 for THERMAL POWER ≥ 90% RTP or 0.9 APL (whichever is less). Because this change could potentially result in remaining outside the target band and accumulation of penalty deviation time at exactly 90% RTP or 0.9 APL (whichever is less) or could result in the reduction of THERMAL POWER to below 90% RTP or 0.9 APL (whichever is less) rather than ≤ 90% RTP or 0.9 APL (whichever is less), this change is more restrictive, and has no adverse impact on safety. This change is being made solely to maintain consistency with NUREG-1431.

Similarly, the CTS requirement to accumulate penalty deviation time at THERMAL POWER > 50% RTP and < 90% RTP or 0.9 APL (whichever is less), is revised in ITS to define the allowable range for cumulative penalty time to be THERMAL POWER ≥ 50% RTP and < 90% RTP or 0.9 APL (whichever is less). Because this change could result in the accumulation of penalty deviation time at exactly 50% RTP at the rate defined in LCO 3.2.3.b, this change is more restrictive, and has no adverse impact on safety. This change is being made solely to maintain consistency with NUREG-1431.

M17 CTS Specification 3.10.2.6, which contains the Required Action to return the AFD to the target band or reduce reactor power to less than 90% rated power, is modified in the ITS to require a Completion Time of 15 minutes to reduce THERMAL POWER to < 90% RTP, if the Required Action and associated Completion Time for restoration of AFD to within its target band is not met. No Completion Time for reduction of THERMAL POWER is required in the CTS. The addition of a Completion Time of 15 minutes to reduce power in ITS Required Action B.1 imposes new requirements; therefore, this change is more restrictive. The allowed Completion Time of 15 minutes provides an acceptable time to reduce power to < 90% RTP or 0.9 APL whichever is less without allowing the plant to remain in an unanalyzed condition for an extended period of time. Therefore, this change has no adverse impact on safety.

M18 The CTS is revised in the ITS to add a Note to Condition D, add Required Action D.1, and add SR 3.2.3.1, from ISTS 3.2.3, "Axial Flux Difference (AFD) (Constant Axial Offset Control (CAOC) Methodology)." CTS Specification 3.10.2.1.1 does not specifically apply when reactor power is less than 50% rated power except for the purposes of accumulating penalty hours.

Required Action D.1 requires that THERMAL POWER be reduced to < 15% RTP if the Required Actions and Completion Times of Condition C are not met. The CTS has no explicit required action if the Required Actions equivalent to Condition C are not met. If Required Action C.1 is not

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completed within its required Completion Time of 30 minutes, the axial xenon distribution starts to become significantly skewed with the THERMAL POWER ≥ 50% RTP. In this situation, the assumption that a cumulative penalty deviation time of 1 hour or less during the previous 24 hours while the AFD is outside its target band is acceptable at < 50% RTP, is no longer valid. Reducing the power level to < 15% RTP within the Completion Time of 9 hours and complying with LCO penalty deviation time requirements for subsequent increases in THERMAL POWER ensure that acceptable xenon conditions are restored.

ITS SR 3.2.3.1 requires verification that the AFD as indicated by the NIS excore channels is within the target band and consistent with the status of the AFD monitor alarm. The Surveillance Frequency of 7 days is adequate because the AFD is controlled by the operator and monitored by the process computer. Since these changes impose new requirements, they are more restrictive and have no adverse impact on safety.

M19 CTS Specification 3.10.2.8.a, which allows the indicated AFD to deviate from its target band at reactor power ≤ 50% rated power, is revised in the ITS to allow AFD to deviate outside the target band with THERMAL POWER < 50% RTP. Since this change does not allow operation with AFD outside of the target band at exactly 50% RTP, this change imposes more restrictive requirements, and has no adverse impact on safety. This change is being made solely to maintain consistency with NUREG-1431.

M20 CTS Specification 3.10.2.10, which requires that the AFD be logged every hour for the first 24 hours, and half-hourly thereafter, when the AFD alarm is out of service, is revised in ITS SR 3.2.3.2 to have a Frequency of once within 15 minutes and every 15 minutes thereafter when THERMAL POWER is ≥ 90% RTP, and once within 1 hour and every 1 hour thereafter when THERMAL Power is < 90 % RTP. This change is more restrictive in the case where THERMAL POWER ≥ 90% RTP, without regard to how long the AFD monitor has been out of service. This change incorporates a more appropriate frequency for logging AFD in the plant condition where the AFD limits are most restrictive, and where and deviation of AFD outside the target band would most impact the assumptions of the safety analyses. Therefore, this change has no adverse impact on safety.

M21 CTS Specification 3.10.2.1.1, which requires that the target AFD as a function of power level be established following initial core loading, has the Frequency changed in SR 3.2.3.3 of the ITS to within 31 EFPDs following each refueling. This change imposes a time limit in the Frequency for the initial performance of SR 3.2.3.3 after refueling. This change imposes more restrictive requirements, and has no adverse

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impact on safety. This change is being made solely to maintain consistency with NUREG-1431.

M22 CTS Specification 3.10.3.1, which excludes applicability for required actions when QPTR exceeds 1.02 during physics testing, is not retained in ITS. ISTS Specification 3.2.4, "Quadrant Power Tilt Ratio (QPTR)," does not allow a physics test exception for QPTR applicability. The QPTR must be maintained as specified by LCO 3.2.4 in MODE 1 with THERMAL POWER ≥ 50% RTP. There are no physics tests performed that require an exception to the QPTR limit. Additionally, since limits on QPTR ensure that the quadrant tilt assumed in the accident analyses remains valid, a physics test exception to the QPTR limit is inappropriate. Since this change imposes new requirements, it is more restrictive and has no adverse impact on safety.

M23 CTS Specification 3.10.3.1, which requires that actions be taken when QPTR exceeds the limit of 1.02, includes the required action that ". . . the tilt condition shall be eliminated within two hours. . ." This required action, which can be taken in lieu of other required actions that result in a reduction in THERMAL POWER, is not retained in ITS. Because the CTS required action allows two hours to lapse prior to applying a required action to reduce power, this change is more restrictive. The Completion Time of 2 hours in Required Action A.1 allows sufficient time to identify the cause and correct the tilt without the need for an additional 2 hours as allowed by the CTS. Therefore, this change has no adverse impact on safety.

M24 CTS Specification 3.10.3.1.a, which requires that power level be reduced in response to QPTR in excess of limit, is revised in the ITS to require a Completion Time of 2 hours to achieve the reduction in THERMAL POWER. Since this change adds a Completion Time requirement, this change is more restrictive. The Completion Time of 2 hours in Required Action A.1 allows sufficient time to identify the cause and correct the tilt. Therefore, this change has no impact on safety.

M25 CTS Specification 3.10.3.1.a, which requires reactor power to be reduced by more than two (2) percent of rated reactor power for every one (1) percent that the QPTR is in excess of the limit, is retained in ITS 3.2.4 Required Action A.1 and is revised to specify a Completion Time of 2 hours and to reduce power by three (3) percent for every percent of QPTR in excess of the limit. Since this change adds a Completion Time for the Required Action which did not exist previously, and restricts THERMAL POWER further as revised in the Required Action, this change is more restrictive.

ITS

Specification 3.5.2 (A1)

i. Power operation with less than three loops in service is prohibited — see 3.4

3.3.1.2 [RA A.1] [RA C.1] [RA C.2]

~~During power operation, the requirements of 3.3.1.1 may be modified to allow any one of the following components to be inoperable. If the system is not restored to meet the requirements of 3.3.1.1 within the time period specified, the reactor shall be placed in the hot shutdown condition utilizing normal operating procedures. If the requirements of 3.3.1.1 are not satisfied within an additional 48 hours, the reactor shall be placed in the cold shutdown condition utilizing normal operating procedures.~~

Required Action and associated completion time are not met

Mode 3 within 6 hours and Mode 4 within 12 hours

a. One accumulator may be isolated or otherwise inoperable relative to the requirements of 3.3.1.1.b for a period not to exceed four hours. — See 3.5.1 (L2)

~~b. If one safety injection pump becomes inoperable during normal reactor operation, the reactor may remain in operation for a period not to exceed 24 hours.~~

~~c. If one residual heat removal pump becomes inoperable during normal reactor operation, the reactor may remain in operation for a period not to exceed 24 hours.~~ — (L2)

With one or more train inoperable and at least 100% of the ECCS flow equivalent to a single OPERABLE train available, Restore train(s) to OPERABLE status within 72 hours — (L2)

Amendment No. ~~97~~, ~~115~~, ~~119~~, ~~146~~, 153

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Similarly, CTS Specification 3.10.3.2, which requires reactor power to be reduced by more than two (2) percent of rated thermal power for every one (1) percent of indicated power tilt, is revised in ITS 3.2.4 Required Action A.1 to reduce power by three (3) percent for every percent of QPTR in excess of the limit. This change is more restrictive.

With the QPTR exceeding its limit, a power level reduction of 3% RTP for each 1% by which the QPTR exceeds 1.00 is a conservative tradeoff of total core power with peak linear power. The Completion Time of 2 hours allows sufficient time to identify the cause and correct the tilt. Note that the power reduction itself may cause a change in the tilted condition. Therefore, these changes have no adverse impact on safety.

M26 The CTS is revised in the ITS to add a specific LCO to maintain QPTR ≤ 1.02, and add Required Actions A.2, A.3, A.4, A.5 and Note, A.6 and Note, SR 3.2.4.1 and Note 2, and SR 3.2.4.2 and Note from ISTS 3.2.4, "Quadrant Power Tilt Ratio (QPTR)." Since this change imposes new requirements, it is more restrictive and has no adverse impact on safety. LCO 3.2.4 precludes core power distributions that violate the fuel design criteria. Required Actions A.2 and A.4 ensure that the QPTR at the reduced thermal power levels reached in response to Required Action A.1 continues to be within its limit, or additional power reduction is required. The 12 hour Completion Time for Required Action A.2 is sufficient because any additional change in QPTR would be relatively slow.

The Completion Time of 24 hours for Required Action A.2 takes into consideration the rate at which peaking factors are likely to change, and the time required to stabilize the plant and perform a flux map. If these peaking factors are not within their limits, the Required Actions of these Surveillances provide an appropriate response for the abnormal condition. If the QPTR remains above its specified limit, the peaking factor surveillances are required each 7 days thereafter to evaluate $F^N_{\Delta H}$ and $F_Q(Z)$ with changes in power distribution.

The re-evaluation in Required Action A.4 is required to ensure that, before increasing THERMAL POWER to above the limit of Required Action A.1, the reactor core conditions are consistent with the assumptions in the safety analyses.

If the QPTR has exceeded the 1.02 limit and a re-evaluation of the safety analysis is completed and shows that safety requirements are met, the excore detectors are normalized in accordance with Required Action A.5 to eliminate the indicated tilt prior to increasing THERMAL POWER to above the limit of Required Actions A.1 or A.2. This is done to detect

TECHNICAL CHANGES - MORE RESTRICTIVE
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any subsequent significant changes in QPTR. Required Action A.5 is modified by a Note that prevents any ambiguity about the required sequence of actions.

Once the excore detectors are normalized to eliminate the indicated tilt (i.e., Required Action A.5 is performed), it is acceptable to return to full power operation in accordance with Required Action A.6. However, as an added check that the core power distribution at RTP is consistent with the safety analysis assumptions, Required Action A.6 requires verification that $F_Q(Z)$ and $F^N_{\Delta H}$ are within their specified limits within 24 hours of reaching RTP. As an added precaution, if the core power does not reach RTP within 24 hours, but is increased slowly, then the peaking factor surveillances must be performed within 48 hours of the time when the ascent to power was begun. These Completion Times are intended to allow adequate time to increase THERMAL POWER to above the limit of Required Action A.1 and A.2, while not permitting the core to remain with unconfirmed power distributions for extended periods of time. Required Action A.6 is modified by a Note that requires the peaking factor surveillances be performed at operating power levels, which can only be accomplished after the excore detectors are normalized in accordance with Required Action A.5 to remove the tilt and the core returned to power.

Therefore, these changes have no adverse impact on safety.

M27 CTS Specification 1.8 allows calculation of QPTR with only three (3) operable power range nuclear instruments without restrictions on reactor power. This specification is retained as Note 1 to ITS SR 3.2.4.1 with the additional restriction from ISTS SR 3.2.4.1 that THERMAL POWER must be < 75% RTP prior to determining QPTR with only three operable excore detectors. Since this change adds a restriction for THERMAL POWER levels when performing a surveillance under certain conditions, this change is more restrictive. In this condition, it would be inappropriate to determine QPTR above 75% RTP since the QPTR results would be invalid for the quadrant where the inoperable excore detector exists. When THERMAL POWER is ≥ 75% RTP and one power range nuclear instrument is inoperable, QPTR is determined using the Incore Flux Mapping System. Therefore, this change has no adverse impact on safety.

M28 CTS Specification 3.10.2.1.1, which requires that the Overpower Delta-Temperature (OPΔT) and Overtemperature Delta-Temperature (OTΔT) trip setpoints be reduced if ". . .subsequent incore mapping cannot . . . demonstrate that the hot channel factors are met." is retained in ITS 3.2.1 Required Action A.2.3 to reduce the OPΔT and OTΔT setpoints. However, Required Action A.2.3 must be followed in ITS regardless of the

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means by which the hot channel factors are measured, i.e., "subsequent incore mapping." Because the ITS Required Action is not restricted by the method used for hot channel factor measurement, and it is inappropriate to identify flux mapping as a means to satisfying the Required Action to reduce the OPΔT and OTΔT setpoints, this change is considered more restrictive, and has no adverse impact on safety.

TECHNICAL CHANGES - LESS RESTRICTIVE (GENERIC)

LA1 The details contained in CTS Specifications 3.10.2.1, and 3.10.2.2 related to the power distribution limits of $F_Q(Z)$, are relocated to the COLR. This detail, which includes the mathematical relationship of the F_Q, i.e., $F_Q(Z)$, to the normalized hot channel factor, i.e., K(Z), as a function of power, and the associated engineering uncertainty factors, is not required to be in the ITS to provide adequate protection of the health and safety of the public, since the ITS still retains the requirement to remain within the hot channel factor limits specified in the COLR. This approach provides an effective level of regulatory control and provides for a more appropriate change control process. The level of safety of facility operation is unaffected by the change because there is no change in the operational requirements. Furthermore, NRC and licensee resources associated with processing license amendments to these requirements will be reduced. Therefore, relocation of this detail is acceptable.

LA2 The details contained in CTS Specifications 3.10.2.1, related to the power distribution limits of $F^N_{\Delta H}$, are relocated to the COLR. This detail, and the associated engineering uncertainty factors, is not required to be in the ITS to provide adequate protection of the health and safety of the public, since the ITS still retains the requirement to remain within the hot channel factor limits specified in the COLR. This approach provides an effective level of regulatory control and provides for a more appropriate change control process. The level of safety of facility operation is unaffected by the change because there is no change in the operational requirements. Furthermore, NRC and licensee resources associated with processing license amendments to these requirements will be reduced. Therefore, relocation of this detail is acceptable.

LA3 The details contained in CTS Specification 3.10.2.11, third and fourth sentences, related to the redefinition of the target band between the less restrictive and the more restrictive ranges, are relocated to the Bases to ITS LCO 3.2.1. This detail, which redefines the target band from the more restrictive to the less restrictive range for AFD, is not required to be in the ITS to provide adequate protection of the health and safety of the public, since the ITS still retains the requirement to remain within the target band specified in the COLR. This approach provides an effective level of regulatory control and provides for a more appropriate change control process. The level of safety of facility operation is unaffected by the change because there is no change in the operational requirements. Furthermore, NRC and licensee resources associated with processing license amendments to these requirements will be reduced. Therefore, relocation of this detail is acceptable.

LA4 The details contained in CTS Specification 3.10.2.2.2, that define the variable expression $F_Q(Z)$ as the measured hot channel factor, are

TECHNICAL CHANGES - LESS RESTRICTIVE (GENERIC)
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relocated to the ITS bases. This detail, is not required to be in the ITS to provide adequate protection of the health and safety of the public, since the ITS still retains the requirement to remain within the limits of $F_Q(Z)$. Changes to the ITS bases are controlled in accordance with the ITS Section 5.5.14, "Technical Specifications (TS) Bases Control Program." This approach provides an effective level of regulatory control and provides for a more appropriate change control process. The level of safety of facility operation is unaffected by the change because there is no change in the operational requirements. Furthermore, NRC and licensee resources associated with processing license amendments to these requirements will be reduced. Therefore, relocation of this detail is acceptable.

LA5 Details from CTS Specification 3.10.2.1.1 which require power distribution maps using the movable detector system shall be made are relocated to the Background section of the Bases to ITS 3.2.2, which states, "$F^N_{\Delta H}$ is not directly measurable but is inferred from a power distribution map obtained with the moveable incore detector." This statement, in conjunction with the requirements of ITS SR 3.2.2.1 to verify $F^N_{\Delta H}$ is within limits, effectively retains the CTS requirement in the ITS bases. This detail, is not required to be in the ITS to provide adequate protection of the health and safety of the public, since the ITS still retains the requirement to remain within the limits of $F^N_{\Delta H}$. Changes to the ITS bases are controlled in accordance with the ITS Section 5.5.14, "Technical Specifications (TS) Bases Control Program." This approach provides an effective level of regulatory control and provides for a more appropriate change control process. The level of safety of facility operation is unaffected by the change because there is no change in the operational requirements. Furthermore, NRC and licensee resources associated with processing license amendments to these requirements will be reduced. Therefore, relocation of this detail is acceptable.

TECHNICAL CHANGES - LESS RESTRICTIVE (SPECIFIC)

L1 CTS Specification 3.10.2.1, requires that the F_Q limits be applicable at all times except during physics testing, is revised in ITS Specification 3.2.1, "Heat Flux Hot Channel Factor ($F_Q(Z)$) (F_Q Methodology)," to require that the F_Q limits be applicable in MODE 1 only, and is less restrictive. This change is acceptable, however, since it is only in MODE 1 that a challenge to the F_Q limits can be made. This change does not reduce any margins to safety and is consistent with NUREG-1431 which utilizes a similar power distribution limit methodology.

L2 CTS Specification 3.10.2.1.1, which requires that if the hot channel factors cannot be returned to within limits within 24 hours then the OPΔT and OTΔT setpoints will be reduced by a fraction equal to $F_Q(Z)_{limit}/F_Q^V(Z)_{actual}$, is revised in ITS Specification 3.2.1, "Power Distribution Limits," Required Action A.2.3, to require that if F_Q cannot be returned to within limits within 72 hours the OPΔT and OTΔT setpoints will be reduced. This is a relaxation of requirements, and is less restrictive. This change is acceptable because appropriate time is needed to change the OTΔT and OPΔT setpoints; the 72 hour time period permits the possibility of restoring hot channel factors within limits and may avoid resetting the OTΔT and OPΔT setpoints twice while in Condition A; and, THERMAL POWER has already been reduced to ensure that the hot channel factors are within limits during the time that the plant remains in Condition A. This change is consistent with NUREG-1431.

L3 CTS Specification 3.10.2.1.1, which requires that the $F_{\Delta H}$ limits be applicable at all times except during physics testing, is revised in ITS Specification 3.2.2, "Nuclear Enthalpy Rise Hot Channel Factor ($F^N_{\Delta H}$)," to require that the $F_{\Delta H}$ limits be applicable in MODE 1 only, which is less restrictive with respect to applicability to MODEs other than MODE 1. This change is acceptable, however, since it is only in MODE 1 that sufficient THERMAL POWER occurs that could result in a challenge to the $F_{\Delta H}$ limits. This change does not reduce any margins to safety and is consistent with NUREG-1431 which utilizes a similar power distribution limit methodology.

L4 CTS Specification 3.10.2.1.1 contains a surveillance requirement that $F_{\Delta H}$ be verified after exceeding by 10% the power level at which $F_Q(Z)$ was last determined once equilibrium conditions are established following refueling. This surveillance requirement is retained in ITS as SR 3.2.2.1 with the Frequency requirement that $F_{\Delta H}$ be verified prior to exceeding 75% RTP following refueling, and once per 31 EFPDs thereafter, but without the additional restriction of verifying $F_{\Delta H}$ after exceeding by 10% the power level at which F_Q was last measured. This is a relaxation of requirements, and is less restrictive. This change is acceptable, however, since further reconfirmation of $F_{\Delta H}$ in addition to the Frequency stated in ITS SR 3.2.2.1 is unnecessary. The measurement of $F_{\Delta H}$ is a function of fuel burnup and is relatively insensitive to

TECHNICAL CHANGES - LESS RESTRICTIVE (SPECIFIC)
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changes in reactor power. This change is consistent with NUREG-1431 which utilizes a similar power distribution limit methodology.

L5 CTS Specification 3.10.2.1.1, second and third paragraph, which requires that the OTΔT and OPΔT setpoints be reduced by the fraction $F_{\Delta H limit}/F_{\Delta H actual}$ if the out of limit condition for $F_{\Delta H}$ is not corrected within 24 hours, is not retained in ITS. This is a relaxation of requirements, and is less restrictive. This change is acceptable, since the Required Action to reduce THERMAL POWER to below 50% will likely result in an enthalpy rise hot channel factor that is well below the limiting value at this power level. Further reduction of the OPΔT and OTΔT setpoints is a small contribution to the safety margin, i.e., a OPΔT or OTΔT trip could potentially occur at the reduced setpoint prior to a high neutron flux trip at 55% RTP in response to a transient. While the earlier OPΔT or OTΔT trip could result in a slight improvement in safety margin, this contribution to the safety margin is expected to be small. This change is consistent with NUREG-1431 which utilizes a similar power distribution limit methodology, and has no adverse impact on safety.

The requirement for reducing the ΔT setpoints was introduced into CTS by Amendment 13 by NRC letter dated October 17, 1975. Amendment 13 incorporated into the CTS the Westinghouse Constant Axial Offset Control (CAOC) methodology in use at the time. This methodology included fractional reductions in THERMAL POWER and reactor trip setpoints, and accommodated partial length control rods for axial offset control and load following, which are no longer in use. The methodology did not include the more limiting percentage reductions in THERMAL POWER for the condition when $F_Q(Z)$ was in excess of limits which was introduced into the CTS by Amendment 87 by NRC letters dated November 7 and 20, 1984. Amendment 87 to the CTS utilized the Siemens Power Corporation Axial Offset Control Methodology. Because Required Action A.1.2.1 requires THERMAL POWER to be reduced below 50% power in the event that $F_{\Delta H}$ is in excess of limits, the CTS requirement to reduce the ΔT setpoints by a fractional amount is not required in ITS to maintain assumptions within the safety analyses.

L6 CTS Specification 3.10.2.7.a, which requires that in the event that the cumulative penalty time for AFD outside the target band exceeds one hour, the high neutron flux setpoint be reduced to no greater than 55% of rated power, is not retained in the ITS. This is a relaxation of requirements, and is less restrictive. This change is acceptable because Required Action C.1 assures that the plant remains within analyzed parameters for AFD by reducing power and thereby adding margin for AFD to the analyzed assumptions; Required Action D.1 assures that if Required Action C.1 cannot be met within the Completion Time, reactor power is further reduced to add additional margin for AFD to the

TECHNICAL CHANGES - LESS RESTRICTIVE (SPECIFIC)
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analyzed assumptions; lowering the high neutron flux setpoints as an additional action does not add appreciable margin to the AFD assumptions in the accident analyses; and, the lower high flux setpoints are not included in the safety analysis assumptions. This change is consistent with NUREG-1431 which utilizes a similar power distribution limit methodology.

L7 CTS Specification 3.10.2.9, which allows calibration of the excore detectors if the AFD is not outside of the target band for > 90% rated power, and if the AFD does not exceed the limits specified in the COLR for reactor power between 50% and 90% rated power, is revised in the ITS Note to Applicability for LCO 3.2.3 to allow up to 16 hours to be accumulated with AFD outside of the target band without penalty deviation time while the excore detectors are being calibrated. This is a relaxation of requirements, and is less restrictive. This change is acceptable because some deviation from the target band is necessary to perform the calibration. The typical calibration for the excore detectors is to cause the core to deviate from the target band along the Z axis by inserting rods and measuring the indicated axial offset or detector currents of the excore detectors against actual axial offset and then raising the rods to allow xenon effects to cause power to shift up along the Z axis and then taking another measurement. The resulting plot of axial offsets are used to calibrate the slope of excore detector response to determine the excore detector indicated axial offset as a function of actual axial offset. The effect of this calibration method is to alternate axial offset between a plus and minus axial offset for short periods of time, but the overall effect on axial xenon distribution is small. It should be noted that this method of calibration has been replaced with a single point of measurement calibration methodology which utilizes past calibration data to determine an empirical geometric relationship such that a single point of measurement may be used to establish the slope of the excore detector calibration curve. As a result the necessary number of times and time duration that the axial offset is required to be outside the target band for the purpose of excore calibration has been significantly reduced. However, since the multipoint calibration may be required in the future (i.e., such as in the event of a changeout of excore detector), this note is necessary to allow a multipoint calibration to be performed without requiring multiple entries into LCO 3.2.3 conditions due to the calibration. This change is consistent with NUREG-1431 which utilizes a similar power distribution limit methodology.

L8 CTS Specification 3.10.2.10, which requires that the AFD be logged every hour for the first 24 hours, and half-hourly thereafter, when the AFD alarm is out of service, is revised in ITS SR 3.2.3.2 to have a Frequency of once within 15 minutes and every 15 minutes thereafter when

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THERMAL POWER is ≥ 90% RTP, and once within 1 hour and every 1 hour thereafter when THERMAL POWER is < 90 % RTP. This change is less restrictive in the case that the AFD monitor alarm remains out of service for greater than 24 hours and THERMAL POWER < 90% RTP. This change is acceptable because the likelihood of AFD being out of the target band decreases as steady state operation continues; and, AFD is also more likely to remain within the target band with THERMAL POWER < 90%. This change is consistent with NUREG-1431 which utilizes a similar power distribution limit methodology.

L9 CTS Specification 3.10.2.1.1 contains a surveillance requirement that the target AFD be established after exceeding by 10% the power level at which F_Q was last determined once equilibrium conditions are established following refueling. This surveillance requirement is retained in ITS as SR 3.2.3.3 with the Frequency requirement that the target AFD be established prior to exceeding 75% RTP following refueling, and once per 31 EFPDs thereafter, but without the additional restriction of establishing the target again after exceeding by 10% the power level at which F_Q was last measured. This is a relaxation of requirements, and is less restrictive. This change is acceptable, however, since determination of the target AFD is adequately addressed in the PDC-3 axial offset control methodology and is reflected in the requirements stated in ITS LCO 3.2.3. This change is consistent with NUREG-1431 which utilizes a similar power distribution limit methodology.

L10 CTS Specification 3.10.3.1, which excludes applicability for required actions when QPTR exceeds the limit is retained in ITS as an Applicability of MODE 1 with THERMAL POWER > 50% RTP. Since the restated applicability excludes the CTS required applicability for QPTR of exactly 50% RTP this change is considered less restrictive. This change is acceptable since the likelihood of a quadrant power tilt in excess of the limit at exactly 50% RTP resulting in an unanalyzed condition is very small. This probability is small because extremely small variations in THERMAL POWER occur at any power level and these variations result in THERMAL POWER being different than exactly 50% RTP. Depending upon the degree of resolution that the nuclear instrumentation signals are measured (i.e., resolution obtainable by observing the Emergency Response Facility Information System computer data in a range of approximately ± 0.1% RTP), the small differences can be readily observed and indicate that power levels vary slightly about any referenced value. Since the CTS requires QPTR to be within limits except for power increases below 50% rated power, a small variation about a power level slightly below exactly 50% RTP could result in a small "power increase" below 50% RTP. However, when the power level reaches exactly 50% RTP, the small "power increase" that may occur beyond exactly 50% RTP falls within the applicability of the CTS and the

TECHNICAL CHANGES - LESS RESTRICTIVE (SPECIFIC)
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core analysis in accordance with the PDC-3 Axial Offset Control Methodology. Therefore, no practical condition exists where an unanalyzed condition could occur at exactly 50% RTP due to this change. This change is consistent with NUREG-1431.

L11 CTS Specification 3.10.3.1.a, which requires that the power range high flux setpoint be reset by two (2) percent for every percent that QPTR exceeds 1.0, is not retained in the ITS. Because this change eliminates a requirement, this change is less restrictive. This change is acceptable because the Required Actions remaining in the ITS result in an appropriate reduction in THERMAL POWER to maintain the required safety margins when QPTR is in excess of the limit. The peaking factors $F^N_{\Delta H}$ and $F_q(Z)$ are of primary importance in ensuring that the power distribution remains consistent with the initial conditions used in the safety analyses. Required Action A.3 assures that the peaking factors are maintained within limits. If the results from the flux map performed in accordance with Required Action A.3 indicate that peaking factors are not within limits, Required Actions associated with LCO 3.2.1 and/or LCO 3.2.2 would be performed which could result in additional THERMAL POWER restrictions and if not corrected, a reduction in power range trip setpoints, and possibly overtemperature and overpressure trip setpoints. In so doing, the assumptions in the safety analyses would be assured by the performance of Required Actions associated with LCOs 3.2.1 and 3.2.2. Conversely, if the results from the flux map indicate that LCOs 3.2.1 and 3.2.2, the assumptions in the safety analyses are assured without further need to reduce the power range high neutron flux trip setpoints. This change is consistent with NUREG-1431.

L12 CTS Specification 3.10.3.1.b, which requires that reactor power be reduced to 50% rated power and the power range high flux setpoint reset to 55%, if QPTR is not eliminated within 24 hours, is revised as Required Action B.1 to ITS LCO 3.2.4, This change is less restrictive for two reasons. First, the addition of Required Actions A.2, A.3, A.4, A.5, and A.6, result in the possibility of continued plant operation above 50% RTP with QPTR in excess of the limit as long as the required power reductions are maintained, the F_Q and $F_{\Delta H}$ limits are maintained, and the QPTR condition remains analyzed for the duration of the cycle. CTS Specification 3.10.3.1.b has no such provisions to allow operation above 50% power if the quadrant power tilt remains for more than 24 hours. This change is acceptable because the Required Actions added to Condition A result in the plant remaining in an analyzed condition when the Required Actions are satisfied.

Secondly, the requirement to reset the power range high flux setpoints to 55% power is not retained in Required Action B.1. This change is

TECHNICAL CHANGES - LESS RESTRICTIVE (SPECIFIC)
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acceptable because the Required Actions remaining in the ITS result in an appropriate reduction in THERMAL POWER to maintain the required safety margins when QPTR is in excess of the limit. The peaking factors $F^N_{\Delta H}$ and $F_q(Z)$ are of primary importance in ensuring that the power distribution remains consistent with the initial conditions used in the safety analyses. Required Action A.3 assures that the peaking factors are maintained within limits. If the results from the flux map performed in accordance with Required Action A.3 indicate that peaking factors are not within limits, Required Actions associated with LCO 3.2.1 and/or LCO 3.2.2 would be performed which could result in additional THERMAL POWER restrictions and if not corrected, a reduction in power range trip setpoints, and possibly overtemperature and overpressure trip setpoints. In so doing, the assumptions in the safety analyses would be assured by the performance of Required Actions associated with LCOs 3.2.1 and 3.2.2. Conversely, if the results from the flux map indicate that LCOs 3.2.1 and 3.2.2, the assumptions in the safety analyses are assured without further need to reduce the power range high neutron flux trip setpoints. This change is consistent with NUREG-1431.

L13 CTS Specifications 3.10.3.2 and 3.10.3.3, which restrict operation with the QPTR in excess of 1.09, is not retained in the ITS. The required actions could potentially result in transition to MODE 3. These restrictions are not retained in the ITS to be consistent with NUREG-1431 which defines required actions specific to the individual LCOs and does not mix required actions from different LCOs. Additionally, the Required Actions to LCO 3.2.4 can result in operation at reduced THERMAL POWER levels. This change is a relaxation of requirements and is less restrictive because plant operation may continue in MODE 1 with QPTR > 1.09 if Required Actions to ITS LCO 3.2.4 were met. This change is acceptable because operation of the plant in accordance with the Required Actions of ITS LCO 3.2.4 in conjunction with LCOs 3.2.1, 3.2.2, and 3.2.3, assures that assumptions in the safety analysis are maintained.

The QPTR limits ensure that $F^N_{\Delta H}$ and $F_Q(Z)$ remain below their limiting values by preventing an undetected change in the gross radial power distribution. The peaking factors $F^N_{\Delta H}$ and $F_q(Z)$ are of primary importance in ensuring that the power distribution remains consistent with the initial conditions used in the safety analyses. If the QPTR is greater than 1.09, Required Action A.3 assures that the peaking factors are maintained within limits. If the results from the flux map performed in accordance with Required Action A.3 indicate that peaking factors are not within limits, Required Actions associated with LCO 3.2.1 and/or LCO 3.2.2 would be performed which could result in additional THERMAL POWER restrictions. In so doing, the assumptions in

TECHNICAL CHANGES - LESS RESTRICTIVE (SPECIFIC)
(continued)

the safety analyses would be assured by the performance of Required Actions associated with LCOs 3.2.1 and 3.2.2. Conversely, if the results from the flux map indicate that LCOs 3.2.1 and 3.2.2, the assumptions in the safety analyses are assured. This change is consistent with NUREG-1431.

REFERENCES

1. ANF-88-054(P), "PDC-3: Advanced Nuclear Fuels Corporation Power Distribution Control for Pressurized Water Reactors and Application of PDC-3 to H. B. Robinson Unit 2," Advanced Nuclear Fuels Corporation, Richland, WA 99352 (Submitted to NRC by CP&L letter dated August 24, 1989, Proprietary).

1 In the conversion of the HBRSEP current Technical Specifications (CTS) to the proposed plant specific Improved Technical Specifications (ITS), certain sections of NUREG-1431, "Improved Standardized Technical Specifications - Westinghouse Plants," Revision 1, (ISTS) are not incorporated into ITS because they are not applicable. Specifically, ISTS Section 3.2.1A, "Heat Flux Hot Channel Factor ($F_Q(Z)$) (F_{xy} Methodology)," and Section 3.2.3B, "Axial Flux Difference (AFD) (Relaxed Axial Offset Control (RAOC) Methodology), are not applicable to the H. B. Robinson Steam Electric Plant (HBRSEP), Unit No 2, and are not included in the ITS.

2 In the conversion of the HBRSEP current Technical Specifications (CTS) to the proposed plant specific Improved Technical Specifications (ITS), certain wording preferences and conventions are adopted which do not result in technical changes (either actual or interpretational). Editorial changes which involve the insertion of plant specific terms or parameters are used to preserve consistency with the CTS and licensing basis.

3 The expression F_Q^W is replaced with $F_Q^V(Z)$; the expression F_Q^C is not used; and, ISTS Surveillance Requirement (SR) 3.2.1.1 is not included in the ITS, to be consistent with the PDC-3 axial offset control methodology. The expression $F_Q^V(Z)$ is equivalent to F_Q^W utilized in the ISTS for Constant Axial Offset Control (CAOC) Methodology. The expression F_Q^C as used in the CAOC methodology is bounded by the other expressions of $F_Q(Z)$ in the PDC-3 axial offset control methodology. ISTS SR 3.2.1.2 is included in ITS as SR 3.2.1.1, which verifies that $F_Q^V(Z)$ is within the limits to satisfy Limiting Condition for Operations (LCO) 3.2.1. The changes in nomenclature and surveillance requirements are in accordance with the PDC-3 Axial Offset Control Methodology (Ref. 1).

4 ITS 3.2.1 Required Action A.1 is added to reduce AFD target band limits to restore F^V_Q (Z) to within limits and to delete Required Action B.1 as presented in the ISTS. The PDC-3 axial offset control methodology provides two distinct target bands for operation which consist of a ± 3% target band and a ± 5% target band. Required Action B.1 as presented in ISTS is not a valid action for the PDC-3 methodology (Ref. 1), and is not used in the ITS. Core analyses in accordance with the PDC-3 axial offset control methodology do not include a means to interpolate between the two allowable target bands. Either one of the two target bands must be chosen in order to evaluate the F^V_Q (Z) limits.

5 ISTS Specification 3.2.1 is modified to increase the Completion Time for Required Action A.2.1 to 30 minutes to allow for accomplishment of ITS - 3.2.1 Required Action A.1 before Required Actions A.2.1 through A.2.4. In the event that the out of limit condition cannot be corrected within 15 minutes by operating at the more restrictive target band in accordance with ITS RA A.1, the remaining Required Actions in Condition A to reduce THERMAL POWER and reduce ΔT trip setpoints are required.

These actions are in accordance with the current licensing basis (Ref. 1). Discussion of Change (DOC) M4 describes the more restrictive aspect of the ITS requirement of 30 minutes in comparison to the CTS requirement.

6 ISTS Specification 3.2.1 is revised to include the requirement to additionally reduce the Overtemperature Delta-Temperature (OTΔT) setpoint in ITS 3.2.1 Required Action A.2.3, in order to maintain consistency with the safety analyses and the Current Licensing Basis.

7 ISTS Specification 3.2.1 is modified to delete an applicability Note that relates to trending $F_Q^C(Z)/K(Z)$, and ISTS Specification 3.2.2 is modified to include an applicability Note to SR 3.2.2.1 to account for trending of $F_{\Delta H}^N$. The PDC-3 axial offset control methodology utilizes $F_{\Delta H}^N$ to trend the approach to power distribution limits rather than $F_Q^C(Z)/K(Z)$, which is used in the CAOC methodology.

8 ISTS Specification 3.2.3 is modified in title to be "Axial Flux Difference (AFD) (PDC-3 Axial Offset Control Methodology)," to replace the term "Constant Axial Offset Control (CAOC)." The PDC-3 axial offset control methodology is the approved methodology for neutronic calculations for Siemens Power Corporation manufactured fuel for HBRSEP. CAOC refers to the approved methodology for Westinghouse manufactured fuel.

9 ISTS Specification 3.2.3 is modified throughout where references to 90% RATED THERMAL POWER (RTP) occur to also require that THERMAL POWER be less than 0.9 Allowable Power Level (APL), whichever is less. The limitation to 0.9 APL is necessary to keep Axial Flux Difference (AFD) deviations from the target band within the requirements of the PDC-3 axial offset control methodology.

10 ISTS Specification 3.2.3 is modified in a note to LCO 3.2.3 to include the APL limitation that applies to THERMAL POWER when determining the AFD to its target band. The PDC-3 axial offset control methodology requires that the AFD limitation curves be adjusted when the APL is less than 100%.

11 ISTS SR 3.2.3.3 is deleted, and subsequent SR are renumbered. The Specification is also modified by changing the Frequency of ITS SR 3.2.3.3 to 31 Effective Full Power Days (EFPDs) from 92 EFPDs. The PDC-3 axial offset control methodology does not allow the use of linear interpolation to determine the target flux values. The Frequency is increased in order to provide an adequate interval between measurement of the target flux difference in the absence of an approved interpolative method.

12 ISTS Specification 3.2.3 is modified by adding Note 2 to ITS SR 3.2.3.3 to require that the target flux difference be determined in conjunction

with the measurement of the heat flux hot channel factor, $F_Q(Z)$, in accordance with ITS SR 3.2.1.1. The performance of SR 3.2.3.3. in conjunction with SR 3.2.1.1 is a requirement of the PDC-3 axial offset control methodology.

13 ISTS Specification 3.2.4 is modified to replace the term, "calibrated excore detectors to show a zero tilt," with, "normalize excore detectors to eliminate the tilt," in order to clarify that the measured QUADRANT POWER TILT RATIO (QPTR) need not precisely equate to zero prior to increasing THERMAL POWER above the level determined by ITS 3.2.4 Required Action A.1, and that the Required Action is a normalization of excore detector indications rather than a calibration, i.e., performance of SR 3.3.1.10.

14 ISTS Specification 3.2.4 is modified to include applicability of ITS 3.2.4 Required Action A.2 to the Completion Time for Required Actions A.5 and A.6, to reflect that THERMAL POWER limitations from either Required Action A.1 or A.2 may be more limiting.

15 Not used.

16 ISTS Specification 3.2.3 is modified to include Required Action C.2 to ensure consistency in the analyses performed in accordance with the PDC-3 axial offset control methodology.

REFERENCES

1. ANF-88-054(P), "PDC-3: Advanced Nuclear Fuels Corporation Power Distribution Control for Pressurized Water Reactors and Application of PDC-3 to H. B. Robinson Unit 2," Advanced Nuclear Fuels Corporation, Richland, WA 99352 (Submitted to NRC by CP&L letter dated August 24, 1989, Proprietary).

BASES

LCO (continued)

The expression for $F_Q^W(Z)$ is:

$$F_Q^W(Z) = F_Q^C(Z)\,W(Z)$$

where $W(Z)$ is a cycle dependent function that accounts for power distribution transients encountered during normal operation. The $W(Z)$ is included in the COLR.

The $F_Q(Z)$ limits define limiting values for core power peaking that precludes peak cladding temperatures above 2200°F during either a large or small break LOCA.

Insert B.3.2.1-2

This LCO requires operation within the bounds assumed in the safety analyses. Calculations are performed in the core design process to confirm that the core can be controlled in such a manner during operation that it can stay within the LOCA $F_Q(Z)$ limits. If $F_Q(Z)$ cannot be maintained within the LCO limits, reduction of the core power is required.

Violating the LCO limits for $F_Q(Z)$ produces unacceptable consequences if a design basis event occurs while $F_Q(Z)$ is outside its specified limits.

APPLICABILITY

The $F_Q(Z)$ limits must be maintained in MODE 1 to prevent core power distributions from exceeding the limits assumed in the safety analyses. Applicability in other MODES is not required because there is either insufficient stored energy in the fuel or insufficient energy being transferred to the reactor coolant to require a limit on the distribution of core power.

Insert B.3.2.1-1

ACTIONS

A.1

Reducing THERMAL POWER by ≥ 1% RTP for each 1% by which $F_Q^W(Z)$ exceeds its limit, maintains an acceptable absolute power density. $F_Q^W(Z)$ is $F_Q^M(Z)$ multiplied by a factor accounting for manufacturing tolerances and measurement uncertainties. $F_Q^M(Z)$ is the measured value of $F_Q(Z)$. The Completion Time of 15 minutes provides an acceptable time to reduce power in an orderly manner and without allowing the plant to remain in an unacceptable condition for an extended period of time.

(continued)

A.1

When operation is restricted to the ±3% target band, the V(Z) penalty is minimized, and the $F_Q^V(Z)$ is reduced. Thus when operation is restricted to the more restrictive target band, the result may be that $F_Q^V(Z)$ is within limits, and no reduction in THERMAL POWER is required. In the event that the reduced target band does not result in an acceptable $F_Q^V(Z)$, the THERMAL POWER will be reduced in accordance with Required Action A.2.1. The Completion Time of 15 minutes provides an acceptable time to reevaluate $F_Q^V(Z)$ within the more restrictive target band to determine if $F_Q^V(Z)$ remains within limits.

INSERT B.3.2.1-2

$F_Q(Z)$ is the measured $F_Q^N(Z)$ multiplied by the engineering factor F_Q^E (1.03) and the measurement uncertainty factor F_U^N (1.05) at the time of target flux determination from a power distribution map using the movable incore detectors. The PDC-3 axial offset control methodology provides two distinct target bands for operation which are the ±3% and the ±5% target bands. The target band that is selected determines the V(Z) penalty to be applied in the calculation of $F_Q^V(Z)$. Redefinition of the target band from more restrictive to less restrictive ranges between determinations of the target AFD is allowed when appropriate redefinitions of Allowable Power Level (APL) are made. Redefinition of the target band from less restrictive to more restrictive ranges is allowed only in conjunction with the determination of a new target AFD. AFD and APL requirements are discussed in the Bases of LCO 3.2.3.

BASES

SURVEILLANCE REQUIREMENTS

SR 3.2.1~~2~~ (continued)

maximum $F_Q(Z)$ calculated to occur in normal operation. $F_Q^W(Z)$.

The limit with which $F_Q^W(Z)$ is compared varies inversely with power and directly with the function K(Z) provided in the COLR.

The W(Z) curve is provided in the COLR for discrete core elevations. Flux map data are typically taken for 30 to 75 core elevations. $F_Q(Z)$ evaluations are not applicable for the following axial core regions, measured in percent of core height:

a. Lower core region, from 0 to 15% inclusive; and

b. Upper core region, from 85 to 100% inclusive.

The top and bottom 15% of the core are excluded from the evaluation because of the low probability that these regions would be more limiting in the safety analyses and because of the difficulty of making a precise measurement in these regions.

This Surveillance has been modified by a Note that may require that more frequent surveillances be performed. If $F_Q^W(Z)$ is evaluated and found to be within its limit, an evaluation of the expression below is required to account for any increase to $F_Q^W(Z)$ that may occur and cause the $F_Q(Z)$ limit to be exceeded before the next required $F_Q(Z)$ evaluation.

If the two most recent $F_Q(Z)$ evaluations show an increase in the expression

$$\text{maximum over } z \left[\frac{F_Q^C(Z)}{K(Z)} \right].$$

it is required to meet the $F_Q(Z)$ limit with the last $F_Q^W(Z)$ increased by a factor of [1.02], or to evaluate $F_Q(Z)$ more frequently, each 7 EFPD. These alternative requirements prevent $F_Q(Z)$ from exceeding its limit for any significant period of time without detection.

(continued)

Supplement 5

INSERT IB3.2.3-1

Part B of the LCO is modified by a Note (Note 2) that describes the relationship of Allowable Power Level (APL) to RTP as a function of the heat flux hot channel factor at RTP, $F_Q^{RTP}(Z)$. The reactor core AFD is analyzed to 100% RTP or 100% APL, whichever is less. When $F_Q^V(Z)$ is less than its limits, 100% RTP is more limiting than 100% APL. When $F_Q^V(Z)$ is greater than its limits, 100% APL is more limiting than 100% RTP. Hence the APL results in a more restrictive operating envelope for AFD when $F_Q^V(Z)$ is greater than its limits. The K(Z) function is specified in the COLR. $F_Q(Z)$ is defined in the Bases of LCO 3.2.1, "Heat Flux Hot Channel Factor ($F_Q(Z)$)."

1 In the conversion of the HBRSEP current Technical Specifications (CTS) to the proposed plant specific Improved Technical Specifications (ITS), certain bases sections of NUREG-1431, "Improved Standardized Technical Specifications - Westinghouse Plants," Revision 1, (ISTS) are not incorporated into ITS because they are not applicable. Specifically, the bases to ISTS Section 3.2.1A, "Heat Flux Hot Channel Factor (F_Q (Z)) (F_{xy} Methodology), " and Section 3.2.3B, "Axial Flux Difference (AFD) (Relaxed Axial Offset Control (RAOC) Methodology), are not applicable to the H. B. Robinson Steam Electric Plant (HBRSEP), Unit No 2, and are not included in the ITS bases.

2 In the conversion of the HBRSEP current Technical Specifications (CTS) to the proposed plant specific Improved Technical Specifications (ITS), certain wording preferences or conventions are adopted in the bases which do not result in technical changes (either actual or interpretational). Editorial changes or clarifications which involve the insertion of plant specific terms, parameters, or descriptions are used to preserve consistency with the CTS and licensing basis.

3 Bases 3.2.1 and 3.2.3 are modified to reflect the PDC-3 axial offset control methodology utilized for fuel manufactured by Siemens Power Corporation.

4 Bases 3.2.1 are modified to replace the variable expression $F_Q^W(Z)$ with $F_Q^V(Z)$, to describe the relationship of $F_Q^V(Z)$ to $F_Q^C(Z)$ and $F_Q(Z)$, and to provide other clarifications throughout in order to reflect the PDC-3 axial offset control methodology.

5 Bases 3.2.1 are modified to add a paragraph to describe the addition of ITS 3.2.1 Required Action A.1, with subsequent renumbering, and to change the required Completion Time for Required Action A.2 to 30 minutes in order to maintain consistency with Required Action A.1.

6 Bases 3.2.1 are modified to add the Overtemperature ΔT (OTΔT) trip function to be consistent with the addition of the OTΔT function to Required Action A.3 in the ITS.

7 Bases 3.2.1 are modified to delete ISTS 3.2.1 Required Action B.1, with subsequent renumbering, and ISTS SR 3.2.1.1, with subsequent renumbering, to be consistent with the ITS and the PDC-3 axial offset control methodology.

8 Bases 3.2.1 are modified to reflect the PDC-3 axial offset control methodology, which applies only to the middle 80% of the core.

BASES

LCO (continued)

The expression for $F_Q^V(Z)$ is:

$$F_Q^V(Z) = F_Q^C(Z)\ V(Z)$$

where V(Z) is a cycle dependent function that accounts for power distribution transients encountered during normal operation. The V(Z) axial variation function is included in the COLR.

The $F_Q(Z)$ limits define limiting values for core power peaking that precludes peak cladding temperatures above 2200°F during either a large or small break LOCA. $F_Q(Z)$ is the measured $F_Q^M(Z)$ multiplied by the engineering factor F_Q^E (1.03) and the measurement uncertainty factor F_U^M (1.05) at the time of target flux determination from a power distribution map using the movable incore detectors. The PDC-3 axial offset control methodology provides two distinct target bands for operation which are the ±3% and the ±5% target bands. The target band that is selected determines the V(Z) penalty to be applied in the calculation of $F_Q^V(Z)$. Redefinition of the target band from more restrictive to less restrictive ranges between determinations of the target AFD is allowed when appropriate redefinitions of Allowable Power Level (APL) are made. Redefinition of the target band from less restrictive to more restrictive ranges is allowed only in conjunction with the determination of a new target AFD. AFD and APL requirements are discussed in the Bases of LCO 3.2.3.

This LCO requires operation within the bounds assumed in the safety analyses. Calculations are performed in the core design process to confirm that the core can be controlled in such a manner during operation that it can stay within the LOCA $F_Q(Z)$ limits. If $F_Q(Z)$ cannot be maintained within the LCO limits, reduction of the core power is required.

Violating the LCO limits for $F_Q(Z)$ produces unacceptable consequences if a design basis event occurs while $F_Q(Z)$ is outside its specified limits.

APPLICABILITY

The $F_Q(Z)$ limits must be maintained in MODE 1 to prevent core power distributions from exceeding the limits assumed in the safety analyses. Applicability in other MODES is not required because there is either insufficient stored energy

(continued)

BASES

APPLICABILITY (continued)

in the fuel or insufficient energy being transferred to the reactor coolant to require a limit on the distribution of core power.

ACTIONS

A.1

When operation is restricted to the ±3% target band, the V(Z) penalty is minimized, and the $F_Q^V(Z)$ is reduced. Thus when operation is restricted to the more restrictive target band, the result may be that $F_Q^V(Z)$ is within limits, and no reduction in THERMAL POWER is required. In the event that the reduced target band does not result in an acceptable $F_Q^V(Z)$, the THERMAL POWER will be reduced in accordance with Required Action A.2.1. The Completion Time of 15 minutes provides an acceptable time to reevaluate $F_Q^V(Z)$ within the more restrictive target band to determine if $F_Q^V(Z)$ remains within limits.

A.2.1

Reducing THERMAL POWER by ≥ 1% RTP for each 1% by which $F_Q^V(Z)$ exceeds its limit, maintains an acceptable absolute power density. $F_Q^V(Z)$ is $F_Q^M(Z)$ multiplied by engineering uncertainty factors and the maneuvering penalty factor V(Z) as stated in the COLR. $F_Q^M(Z)$ is the measured value of $F_Q(Z)$. The Completion Time of 30 minutes provides an acceptable time to reduce power in an orderly manner and without allowing the plant to remain in an unacceptable condition for an extended period of time.

A.2.2

A reduction of the Power Range Neutron Flux–High trip setpoints by ≥ 1% for each 1% by which $F_Q^V(Z)$ exceeds its limit, is a conservative action for protection against the consequences of severe transients with unanalyzed power distributions. The Completion Time of 72 hours is sufficient considering the small likelihood of a severe transient in this time period and the preceding prompt reduction in THERMAL POWER in accordance with Required Action A.1.

(continued)

BASES

ACTIONS (continued)

A.2.3

Reduction in the Overpower and Overtemperature ΔT trip setpoints by ≥ 1% for each 1% by which $F_Q^V(Z)$ exceeds its limit, is a conservative action for protection against the consequences of severe transients with unanalyzed power distributions. The Completion Time of 72 hours is sufficient considering the small likelihood of a severe transient in this time period, and the preceding prompt reduction in THERMAL POWER in accordance with Required Action A.1.

A.2.4

Verification that $F_Q^V(Z)$ has been restored to within its limit, by performing SR 3.2.1.1 prior to increasing THERMAL POWER above the limit imposed by Required Action A.2.1, ensures that core conditions during operation at higher power levels are consistent with safety analyses assumptions.

B.1

If Required Actions of Condition A are not met within their associated Completion Times, the plant must be placed in a mode or condition in which the LCO requirements are not applicable. This is done by placing the plant in at least MODE 2 within 6 hours.

This allowed Completion Time is reasonable based on operating experience regarding the amount of time it takes to reach MODE 2 from full power operation in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.2.1.1 is modified by a Note. The Note applies during the first power ascension after a refueling. It states that THERMAL POWER may be increased until an equilibrium power level has been achieved at which a power distribution map can be obtained. This allowance is modified, however, by one of the Frequency conditions that requires verification that $F_Q^V(Z)$ is within specified limits after a power rise of more than 10% RTP over the THERMAL POWER at which it was

(continued)

BASES

SURVEILLANCE REQUIREMENTS (continued)

last verified to be within specified limits. Because $F_Q^V(Z)$ could not have previously been measured in this reload core, there is a second Frequency condition, applicable only for reload cores, that requires determination of these parameters before exceeding 75% RTP. This ensures that some determination of $F_Q^V(Z)$ is made at a lower power level at which adequate margin is available before going to 100% RTP. Also, this Frequency condition, together with the Frequency condition requiring verification of $F_Q^V(Z)$ following a power increase of more than 10%, ensures that they are verified as soon as RTP (or any other level for extended operation) is achieved. In the absence of these Frequency conditions, it is possible to increase power to RTP and operate for 31 days without verification of $F_Q^V(Z)$. The Frequency condition is not intended to require verification of these parameters after every 10% increase in power level above the last verification. It only requires verification after a power level is achieved for extended operation that is 10% higher than that power at which F_Q was last measured.

SR 3.2.1.1

The nuclear design process includes calculations performed to determine that the core can be operated within the $F_Q(Z)$ limits. Because flux maps are taken in steady state conditions, the variations in power distribution resulting from normal operational maneuvers are not present in the flux map data. These variations are, however, conservatively calculated by considering a wide range of unit maneuvers in normal operation. The maximum peaking factor increase over steady state values, calculated as a function of core elevation, Z, is called V(Z). Multiplying the measured total peaking factor, $F_Q^C(Z)$, by V(Z) gives the maximum $F_Q(Z)$ calculated to occur in normal operation, $F_Q^V(Z)$.

The limit with which $F_Q^V(Z)$ is compared varies inversely with power and directly with the function K(Z) provided in the COLR.

The V(Z) curve is provided in the COLR for discrete core elevations. Flux map data are typically taken for 30 to 75 core elevations. $F_Q^V(Z)$ evaluations are not applicable for the following axial core regions, measured in percent of core height:

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.2.1.1 (continued)

a. Lower core region, from 0 to 10% inclusive; and

b. Upper core region, from 90 to 100% inclusive.

The top and bottom 10% of the core are excluded from the evaluation because of the low probability that these regions would be more limiting in the safety analyses and because of the difficulty of making a precise measurement in these regions.

Performing the Surveillance in MODE 1 prior to exceeding 75% RTP ensures that the $F_Q(Z)$ limit is met when RTP is achieved, because peaking factors are generally decreased as power level is increased.

$F_Q(Z)$ is verified at power levels ≥ 10% RTP above the THERMAL POWER of its last verification, 12 hours after achieving equilibrium conditions to ensure that $F_Q(Z)$ is within its limit at higher power levels.

The Surveillance Frequency of 31 EFPD is adequate to monitor the change of power distribution with core burnup. The Surveillance may be done more frequently if required by the results of $F_Q(Z)$ evaluations.

The Frequency of 31 EFPD is adequate to monitor the change of power distribution because such a change is sufficiently slow, when the plant is operated in accordance with the TS, to preclude adverse peaking factors between 31 day surveillances.

REFERENCES

1. 10 CFR 50.46, 1974.

2. UFSAR Section 4.4.2.1.

3. UFSAR Section 15.4.8.

4. UFSAR Section 3.1.

BASES

LCO (continued)

Part B of the LCO is modified by a Note (Note 2) that describes the relationship of Allowable Power Level (APL) to RTP as a function of the heat flux hot channel factor at RTP, $F_Q^{RTP}(Z)$. The reactor core AFD is analyzed to 100% RTP or 100% APL, whichever is less. When $F_Q^V(Z)$ is less than its limits, 100% RTP is more limiting than 100% APL. When $F_Q^V(Z)$ is greater than its limits, 100% APL is more limiting than 100% RTP. Hence the APL results in a more restrictive operating envelope for AFD when $F_Q^V(Z)$ is greater than its limits. The K(Z) function is specified in the COLR. $F_Q^V(Z)$ is defined in the Bases of LCO 3.2.1, "Heat Flux Hot Channel Factor ($F_Q^V(Z)$)."

For THERMAL POWER levels > 15% RTP and < 50% RTP (i.e., Part C of this LCO), deviations of the AFD outside of the target band are less significant. The accumulation of 1/2 minute penalty deviation time per 1 minute of actual time outside the target band reflects this reduced significance. With THERMAL POWER < 15% RTP, AFD is not a significant parameter in the assumptions used in the safety analysis and, therefore, requires no limits. Because the xenon distribution produced at THERMAL POWER levels less than 50% RTP does affect the power distribution as power is increased, unanalyzed xenon and power distribution is prevented by limiting the accumulated penalty deviation time.

The frequency of monitoring the AFD by the unit computer is once per minute providing an essentially continuous accumulation of penalty deviation time that allows the operator to accurately assess the status of the penalty deviation time.

Violating the LCO on the AFD could produce unacceptable consequences if a Condition 2, 3, or 4 event occurs while the AFD is outside its limits.

Figure B 3.2.3-1 shows a typical target band and typical AFD acceptable operation limits.

APPLICABILITY

AFD requirements are applicable in MODE 1 above 15% RTP. Above 50% RTP, the combination of THERMAL POWER and core peaking factors are the core parameters of primary importance in safety analyses (Ref. 3).

(continued)

SUPPLEMENT 5
CONVERSION PACKAGE SECTION 3.3
PAGE INSERTION INSTRUCTIONS

Remove and insert the following pages into Enclosure 11 to Serial: RNP-RA/96-0141.

	Remove Page	Insert Page
a.	Part 1, "Markup of Current Technical Specifications (CTS)"	
	3.5-14, 3.5-15a, 3.5-17, 4.1-8	3.5-14, 3.5-15a, 3.5-17, 4.1-8
b.	Part 2, "Discussion of Changes (DOCs) for CTS Markup"	
	7, 22 & 50	7, 22 & 50
c.	Part 3, " No Significant Hazards Consideration (NSHC), And Basis for Categorical Exclusion from 10 CFR 51.22	
	NA	
d.	Part 4, "Markup of NUREG-4131, Revision 1, Standard Technical Specifications-Westinghouse Plants, (ISTS)"	
	3.3-32, 3.3-33, 3.3-34, 3.3-35, 3.3-36	3.3-32, 3.3-33, 3.3-34, 3.3-35, 3.3-36
	3.3-39a, 3.3-42, 3.3-45	3.3-39a, 3.3-42, 3.3-45
e.	Part 5, "Justification of Differences (JFDs) to ISTS"	
	14	14
f.	Part 6, "Markup of ISTS Bases"	
	B 3.3-136, B 3.3-137a, B 3.3-143	B 3.3-136, B 3.3-137a, B 3.3-143
g.	Part 7, "Justification for Differences (JFDs) to ISTS Bases"	
	NA	
h.	Part 8, "Proposed HBRSEP, Unit No. 2 ITS"	
	3.3-25, 3.3-27, 3.3-31, 3.3-34	3.3-25, 3.3-27, 3.3-31, 3.3-34
i.	Part 9. "Proposed Bases to HBRSEP, Unit No. 2 ITS Bases"	
	B 3.3-104, B 3.3-105, B 3.3-105a	B 3.3-104, B 3.3-105, B 3.3-105a
	B 3.3-110	B 3.3-110
j.	Part 10. "ISTS Generic Changes"	
	NA	

Specification 3.3.2 (A1)

ITS

TABLE 3.5-3

ENGINEERED SAFETY FEATURES INSTRUMENTATION LIMITING OPERATING CONDITIONS

(A27)

NO.	FUNCTIONAL UNIT	1 TOTAL NO. OF CHANNELS	2 MINIMUM CHANNELS OPERABLE	3 OPERABLE ACTION IF COLUMN 1 OR 2 CANNOT BE MET	APPLICABLE CONDITIONS
	1. SAFETY INJECTION				MODE 1,2,3,4
[T3.3.2-1(1a)]	A. Manual	2	2	ACTION 11 (B)	≥200°F
[T3.3.2-1(1c)]	B. High Containment Pressure (Hi Level)	3	2	ACTION 12 (E)	≥200°F
[T3.3.2-1(1d)]	C. High Differential Pressure between Any Steam Line and the Steam Header	3/Steam Line	2/Steam Line	ACTION 12 (D)	MODE 1,2,3(a) #
[T3.3.2-1(1d)]	D. Pressurizer Low Pressure	3	2	ACTION 12 (D)	MODES 1,2,3(a) #
[T3.3.2-1(1f)]	E. High Steam Flow in 2/3 Steam Lines Coincident with Low T_{avg} in 2/3 loops	2/Steam Line and 1 T_{avg} Loop	1/Steam Line and 1 T_{avg} in 2 Loops OR 2/Steam Line and 1 T_{avg}	ACTION 12 (D)	≥350°F ## (A11) MODES 1,2(b),3(b)
[T3.3.2-1(1g)]	F. High Steam Flow in 2/3 Steam Lines Coincident with Low Steam Pressure in 2/3 lines	2/Steam Line and 1 Press/Line	1/Steam Line and 1 Press in 2 Lines OR 2/Steam Line and 1 Press	ACTION 12 (E)	≥350°F ##
	2. CONTAINMENT SPRAY				(M24) MODES 1,2,3,4
[T3.3.2-1(3a)]	A. Manual	2	2	ACTION 13 (I)	≥200°F
[T3.3.2-1(3c)]	B. High Containment Pressure (Hi Hi Level)	3/Set	2/Set	ACTION 12 (E)	≥200°F

Amendment No. ~~85~~, 175

Supplement 5

Specification 3.3.2

A1

TABLE 3.5-3 (Continued)

ITS

ENGINEERED SAFETY FEATURES INSTRUMENTATION LIMITING OPERATING CONDITIONS

TABLE NOTATIONS

[T3.3.2-1 NOTE A] # Above Low Pressure SI Block Permit interlock.

[T3.3.2-1 Note(b)] ## Trip function may be blocked below Low T_{avg} Interlock setpoint.

The reactor may remain critical below the Power Operating conditions with this feature inhibited for the purpose of starting reactor coolant pumps.

See 3.3.5

[ACTION B] ACTION 11 With the number of OPERABLE channels one less than the Minimum Channels OPERABLE requirement, restore the inoperable channel to OPERABLE status within 48 hours or be in at least the Hot Shutdown Condition within the next 8 hours and the Cold Shutdown Condition within the following 30 hours. 6

M22

[ACTION C,G] ACTION 12 With the number of OPERABLE channels one less than the Total Number of Channels. Power Operation may proceed ~~until performance of the next required operational test~~ provided the inoperable channel is placed into the tripped condition within 1 hour. 6

or restore OPERABLE in 6 hours

L21

[ACTION D ACTION E]

[ACTION I] ACTION 13 With the number of OPERABLE channels one less than the Minimum Channels OPERABLE requirement, restore the inoperable channel to OPERABLE status within 1 hour or be in ~~at least the Hot Shutdown Condition within the next 8 hours~~ and the ~~Cold Shutdown Condition within the following 30 hours~~.

M24

MODE 3 in 7 hrs, MODE 4 in 13 hrs, MODE 5 in 37 hrs

ACTION 14 With the number of OPERABLE channels one less than the Total Number of Channels; place the inoperable channel into the blocked condition within 1 hour, and restore the inoperable channel to OPERABLE status within 48 hours or be in at least the Hot Shutdown Condition within the next 8 hours and the Cold Shutdown Condition within the following 30 hours.

See 3.3.5

M23

[ACTION C] or be in MODE 3 in 12 hours and MODE 5 in 42 hours

[ACTION D,G] or be in MODE 3 in 12 hours and MODE 4 in 18 hours

[ACTION E] or be in MODE 3 in 12 hours, MODE 4 in 18 hours and MODE 5 in 42 hours

A5

Add ACTIONS "Note1"

L50

Add ACTIONS Note2

Amendment No. 175

Supplement 5

A1

TABLE 3.5-4 (Continued)

ISOLATION FUNCTIONS INSTRUMENTATION LIMITING OPERATING CONDITIONS

ITS

A27

NO.	FUNCTIONAL UNIT	1 TOTAL NO. OF CHANNELS	2 ~~MINIMUM CHANNELS OPERABLE~~	3 OPERABLE ACTION IF COLUMN 1 OR 2 CANNOT BE MET	APPLICABLE CONDITIONS
2.	STEAM LINE ISOLATION				
[T3.3.2-1 (4d)] A.	High Steam Flow in 2/3 Steam Lines Coincident with Low T_{avg} in 2/3 loops	See Item No. 1.E of Table 3.5-3 for initiating functions and requirements			
[T3.3.2-1 (4e)] B.	High Steam Flow in 2/3 Steam Lines Coincident with Low Steam Pressure in 2/3 lines	See Item No. 1.F of Table 3.5-3 for initiating functions and requirements			
[T3.3.2-1 (4c)] C.	High Containment Pressure (Hi Hi Level)	See Item No. 2.B of Table 3.5-3 for initiating functions and requirements			
[T3.3.2-1 (4a)] D.	Manual	1/Line	~~1/Line~~	ACTION ~~16~~	
3.	FEEDWATER LINE ISOLATION				
[T3.3.2-1 (5)] A.	Safety Injection	See Item No. 1 of Table 3.5-3 for all Safety Injection initiating functions and requirements			

L42 Add Note (e) to Applicability

A1 ACTION D

L42 Add Note (e) to MODE 2 and 3 APPLICABILITY

L49 Delete MODE 4

A27 M25 F ACTION ~~16~~ ≥350°F

MODES 1, 2(e), 3(e) L42

L50 Add Note 2 to Surveillance Requirements

M27 Add ACTION H

SR 3.3.2.1 SR 3.3.3.5

SR 3.3.2.3 SR 3.3.2.7

SR 3.3.2.4

T 3.3.2-1 Item 6

M12 Add T 3.3.2-1 "Allowable Value" column

TABLE 4.1-1 (Continued)

MINIMUM FREQUENCIES FOR CHECKS, CALIBRATIONS AND TEST OF INSTRUMENT CHANNELS

	Channel Description	Check	Calibration	Test	Remarks
32.	Loss of Power				See 3.3.5
	a. 480 Emerg. Bus Undervoltage (Loss of Voltage)	N.A.	R	R	
	b. 480 Emerg. Bus Undervoltage (Degraded Voltage)	N.A.	R	R	
33.	Auxiliary Feedwater Flow**** Indication	M [SR 3.3.3.1]	R [SR 3.3.3.2]	N.A.	
34.	Reactor Coolant System** Subcooling Monitor	M	R	N.A.	R1
35.	PORV Position Indicator***	N.A.	N.A.	R [SR 3.3.3.3]	A39
36.	PORV Blocking Valve*** Position Indicator	N.A.	N.A.	R [SR 3.3.3.3]	
37.	Safety Relief Valve Position*** Indicator	N.A.	N.A.	R [SR 3.3.3.3]	L27
38.	Noble Gas Effluent Monitors*****				
	a. Main Steam Line	D	R	Q	R1

** Instrument for Detection of Inadequate Core Cooling - NUREG 0578 Item 2.1.3.b.
*** Direct Indication of Power Operated Relief Valve and Safety Valve Position - NUREG 0578 Item 2.1.3.a.
**** Auxiliary Feedwater Flow Indication to Steam Generator - NUREG 0578 Item 2.1.7.b.
***** Noble Gas Effluent Monitors - NUREG-0737 Item II.F.1.1.

A19

Supplement 5

Specification 3.3.3

A36 CTS Table 4.1-1, Items 1, 2, and 3 (Nuclear Power Range, Nuclear Intermediate Range, and Nuclear Source Range instrumentation), requires testing of the channels to be performed. ITS SR 3.3.1.8 applies to this same instrumentation and is modified by a Note which states that this CHANNEL OPERATIONAL TEST shall include verification that interlocks P-6 and P-10 are in their required state for existing unit conditions. Explicitly adding this Note in the ITS is consistent with current interpretation of the testing requirements and the definition of OPERABILITY as it relates to these instruments. Therefore, this change is administrative.

A37 CTS Table 4.1-1, Items 8, 22, 29, 46, and 47 (4 kV Voltage, Turbine Trip Logic, 4 kV Frequency, Manual Reactor Trip and Reactor Trip Bypass Breakers), requires testing of the channels to be performed. ITS SR 3.3.1.9, SR 3.3.1.14, and SR 3.3.1.15 apply to this instrumentation, as applicable, and are modified by Notes which state that verification of setpoint is not required during these TRIP ACTUATING DEVICE OPERATIONAL TESTS. Explicitly adding these Notes in the ITS is consistent with current interpretation of these testing requirements. Therefore, this change is administrative.

A38 CTS Table 4.1-1, Items 4, 5, 6, 7, 8, 11, 25, 29, 39, and 40 (Reactor Coolant Temperature, Reactor Coolant Flow, Pressurizer Water Level, Pressurizer Pressure, 4 kV Voltage, Steam Generator Level, Turbine First Stage Pressure, 4 kV Frequency, Steam/Feedwater Flow Mismatch and Low Steam Generator Water Level), requires calibration of the channels to be performed. ITS SR 3.3.1.10 and SR 3.3.1.12 apply to this instrumentation, as applicable, and are modified by Notes which state that verification of that time constants are set to required values during these CHANNEL CALIBRATIONS. Explicitly adding these Notes in the ITS is consistent with current interpretation of these testing requirements. Therefore, this change is administrative.

A39 CTS Table 4.1-1, Items 35, 36, and 37 (PORV Position Indicator, PORV Blocking Valve Position Indicator, and Safety Relief Valve Position Indicator) require testing of the valve position indicators to be performed. The current interpretation of this testing requirement, reflected in procedures that implement the testing requirement, is to verify that the valve position indication is in agreement with the actual position of the associated valve. Therefore, the appropriate surveillance for these Functions is the performance of a TADOT (ITS SR 3.3.3.3) since it will verify that the affected valve position indication agrees with the actual position of the associated valve. Setpoint verification is excluded from the TADOT since the affected valve position indication Functions have no associated setpoints. Since this change is consistent with current interpretation of the testing requirement, the change is considered to be administrative.

- Provide information regarding the release of radioactive materials to allow for early indication of the need to initiate action necessary to protect the public, and to estimate the magnitude of any impending threat.

These key variables are identified by the HBRSEP Regulatory Guide 1.97 analyses. These analyses identify the unit specific Type A and Category I variables and provide justification for deviating from the NRC proposed list of Category I variables. In addition, ACTIONS and Surveillance Requirements (including ITS SR 3.3.3.3) are provided for each of the added Functions. Since no similar Specifications or requirements exist in the CTS, this change imposes new requirements and is therefore more restrictive and has no adverse impact on safety.

M33 CTS Table 3.5-5, Note 8 requires that at least one thermocouple be restored to OPERABLE status within a specified time, or be in Hot Shutdown within the next 12 hours and < 350°F within the next 30 hours. ITS Specification 3.3.3 Required Action G requires that, under those circumstances, the unit be placed in MODE 3 in 6 hours, and in MODE 4 in 12 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required unit conditions from full power conditions in an orderly manner and without challenging unit systems. This change imposes shorter Completion Times, and is therefore more restrictive and has no adverse impact on safety.

M34 Not used.

M35 The CTS is revised to adopt ITS Specification 3.3.4, "Remote Shutdown System," in the ITS. The CTS is revised to adopt ITS Specification 3.3.4, "Remote Shutdown System," in the ITS. The specification for the remote shutdown control and instrumentation functions ensures there is sufficient information available on selected unit parameters to place and maintain the unit in MODE 3 should the control room become inaccessible. The Functions in ITS Table 3.3.4-1, Remote Shutdown System Instrumentation and Controls,

OPERABILITY of the Steam Line Isolation Functions when all MSIVS are closed (ITS Table 3.3.2-1 Note (e)). The Steam Line Isolation Functions are provided to isolate the steam lines to provide protection in the event of a Steam Line Break, inside or outside containment. With the MSIVs closed, the function of the instrumentation is satisfied. As a result, with all MSIVs closed, the Function is not required to isolate the steam lines to provide protection in the event of a Steam Line Break, inside or outside containment. In addition, the opening of these valves is a controlled plant evolution which is performed in accordance with administrative controls.

L43 ITS Table 3.3.2-1 Note (f) is added to Function 5.a, Feedwater Isolation - Automatic Actuation Logic and Actuation Relays Function. (The addition of ITS Table 3.3.2-1 Function 5.a is addressed in Discussion of Change L13.) Note (f) allows Function 5.a to not be OPERABLE when the MFIVs, MFRVs, and bypass valves are closed or isolated by a closed manual valve. The Feedwater Isolation Functions are provided to isolate the feedwater lines to mitigate the effects of overfeeding the Steam Generators (SGs) which could result in excessive cooldown of the primary system. With the MFIVs, MFRVs, and bypass valves closed or isolated by a closed manual valve, the function of the instrumentation is satisfied. As a result, with all MFIVs, MFRVs, and bypass valves closed or isolated by a closed manual valve, the Function is not required to isolate the feedwater lines to mitigate the effects of overfeeding the SGs. In addition, the opening of these valves is a controlled plant evolution which is performed in accordance with administrative controls.

L44 CTS Table 3.5-5 Note 5 is changed for the condition of two affected Post Accident Monitoring Function channels monitors inoperable. With two monitors inoperable for 7 days, ITS 3.3.3 Required Action H.1 requires initiation of action in accordance with ITS 5.6.6. ITS 5.6.6 requires initiating the alternate method of monitoring. With two affected channels inoperable, CTS Table 3.5-5 Note 5 requires that if an alternate method of monitoring the affected parameter is not available and implemented with both channels inoperable, then one channel must be restored within 7 days or the plant be placed in Hot Shutdown within 7 days and be $\leq$ 350 F within the following 30 hours. Elimination of the shutdown requirements from CTS 3.5-5 Note 5 when two monitors are inoperable and initiation of the alternate method of monitoring is not established within 7 days is considered acceptable based on the relatively low probability of an event requiring PAM instrumentation, the passive function of the instruments. In addition, if the alternate method of monitoring is not established within the time frame established in ITS 5.6.6, this would constitute a failure to comply with ITS 3.3.3 Required Action H.1 and a shutdown in accordance with ITS LCO 3.0.3 would be required.

L45 CTS Table 4.1-1 Item 32 requires a Channel Functional Test of the Loss of Voltage and Degraded Voltage Instrumentation to be performed once per

Table 3.3.2-1 (page 1 of 8)
Engineered Safety Feature Actuation System Instrumentation

CTS

	FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS	CONDITIONS	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE	TRIP SETPOINT
	1. Safety Injection						
[T3.5-3(1.A)]	a. Manual Initiation	1,2,3,4	2	B	SR 3.3.2.8	NA	NA
[T4.1-1(27)] [4.5.1.1] [M27] [L13]	b. Automatic Actuation Logic and Actuation Relays	1,2,3,4	2 trains	C	SR 3.3.2.2 SR 3.3.2.5 SR 3.3.2.6	NA	NA
[T3.5-1(1)] [T3.5-3(1.B)]	c. Containment Pressure – High	1,2,3	3	E	SR 3.3.2.1 SR 3.3.2.4 SR 3.3.2.7 ~~SR 3.3.2.10~~	≤ 3.86 psig	≤ 3.6 psig
[T3.5-1(3)] [T3.5-3(1.D)]	d. Pressurizer Pressure – Low	1,2,3	3	D	SR 3.3.2.1 SR 3.3.2.4 SR 3.3.2.7 ~~SR 3.3.2.10~~	≥ 1709.89 psig	≥ 1715 psig
	e. Steam Line Pressure						
	(1) Low	1,2,3[(b)]	3 per steam line	D	SR 3.3.2.1 SR 3.3.2.5 SR 3.3.2.9 SR 3.3.2.10	≥ [635] psig (c)	≥ [675] psig (c)
[T3.5-1(4)] [T3.5-3(1.C)]	(2) High Differential Pressure Between Steam Lines (Header and Steam)	1,2,3 (a)	3 per steam line	D	SR 3.3.2.1 SR 3.3.2.4 SR 3.3.2.7 ~~SR 3.3.2.10~~	≤ 106 psig (108.95)	≤ 100 psig
[T3.5-4(2.A)]	f. High Steam Flow in Two Steam Lines	1,2,3 (b)	2 per steam line	D	SR 3.3.2.1 SR 3.3.2.4 SR 3.3.2.7 ~~SR 3.3.2.10~~	≤ 45 (c)	≤ 45 (d)
	Coincident with T_{avg} – Low Low	1,2,3 (b)	1 per loop	D	SR 3.3.2.1 SR 3.3.2.4 SR 3.3.2.7 ~~SR 3.3.2.10~~	≥ 558.6 °F (541.50)	≥ 555 °F (543)

(continued)

(a) Reviewer's Note: Unit specific implementations may contain only Allowable Value depending on Setpoint Study methodology used by the unit.
(b) Above the P-11 (Pressurizer Pressure) interlock.
(c) Time constants used in the lead/lag controller are $t_1 \geq$ [50] seconds and $t_2 \leq$ [5] seconds.
(d) Above the P-12 (T_{avg} – Low Low) interlock.
(e) Less than or equal to a function defined as ΔP corresponding to [44]% full steam flow below [20]% load, and ΔP increasing linearly from [44]% full steam flow at [20]% load to [114]% full steam flow at [100]% load, and ΔP corresponding to [114]% full steam flow above 100% load.
(f) Less than or equal to a function defined as ΔP corresponding to [40]% full steam flow between [0]% and [20]% load and then a ΔP increasing linearly from [40]% steam flow at [20]% load to [110]% full steam flow at [100]% load.

INSERT 3.3.2-4

Table 3.3.2-1 (page 2 of 8)
Engineered Safety Feature Actuation System Instrumentation

CTS

	FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS	CONDITIONS	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE	TRIP SETPOINT
	1. Safety Injection (continued)						
[T3.5-4(2.B)]	g. High Steam Flow in Two Steam Lines	1,2,3	2 per steam line	D	SR 3.3.2.1 SR 3.3.2.5 SR 3.3.2.9 ~~SR 3.3.2.10~~	(c)	(d)
	Coincident with Steam Line Pressure – Low	1,2,3	1 per steam line	D	SR 3.3.2.1 SR 3.3.2.5 SR 3.3.2.9 ~~SR 3.3.2.10~~	≥ 605.05 psig	≥ 614 psig
	2. Containment Spray						
[T3.5-3(2.A)]	a. Manual Initiation	1,2,3,4	2 per train, 2 trains	I	SR 3.3.2.8	NA	NA
[T4.1-1(27)] [4.5.1.3] [M27] [L13]	b. Automatic Actuation Logic and Actuation Relays	1,2,3,4	2 trains	C	SR 3.3.2.2 SR 3.3.2.6 SR 3.3.2.8	NA	NA
[T3.5-1(2)] [T3.5-3(2.B)]	c. Containment Pressure ~~High-3~~ ~~(High High)~~	1,2,3,4	6 (2 sets of 3)	E	SR 3.3.2.1 SR 3.3.2.5 SR 3.3.2.9 ~~SR 3.3.2.10~~	≤ 20.45 psig	≤ 20 psig
	~~High-3 (Two Loop Plants)~~	~~1,2,3~~	~~[3] sets of [2]~~	~~E~~	~~SR 3.3.2.1~~ ~~SR 3.3.2.5~~ ~~SR 3.3.2.9~~ ~~SR 3.3.2.10~~	~~≤ [12.31] psig~~	~~≤ [12.05] psig~~

(continued)

(a) Reviewer's Note: Unit specific implementations may contain only Allowable Value depending on Setpoint Study methodology used by the unit.
(c) Time constants used in the lead/lag controller are t_1 ≥ [50] seconds and t_2 ≤ [5] seconds.
(d) Above the P-12 (T_{avg} – Low Low) interlock.
(e) Less than or equal to a function defined as ΔP corresponding to [44]% full steam flow below [20]% load, and ΔP increasing linearly from [44]% full steam flow at [20]% load to [114]% full steam flow at [100]% load, and ΔP corresponding to [114]% full steam flow above 100% load.
(f) Less than or equal to a function defined as ΔP corresponding to [40]% full steam flow between [0]% and [20]% load and then a ΔP increasing linearly from [40]% steam flow at [20]% load to [110]% full steam flow at [100]% load.

INSERT 3.3.2-4

Supplement 5

Table 3.3.2-1 (page 3 of 8)
Engineered Safety Feature Actuation System Instrumentation

CTS

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS	CONDITIONS	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE	TRIP SETPOINT
3. Containment Isolation						
a. Phase A Isolation						
[T3.5-4(1.A.ii)] [T4.1-3(5)] (1) Manual Initiation	1,2,3,4	2	B	SR 3.3.2.~~8~~ 6	NA	NA
[T3.5-4(1.A.i)] [M27] [L13] (2) Automatic Actuation Logic and Actuation Relays	1,2,3,4	2 trains	C	SR 3.3.2.2 SR 3.3.2.~~6~~ 3 SR 3.3.2.~~8~~ 5	NA	NA
(3) Safety Injection	Refer to Function 1 (Safety Injection) for all initiation functions and requirements.					
[T3.5-4(1.B)] b. Phase B Isolation						
(1) Manual Initiation	1,2,3,4	~~2 per train, 2 trains~~ I	~~B~~	SR 3.3.2.~~8~~ 6	NA	NA
[M27] [L13] (2) Automatic Actuation Logic and Actuation Relays	1,2,3,4	2 trains	C	SR 3.3.2.2 SR 3.3.2.~~6~~ 3 SR 3.3.2.~~8~~ 5	NA	NA
(3) Containment Pressure ~~(High High)~~	1,2,3,4	~~(5)~~ 6 (2 sets of 3)	E	SR 3.3.2.1 4 SR 3.3.2.~~6~~ 7 SR 3.3.2.~~8~~ ~~SR 3.3.2.10~~	≤ ~~[illegible]~~ 20.45 psig	≤ ~~[illegible]~~ 20 psig
4. Steam Line Isolation						
[T3.5-4(2.D)] a. Manual Initiation	1,2,~~3~~ (e)	~~2~~ 1 per steam line	F	SR 3.3.2.~~8~~ 6	NA	NA
[T4.1-1(27)] [M27] [L13] b. Automatic Actuation Logic and Actuation Relays	1,2,~~3~~ (e)	2 trains	G	SR 3.3.2.2 SR 3.3.2.~~6~~ 3 SR 3.3.2.~~8~~ 5	NA	NA

(continued)

(a) Reviewer's Note: Unit specific implementations may contain only Allowable Value depending on Setpoint Study methodology used by the unit.
(i) Except when all MSIVs are closed and [de-activated].

INSERT 3.3.2-4

Supplement 5

Table 3.3.2-1 (page 4 of 8)
Engineered Safety Feature Actuation System Instrumentation

CTS

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS	CONDITIONS	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE	TRIP SETPOINT
4. Steam Line Isolation (continued)						
[T3.5-4(2.c)] c. Containment Pressure – High	1,2, 3	2	D	SR 3.3.2.1 SR 3.3.2.5 SR 3.3.2.9 ~~SR 3.3.2.10~~	≤ 20.45 psig	≤ 20 psig
d. Steam Line Pressure						
(1) Low	1,2[i], 3[b][i]	3 per steam line	D	SR 3.3.2.1 SR 3.3.2.5 SR 3.3.2.9 SR 3.3.2.10	≥ [635][c] psig	≥ [675][c] psig
(2) Negative Rate – High	3[g][i]	3 per steam line	D	SR 3.3.2.1 SR 3.3.2.5 SR 3.3.2.9 SR 3.3.2.10	≤ [121.6][h] psi/sec	≤ [110][h] psi/sec
[T3.5-1(5)] High Steam Flow in Two Steam Lines	1,2, 3	2 per steam line	D	SR 3.3.2.1 SR 3.3.2.5 SR 3.3.2.9 ~~SR 3.3.2.10~~		
[T3.5-3(1.e)] Coincident with T_{avg} – Low Low	1,2, 3	1 per loop	D	SR 3.3.2.1 SR 3.3.2.5 SR 3.3.2.9 ~~SR 3.3.2.10~~	≥ 541.50 °F	≥ 543 °F

(continued)

(a) Reviewer's Note: Unit specific implementations may contain only Allowable Value depending on Setpoint Study methodology used by the unit.
(b) Above the P-11 (Pressurizer Pressure) interlock.
(c) Time constants used in the lead/lag controller are t_1 ≥ [50] seconds and t_2 ≤ [5] seconds.
(d) Above the P-12 (T_{avg} – Low Low) interlock.
(e) Less than or equal to a function defined as ΔP corresponding to [44]% full steam flow below [20]% load, ΔP increasing linearly from [44]% full steam flow at [20]% load to [114]% full steam flow at [100]% load, and ΔP corresponding to [114]% full steam flow above 100% load.
(f) Less than or equal to a function defined as ΔP corresponding to [40]% full steam flow between [0]% and [20]% load and then a ΔP increasing linearly from [40]% steam flow at [20]% load to [110]% full steam flow at [100]% load.
(g) Below the P-11 (Pressurizer Pressure) interlock.
(h) Time constant utilized in the rate/lag controller is ≤ [50] seconds.
(i) Except when all MSIVs are closed and [de-activated].

INSERT 3.3.2-4

Table 3.3.2-1 (page 5 of 8)
Engineered Safety Feature Actuation System Instrumentation

CTS

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS	CONDITIONS	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE	TRIP SETPOINT
4. Steam Line Isolation (continued)						
[T3.5-1(5)] [T3.5-3(4)] e. High Steam Flow in Two Steam Lines	1,2, 3	2 per steam line	D	SR 3.3.2.1 SR 3.3.2.5 SR 3.3.2.9 ~~SR 3.3.2.10~~	(c)	(d)
Coincident with Steam Line Pressure – Low	1,2, 3	1 per steam line	D	SR 3.3.2.1 SR 3.3.2.5 SR 3.3.2.9 ~~SR 3.3.2.10~~	≥ 605.05 psig	≥ 614 psig
~~g. High Steam Flow~~	~~1,2(i), 3(i)~~	~~2 per steam line~~	~~D~~	~~SR 3.3.2.1 SR 3.3.2.5 SR 3.3.2.9 SR 3.3.2.10~~	~~≤ [25]% of full steam flow at no load steam pressure~~	~~≤ [ ] full steam flow at no load steam pressure~~
~~Coincident with Safety Injection~~	~~Refer to Function 1 (Safety Injection) for all initiation functions and requirements.~~					
~~and~~						
~~Coincident with T_{avg} – Low Low~~	~~1,2(i), 3(d)(i)~~	~~[2] per loop~~	~~D~~	~~SR 3.3.2.1 SR 3.3.2.5 SR 3.3.2.9 SR 3.3.2.10~~	~~≥ [550.6]°F~~	~~≥ [553]°F~~
~~h. High High Steam Flow~~	~~1,2(i), 3(i)~~	~~2 per steam line~~	~~D~~	~~SR 3.3.2.1 SR 3.3.2.5 SR 3.3.2.9 SR 3.3.2.10~~	~~≤ [130]% of full steam flow at full load steam pressure~~	~~≤ [ ] of full steam flow at full load steam pressure~~
~~Coincident with Safety Injection~~	~~Refer to Function 1 (Safety Injection) for all initiation functions and requirements.~~					

(continued)

~~(a) Reviewer's Note: Unit specific implementations may contain only Allowable Value depending on Setpoint Study methodology used by the unit.~~
(d) Above the P-12 (T_{avg} – Low Low) interlock.
(i) Except when all MSIVs are closed and [de-activated].

INSERT 3.3.2-4

Supplement 5

ITS Insert 3.3.2-4 (ESFAS Instrumentation)

CTS

(a) Reviewer's Note: Unit specific implementations may contain only Allowable Value depending on Setpoint Study methodology used by the unit. 16

[T3.5-3(#)] (a) (b) Above the P-11 (Pressurizer Pressure) interlock.

(c) Time constants used in the Lead/Lag controller are $t_1 \geq$ [50] seconds and $t_2 \leq$ [5] seconds. 30

[T3.5-3(##)] (d) (b) Above the P-12 (T_{avg} - Low Low) interlock.

(c) (e) Less than or equal to a function defined as ΔP corresponding to [44]% full steam flow below [20]% load, and ΔP increasing linearly from [44]% full steam flow at [20]% load to [114]% full steam flow at [100]% load, and ΔP corresponding to [114]% full steam flow above 100% load. 41.58 110.5

(d) (f) Less than or equal to a function defined as ΔP corresponding to [40]% full steam flow between [0]% and [20]% load and then a ΔP increasing linearly from 37.25 [40]% steam flow at [20]% load to [110]% full steam flow at [100]% load. 37.25 109

(g) Below the P-11 (Pressurizer Pressure) interlock.

(h) Time constant utilized in the rate/lag controller is ≤ [50] seconds. 30

(e) (h) Except when all MSIVs are closed ~~and [deactivated]~~.

(f) (i) Except when all MFIVs, MFRVs, [and associated bypass valves] are closed and [deactivated] [or isolated by a closed manual valve].

Supplement 5

(1)

CTS

SURVEILLANCE REQUIREMENTS

[A 15] NOTE
SR 3.3.3.1 and SR 3.3.3.2 apply to each PAM instrumentation Function in Table 3.3.3-1 except Functions 9, 22, 23, and 24. SR 3.3.3.3 applies only to Functions 9, 22, 23, and 24. (64)

	SURVEILLANCE		FREQUENCY
[T4.1-1]	SR 3.3.3.1	Perform CHANNEL CHECK for each required instrumentation channel that is normally energized.	31 days
[T4.1-1]	SR 3.3.3.2	NOTE Neutron detectors are excluded from CHANNEL CALIBRATION. Perform CHANNEL CALIBRATION.	(18) months
[M32]	SR 3.3.3.3	NOTE Verification of setpoint not required. Perform TADOT.	18 months

(64)

CTS

SURVEILLANCE REQUIREMENTS

	SURVEILLANCE		FREQUENCY
[M35]	SR 3.3.4.1	Perform CHANNEL CHECK for each required instrumentation channel that is normally energized.	31 days
[M35]	SR 3.3.4.2	Verify each required control circuit and transfer switch is capable of performing the intended function.	18 months
[M35]	SR 3.3.4.3	----------NOTE---------- Neutron detectors are excluded from CHANNEL CALIBRATION. ---------- Perform CHANNEL CALIBRATION for each required instrumentation channel.	18 months
[M35]	SR 3.3.4.4	Perform TADOT of the reactor trip breaker open/closed indication.	18 months

Supplement 5

61 ISTS Table 3.3.6-1 Trip Setpoint values for the Containment Radiation Functions are revised to reflect the current licensing basis in CTS Table 3.5-1. This change adds Note (d) to the Trip Setpoint column in place of specific values. Note (d) states Trip Setpoint shall be in accordance with the methodology described in the Offsite Dose Calculation Manual.

62 Not used.

63 ISTS SR 3.3.5.2 requires a TADOT to be performed on the Loss of Power Diesel Generator Start Instrumentation. ITS SR 3.3.5.1 requires a TADOT to be performed and is modified by a Note that excludes verification of the setpoint. Verification of the setpoint requires a bench calibration since at HBRSEP Unit No. 2 relays are used to perform these functions. The change is acceptable since verification of the setpoint is performed during the CHANNEL CALIBRATION (ITS SR 3.3.5.2), which is performed at the same Frequency as the TADOT.

64 ISTS SR 3.3.3.2 requires a CHANNEL CALIBRATION of the Post Accident Monitoring (PAM) Instrumentation, including valve position Functions. Valve position Functions are typically satisfied by valve position indication which is driven by limit switches on the valves. The definition of CHANNEL CALIBRATION cannot be applied to these functions, particularily with respect to sensor inputs and cross calibration. The appropriate surveillance for these Functions is the performance of a TADOT (ITS SR 3.3.3.3) since it will verify that the valve position indication agrees with the actual position of the associated valve. Setpoint verification is excluded from the TADOT since the valve position indication Functions have no associated setpoints. As a result of this change, ITS SR 3.3.3.2 (CHANNEL CALIBRATION) does not apply to the valve position Functions and the Note to the SURVEILLANCE REQUIREMENTS is modified to reflect the application of the SRs to the associated Functions of ITS Table 3.3.3-1. A generic change has been submitted. In addition, the ISTS requirement to perform CHANNEL CHECKS on the valve position Functions is not included in ITS 3.3.3 consistent with the HBRSEP Unit No. 2 current licensing basis reflected in the CTS.

65 ITS Specification 3.3.1 is modified to incorporate a TADOT (i.e., new SR 3.3.1.14) for ITS Function 17.b, "Low Power Reactor Trips Block, P-7," in lieu of SR 3.3.1.11, which requires a channel calibration for that channel. As stated in the Bases to ITS 3.3.1, "Applicable Safety Analyses, LCO, and Applicability," Section b, "Low Power Reactor Trips Block, P-7," the P-7 interlock is a logic function with train and not channel identity. The change was made because the definition of Channel Calibration to include ". . . the adjustment, as necessary, of the channel so that it responds within the required range and accuracy to known input," cannot be met with a logic function. Therefore, a TADOT is the appropriate surveillance requirement. A generic change has been submitted.

BASES

ACTIONS

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

from full power conditions in an orderly manner and without challenging unit systems.

~~G.1~~

INSERT B 3.3.3-8

At this unit, alternate means of monitoring Reactor Vessel Water Level and Containment Area Radiation have been developed and tested. These alternate means may be temporarily installed if the normal PAM channel cannot be restored to OPERABLE status within the allotted time. If these alternate means are used, the Required Action is not to shut down the unit but rather to follow the directions of Specification 5.6.8, in the Administrative Controls section of the TS. The report provided to the NRC should discuss the alternate means used, describe the degree to which the alternate means are equivalent to the installed PAM channels, justify the areas in which they are not equivalent, and provide a schedule for restoring the normal PAM channels.

SURVEILLANCE REQUIREMENTS

A Note has been added to the SR Table to clarify that SR 3.3.3.1 and SR 3.3.3.3 apply to each PAM instrumentation Function in Table 3.3.3-1; except Function 9, Containment Isolation Valve Position; Function 22, PORV Position (Primary); Function 23, PORV Block Valve Position (Primary); and Function 24, Safety Valve Position (Primary). SR 3.3.3.3 applies only to Functions 9, 22, 23, and 24.

SR 3.3.3.1

Performance of the CHANNEL CHECK once every 31 days ensures that a gross instrumentation failure has not occurred. A CHANNEL CHECK is normally a comparison of the parameter indicated on one channel to a similar parameter on other channels. It is based on the assumption that instrument channels monitoring the same parameter should read approximately the same value. Significant deviations between the two instrument channels could be an indication of excessive instrument drift in one of the channels or of something even more serious. A CHANNEL CHECK will detect gross channel failure; thus, it is key to verifying the instrumentation continues to operate properly between each CHANNEL CALIBRATION. The high radiation instrumentation

(continued)

Supplement 5

SR 3.3.3.3

SR 3.3.3.3 is the performance of a TADOT of containment isolation valve position indication, PORV position (primary) indication, PORV block valve position (primary) indication, and safety valve position (primary) indication. This TADOT is performed every 18 months. The test shall independently verify the OPERABILITY of valve position indication against the actual position of the associated valves.

The Frequency is based upon the known reliability of the Functions and has been shown to be acceptable through operating experience.

The SR is modified by a Note that excludes verification of setpoints from the TADOT. The affected Functions have no setpoints.

1

BASES

SURVEILLANCE REQUIREMENTS

SR 3.3.4.3 (continued)

The Frequency of 18 months is based upon operating experience and consistency with the typical industry refueling cycle.

SR 3.3.4.4

SR 3.3.4.4 is the performance of a TADOT every 18 months. This test should verify the OPERABILITY of the reactor trip breakers (RTBs) open and closed indication on the remote shutdown panel, by actuating the RTBs. The Frequency is based upon operating experience and consistency with the typical industry refueling outage.

REFERENCES

1. ~~10 CFR 50, Appendix A, GDC 19.~~

UFSAR, Section 7.4.1

INSERT B TABLE

118
119
120

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Table 3.3.2-1 (page 1 of 4)
Engineered Safety Feature Actuation System Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS	CONDITIONS	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE	TRIP SETPOINT
1. Safety Injection						
a. Manual Initiation	1,2,3,4	2	B	SR 3.3.2.6	NA	NA
b. Automatic Actuation Logic and Actuation Relays	1,2,3,4	2 trains	C	SR 3.3.2.2 SR 3.3.2.3 SR 3.3.2.5	NA	NA
c. Containment Pressure – High	1,2,3,4	3	E	SR 3.3.2.1 SR 3.3.2.4 SR 3.3.2.7	≤ 4.45 psig	≤ 4 psig
d. Pressurizer Pressure – Low	1,2,3[a]	3	D	SR 3.3.2.1 SR 3.3.2.4 SR 3.3.2.7	≥ 1709.89 psig	≥ 1715 psig
e. Steam Line High Differential Pressure Between Steam Header and Steam Lines	1,2,3[a]	3 per steam line	D	SR 3.3.2.1 SR 3.3.2.4 SR 3.3.2.7	≤ 108.95 psig	≤ 100 psig
f. High Steam Flow in Two Steam Lines	1,2[b],3[b]	2 per steam line	D	SR 3.3.2.1 SR 3.3.2.4 SR 3.3.2.7	(c)	(d)
Coincident with T_{avg} – Low	1,2[b],3[b]	1 per loop	D	SR 3.3.2.1 SR 3.3.2.4 SR 3.3.2.7	≥ 541.50 °F	≥ 543°F
g. High Steam Flow in Two Steam Lines	1,2[b],3[b]	2 per steam line	D	SR 3.3.2.1 SR 3.3.2.4 SR 3.3.2.7	(c)	(d)
Coincident with Steam Line Pressure – Low	1,2[b],3[b]	1 per loop	D	SR 3.3.2.1 SR 3.3.2.4 SR 3.3.2.7	≥ 605.05 psig	≥ 614 psig

(continued)

(a) Above the Pressurizer Pressure interlock.
(b) Above the T_{avg}-Low interlock.
(c) Less than or equal to a function defined as ΔP corresponding to 41.58% full steam flow below 20% load, and ΔP increasing linearly from 41.58% full steam flow at 20% load to 110.5% full steam flow at 100% load, and ΔP corresponding to 110.5% full steam flow above 100% load.
(d) Less than or equal to a function defined as ΔP corresponding to 37.25% full steam flow between 0% and 20% load and then a ΔP increasing linearly from 37.25% steam flow at 20% load to 109% full steam flow at 100% load.

Table 3.3.2-1 (page 3 of 4)
Engineered Safety Feature Actuation System Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS	CONDITIONS	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE	TRIP SETPOINT
4. Steam Line Isolation						
a. Manual Initiation	1,2(e),3(e)	1 per steam line	F	SR 3.3.2.6	NA	NA
b. Automatic Actuation Logic and Actuation Relays	1,2(e),3(e)	2 trains	G	SR 3.3.2.2 SR 3.3.2.3 SR 3.3.2.5	NA	NA
c. Containment Pressure – High High	1,2(e),3(e)	6 (2 sets of 3)	D	SR 3.3.2.1 SR 3.3.2.4 SR 3.3.2.7	≤ 20.45 psig	≤ 20 psig
d. High Steam Flow in Two Steam Lines	1,2(e),3(e)	2 per steam line	D	SR 3.3.2.1 SR 3.3.2.4 SR 3.3.2.7	(c)	(d)
Coincident with T_{avg} – Low	1,2(e), 3(e)(b)	1 per loop	D	SR 3.3.2.1 SR 3.3.2.4 SR 3.3.2.7	≥ 541.50 °F	≥ 543°F
e. High Steam Flow in Two Steam Lines	1,2(e),3(e)	2 per steam line	D	SR 3.3.2.1 SR 3.3.2.4 SR 3.3.2.7	(c)	(d)
Coincident with Steam Line Pressure – Low	1,2(e),3(e)	1 per steam line	D	SR 3.3.2.1 SR 3.3.2.4 SR 3.3.2.7	≥ 605.05 psig	≥ 614 psig

(continued)

(b) Above the T_{avg}-Low interlock.
(c) Less than or equal to a function defined as ΔP corresponding to 41.58% full steam flow below 20% load, and ΔP increasing linearly from 41.58% full steam flow at 20% load to 110.5% full steam flow at 100% load, and ΔP corresponding to 110.5% full steam flow above 100% load.
(d) Less than or equal to a function defined as ΔP corresponding to 37.25% full steam flow between 0% and 20% load and then a ΔP increasing linearly from 37.25% steam flow at 20% load to 109% full steam flow at 100% load.
(e) Except when all MSIVs are closed.

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
H. As required by Required Action F.1 and referenced in Table 3.3.3-1.	H.1 Initiate action in accordance with Specification 5.6.6.	Immediately

SURVEILLANCE REQUIREMENTS

-----NOTE-----
SR 3.3.3.1 and SR 3.3.3.2 apply to each PAM instrumentation Function in Table 3.3.3-1; except Functions 9, 22, 23, and 24. SR 3.3.3.3 applies only to Functions 9, 22, 23, and 24.

SURVEILLANCE		FREQUENCY
SR 3.3.3.1	Perform CHANNEL CHECK for each required instrumentation channel that is normally energized.	31 days
SR 3.3.3.2	-----NOTE----- Neutron detectors are excluded from CHANNEL CALIBRATION. ----- Perform CHANNEL CALIBRATION.	18 months
SR 3.3.3.3	-----NOTE----- Verification of setpoint not required. ----- Perform TADOT.	18 months

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.3.4.1	Perform CHANNEL CHECK for each required instrumentation channel that is normally energized.	31 days
SR 3.3.4.2	Verify each required control circuit and transfer switch is capable of performing the intended function.	18 months
SR 3.3.4.3	----------NOTE---------- Neutron detectors are excluded from CHANNEL CALIBRATION. ---------- Perform CHANNEL CALIBRATION for each required instrumentation channel.	18 months
SR 3.3.4.4	Perform TADOT of the reactor trip breaker open/closed indication.	18 months

BASES

ACTIONS (continued)

H.1

Condition H applies to the Containment Sump Water Level, Containment Pressure, Containment Area Radiation, Auxiliary Feedwater Flow, PORV Position, PORV Block Valve Position, and Safety Valve Position Functions, which have alternate monitoring means available for use. These alternate means may be temporarily installed if the normal PAM channel cannot be restored to OPERABLE status within the allotted time. If these alternate means are used, the Required Action is not to shut down the unit but rather to follow the directions of Specification 5.6.6, in the Administrative Controls section of the TS. The report provided to the NRC should discuss the alternate means used, describe the degree to which the alternate means are equivalent to the installed PAM channels, justify the areas in which they are not equivalent, and provide a schedule for restoring the normal PAM channels.

SURVEILLANCE REQUIREMENTS

A Note has been added to the SR Table to clarify that SR 3.3.3.1 and SR 3.3.3.2 apply to each PAM instrumentation Function in Table 3.3.3-1; except Function 9, Containment Isolation Valve Position; Function 22, PORV Position (Primary); Function 23, PORV Block Valve Position (Primary); and Function 24, Safety Valve Position (Primary). SR 3.3.3.3 applies only to Functions 9, 22, 23, and 24.

SR 3.3.3.1

Performance of the CHANNEL CHECK once every 31 days ensures that a gross instrumentation failure has not occurred. A CHANNEL CHECK is normally a comparison of the parameter indicated on one channel to a similar parameter on other channels. It is based on the assumption that instrument channels monitoring the same parameter should read approximately the same value. Significant deviations between the two instrument channels could be an indication of excessive instrument drift in one of the channels or of something even more serious. A CHANNEL CHECK will detect gross channel failure; thus, it is key to verifying the instrumentation continues to operate properly between each CHANNEL CALIBRATION. The high radiation instrumentation

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.3.3.1 (continued)

should be compared to similar unit instruments located throughout the unit.

Channel deviation criteria are determined by the unit staff, based on a combination of the channel instrument uncertainties, including isolation, indication, and readability. If a channel is outside the criteria, it may be an indication that the sensor or the signal processing equipment has drifted outside its limit. If the channels are within the criteria, it is an indication that the channels are OPERABLE.

As specified in the SR, a CHANNEL CHECK is only required for those channels that are normally energized.

The Frequency of 31 days is based on operating experience that demonstrates that channel failure is rare. The CHANNEL CHECK supplements less formal, but more frequent, checks of channels during normal operational use of the displays associated with the LCO required channels.

SR 3.3.3.2

A CHANNEL CALIBRATION is performed every 18 months, or approximately at every refueling. CHANNEL CALIBRATION is a complete check of the instrument loop, including the sensor. The test verifies that the channel responds to measured parameter with the necessary range and accuracy. This SR is modified by a Note that excludes neutron detectors. The calibration method for neutron detectors is specified in the Bases of LCO 3.3.1, "Reactor Protection System (RPS) Instrumentation." The Frequency is based on operating experience and consistency with the typical industry refueling cycle.

SR 3.3.3.3

SR 3.3.3.3 is the performance of a TADOT of containment isolation valve position indication, PORV position (primary) indication, PORV block valve position (primary) indication, and safety valve position (primary) indication. This TADOT is performed every 18 months. The test shall independently

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.3.3.3 (continued)

verify the OPERABILITY of position indication against the actual position of the associated valves.

The Frequency is based upon the known reliability of the Functions and has been shown to be acceptable through operating experience.

The SR is modified by a Note that excludes verification of setpoints from the TADOT. The affected Functions have no setpoints.

REFERENCES

1. NRC Safety Evaluation Report, H. B. Robinson Steam Electric Plant Unit No. 2, Docket No. 50-261, Conformance to Regulatory Guide 1.97, transmitted to CP&L by letter dated March 5, 1987.

2. Regulatory Guide 1.97, Revision 3, May 1983.

3. NUREG-0737, Supplement 1, "TMI Action Items."

4. CP&L Letter to NRC, "Inadequate Core Cooling Instrumentation, Generic Letter 82-28, NUREG-0737, Item II.F.2, Implementation Letter/License Amendment Request," dated September 16, 1987.

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.3.4.2

SR 3.3.4.2 verifies each required Remote Shutdown System control circuit and transfer switch performs the intended function. This verification is performed from the remote shutdown panel and locally, as appropriate. Operation of the equipment from the remote shutdown panel is not necessary. The Surveillance can be satisfied by performance of a continuity check. This will ensure that if the control room becomes inaccessible, the unit can be placed and maintained in MODE 3 from the remote shutdown panel and the local control stations. The 18 month Frequency is based on the need to perform this Surveillance under the conditions that apply during a plant outage and the potential for an unplanned transient if the Surveillance were performed with the reactor at power. (However, this Surveillance is not required to be performed only during a unit outage.) Operating experience demonstrates that remote shutdown control channels usually pass the Surveillance test when performed at the 18 month Frequency.

SR 3.3.4.3

CHANNEL CALIBRATION is a complete check of the instrument loop and the sensor. The test verifies that the channel responds to a measured parameter within the necessary range and accuracy.

The Frequency of 18 months is based upon operating experience and consistency with the typical industry refueling cycle.

SR 3.3.4.4

SR 3.3.4.4 is the performance of a TADOT every 18 months. This test should verify the OPERABILITY of the reactor trip breakers (RTBs) open and closed indication on the remote shutdown panel, by actuating the RTBs. The Frequency is based upon operating experience and consistency with the typical industry refueling outage.

REFERENCES

1. UFSAR, Section 7.4.1.

SUPPLEMENT 5
CONVERSION PACKAGE SECTION 3.4
PAGE INSERTION INSTRUCTIONS

Remove and insert the following pages into Enclosure 12 to Serial: RNP-RA/96-0141.

	Remove Page	Insert Page
a.	Part 1, "Markup of Current Technical Specifications (CTS)"	
	4.1-14, 4.1-11	4.1-14, 4.1-11
b.	Part 2, "Discussion of Changes (DOCs) for CTS Markup"	
	33, 33a, 33b	33 through 38
c.	Part 3, " No Significant Hazards Consideration (NSHC), And Basis for Categorical Exclusion from 10 CFR 51.22	
	22e through 24	22e through 24
d.	Part 4, "Markup of NUREG-4131, Revision 1, Standard Technical Specifications-Westinghouse Plants, (ISTS)"	
	NA	
e.	Part 5, "Justification of Differences (JFDs) to ISTS"	
	NA	
f.	Part 6, "Markup of ISTS Bases"	
	B 3.4-9	B 3.4-9
	Insert B 3.4.3-4 (no page number)	B 3.4-9a
	Insert B 3.4.11-4 (no page number)	B 3.4-57a
	B 3.4-80	B 3.4-80
	Insert B 3.4.14-3 (no page number)	B 3.4-80a
	B 3.4-81, B 3.4-82, B 3.4-83	B 3.4-81, B 3.4-82, B 3.4-83
	Insert B 3.4.14-4 (no page number)	B 3.4-83b
	B 3.4-84, B 3.4-85	B 3.4-84, B 3.4-85
	Insert B 3.4.14-5 (no page number)	B 3.4-85a
	B 3.4-97	B 3.4-97
g.	Part 7, "Justification for Differences (JFDs) to ISTS Bases"	
	1	1
h.	Part 8, "Proposed HBRSEP, Unit No. 2 ITS"	
	NA	

SUPPLEMENT 5
CONVERSION PACKAGE SECTION 3.4
PAGE INSERTION INSTRUCTIONS

Remove and insert the following pages into Enclosure 12 to Serial: RNP-RA/96-0141.

	Remove Page	Insert Page
i.	Part 9. "Proposed Bases to HBRSEP, Unit No. 2 ITS Bases"	
	B 3.4-9 through B 3.4-17	B 3.4-9 through B 3.4-17
	-	B 3.4-17a
	B 3.4-59	B 3.4-59
	B 3.4-82 through B 3.4-87	B 3.4-82 through B 3.4-87
	-	B 3.4-87a & B 3.4-87b
	B 3.4-99, B 3.4-100	B 3.4-99, B 3.4-100
j.	Part 10. "ISTS Generic Changes"	
	NA	

Specification 3.4.14

A1

TABLE 4.1-3 (Continued)
FREQUENCIES FOR EQUIPMENT TESTS

	Check	Frequency	Maximum Time Between Test
		~~2. Whenever integrity of a pressure isolation valve listed in Table 3.1-1 cannot be demonstrated, the integrity of the remaining valve in each high pressure line having a leaking valve shall be determined and recorded daily. In addition, the position of the other closed valve located in the high pressure piping shall be recorded daily.~~	
18. Automatic Bus Transfers a) Auxiliary Feedwater Header Discharge Valve to Steam Generator A, V2-16A b) Turbine Building, Cooling Water Isolation Valve, V6-16C	Test thermal and magnetic trip elements of respective molded case circuit breakers	Each refueling shutdown.	NA

L24

See 3.8.9

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A1

(HBR-28)

ITS

NOTES TO TABLE 4.1-2

(1) A gross activity analysis shall consist of the quantitative measurement of the total radioactivity of the primary coolant of units of μCi/gram.

(2) A radiochemical analysis shall consist of the quantitative measurement of each radionuclide with half life greater than 30 minutes making up at least 95% of the total activity of the primary coolant.

LA6

A27

Add SR 3.4.16.3 Note

(3) When iodine or particulate radioactivity levels exceed 10% of the limit in Specification 3.9.2.1, the sampling frequency shall be increased to a minimum of once each day.

LA10

(5) Deleted.

[SR 3.4.16.3] (6) Sample to be taken after a minimum of 2EFPD and 20 days of power operation have elapsed since the reactor was last subcritical for 48 hours or longer.

[SR 3.4.16.2] [SR 3.4.16.3] [RA A.1] (7) Samples are to be taken in the power operating condition.

(8) Sample taken at all operating conditions whenever the specific activity exceed 1.0 μCi/gram DOSE EQUIVALENT I-131 or 100/E Ci/gram. These samples are to be taken until the specific activity of the reactor coolant system is restored within its limits.

[SR 3.4.16.2] (9) One sample between 2 and 6 hours following a thermal power change exceeding 15 percent of the rated thermal power within a one-hour period. Samples are required when in the hot shutdown or power operating modes.

(10) Sample whenever that gross activity determination indicates iodine concentrations are greater than 10% of the allowable limit.

(11) Sample whenever the gross activity determination indicates iodine concentrations are below 10 percent of the allowable limit.

See 3.7.15

NA* - Not applicable.

3.0 would be required. CTS 3.0 requires the unit to be placed in hot shutdown within 8 hours and to be placed in cold shutdown within the next 30 hours until the reactor is placed in a condition in which the specification is not applicable. For the same condition, ITS Specification 3.4.5 Required Action A.1 provides an allowed outage of time of 72 hours. If the inoperable component is not restored to OPERABLE status within 72 hours, ITS Specification 3.4.5 Required Action B.1 requires that the unit be placed in MODE 4 (which is outside the Applicability of Specification 3.4.5) within 12 hours. The addition of a Required Action Completion Time of 12 hours to reach MODE for is a more restrictive aspect of this change. In this condition, the remaining OPERABLE and operating RCS loop is adequate to provide the decay heat removal function, ensure that boron stratification does not occur. In addition, the requirements imposed by the LCO when only one RCS loop is operating are adequate to ensure a power excursion resulting from an inadvertent control rod withdrawal event is precluded. However, in this condition, a single failure may place the unit in a condition where adequate decay heat removal and proper mixing of the reactor coolant may not be available. Therefore, an allowable outage time of 72 hours is provided; after which the unit must be placed in MODE 4 within the next 12 hours. These time periods ensure the risk associated with unit operation in this condition is minimized while providing an allowance to attempt restoration prior to subjecting the unit to a cooldown transient. This change is consistent with NUREG-1431.

L19 CTS Specification 3.1.1.1.a allows the number of operating reactor coolant pumps to be reduced provided certain actions are taken. These actions ensure that a power excursion resulting from a inadvertent control rod withdrawal event is precluded. CTS Specification 3.1.1.1.a does not explicitly provide a time period for implementing these requirements in the event of a loss of an operating reactor coolant pump. ITS Specification 3.4.5 Required Action C.1 provides an allowable outage time of 1 hour to comply with the requirements of the LCO. This time period is adequate to be restore compliance with the LCO without exposing the unit to risk for an undue period of time. In addition, this time period is consistent with the 1 hour time provided in ITS LCO 3.0.3 before requiring the unit to be placed in a non-applicable MODE.

L20 The CTS is modified by the addition of ITS LCO 3.4.17 Required Actions D.1, D.2, and D.3 to require that in the event that seal injection to any RCP is not within limits and both required charging pumps are inoperable, the plant be cooled down and depressurized to an RCS pressure < 1400 psig. No comparable action is contained in CTS, and in such a condition, entry into CTS 3.0 would be required, which requires that the plant be placed in hot shutdown within 8 hours and in cold shutdown within an additional 30 hours. The ITS 3.4.17 Required Actions associated with ITS 3.4.17 Condition D are a relaxation of requirements and is more appropriate than requiring entry into cold shutdown. If ITS Condition D were entered, seal injection to the RCPs is not assured. Cooling of the RCPs is only available from the component cooling system, and if the component cooling system were lost, RCP seal failure would eventially occur if seal injection or component cooling were not restored. When no charging capability is available, the RCS will lose RCS inventory through the RCP seals. With no operable means of RCP seal injection, it would be imprudent to require the plant to go to MODE 4, where a requirement for RCP seal injection remains and shutdown margin requirements would be difficult to maintain. Therefore, the appropriate action is to initiate measures to restore RCP seal injection immediately and to continue the action to cool down and depressurize to an RCS pressure less than 1400 psig to allow makeup to the RCS through the Safety Injection (SI) System. The Completion Time of 12 hours is reasonable based on operating experience to allow an orderly transition between MODES 3 and MODE 4, which is the closest condition corresponding to depressurization to an RCS pressure < 1400 psig, without challenging plant systems. Maintaining the plant in MODE 3 with the RCS pressure < 1400 psig until charging is reestablished to the RCPs is reasonable to avoid further challenging plant systems in this condition.

L21 CTS Specification 3.2.2.d requires that system piping, instrumentation, controls, and valves shall be operable to the extent of establishing one flow path from the BASTs and one flow path from the RWST to the RCS. This requirement is modified in ITS LCO 3.4.17 as the requirement that two Makeup Water Pathways from the RWST shall be OPERABLE. The ITS provides more operational flexibility and is less restrictive because the BASTs are not specified to be a pathway source. There are two pathways available from the RWST to the charging pump suction header, any one of which provides an equivalent source of makeup water for RCP seal injection. The Operability requirement for ITS Specifications 3.4.17 is to maintain sufficient seal water injection flow to the RCPs. Two pathways provide redundant capability to assure a continuous source of makeup water without specifying each pathway source. Therefore, the increased flexibility in ITS Specifications 3.4.17 is acceptable.

L22 CTS Specification 3.2.3 permits power operation to continue with one of the two operable charging pumps inoperable for up to 24 hours. In ITS Specification 3.4.17, the allowed outage time for a required charging pump is 72 hours. This is a relaxation of allowed outage time

requirements. The 72 hour Completion Time is reasonable, based on the redundant capabilities afforded by the OPERABLE train, and the low probability of a total loss of RCP seal injection occurring during this period. The 72 hour allowed outage time is consistent with other allowed outage times for a single component in NUREG-1431.

L23 CTS Specification 3.2.3.a allows one boric acid transfer pump to be inoperable for up to 24 hours. CTS Specification 3.2.3.c allows one channel of heat tracing to be inoperable for up to 24 hours. The CTS is modified by not including these specific requirements and by adding ITS LCO 3.4.17 Required Action B.1 which allows a Makeup Water Pathway from the RWST to be inoperable for up to 72 hours. This change adds operational flexibility and is less restrictive because the allowable inoperable components in the Makeup Water Pathways are not specified and because there are no longer Required Actions for the boric acid pumps and heat tracing. There are two pathways available from the RWST to the charging pump suction header. These pathways consist of a remotely operated air operated valve and a locally operated manual valve. Either of these pathways provide an equivalent source of makeup water for RCP seal injection. The Operability requirement for ITS Specifications 3.4.17 is to maintain sufficient seal water injection flow to the RCPs. The two pathways provide redundant capability to assure a continuous source of makeup water without specifying each pathway source.

Additionally, other components than those named in CTS Specification 3.2.3 (i.e., valves) may be inoperable in the makeup water pathways that render the pathway inoperable. In such cases the CTS would require entry into CTS Specification 3.0, which requires that hot shutdown be achieved in 8 hours and cold shutdown be achieved within an additional 30 hours. The addition of ITS LCO 3.4.17 Required Action B.1 avoids entry into ITS Specification 3.0.3 for the valves. This change is acceptable because the allowed outage time places an ultimate time requirement that must be met to exit the Condition.

Therefore, the increased flexibility in ITS Specifications 3.4.17 Required Action B.1 is acceptable.

L24 CTS Table 4.1-3 item 17.2 requires, whenever the integrity of a RCS pressure isolation valve cannot be demonstrated, the integrity of the remaining valve to be determined and recorded daily. In this condition, CTS Table 4.1.3 item 17.2 also requires that the position of the other closed valve located in the high pressure piping to be recorded daily. Under this same condition, ITS 3.4.14, RCS Pressure Isolation Valves (PIVs), Required Actions A.1 and A.2 require isolation of the high pressure portion of the piping from the low pressure portion of the piping by the use of two valves. In addition, ITS 3.4.14 Required Actions A.1 and A.2 are modified by a Note that requires the valves used to meet the requirements of Required Actions A.1 and A.2 to satisfy the leakage criteria of SR 3.4.14.1 (i.e., integrity determined to be

acceptable) and that the valves be in the reactor coolant pressure boundary or high pressure portion of the piping.

The normal periodic surveillance frequency (ITS SR 3.4.14.1) for RCS PIV leakage testing provides adequate assurance of PIV OPERABILITY. Therefore, the CTS requirement to perform the surveillance (in order to record the continued integrity of the associated valves used to comply with Required Actions A.1 and A.2) once per day is deleted. If the Surveillance is not performed within the normal surveillance interval, compliance with the requirements of the Note to Required Actions A.1 and A.2 would not be satisfied for these valves and a shutdown per Required Actions C.1 and C.2 would be required (i.e., action taken to exit the Applicability of the LCO). If at any time it is discovered that the valves used to comply with Required Actions A.1 and A.2 did not satisfy the requirements of SR 3.4.14.1, Condition C must be immediately entered and Required Actions C.1 and C.2 taken. In addition, ITS 3.4.14 Required Actions A.1 and A.2 require isolation of the high pressure portion of the piping from the low pressure portion of the piping by the use of two valves versus the CTS requirement to isolate the high pressure portion of the piping by the use of one valve. The additional valve required to provide isolation ensures that a single failure of one of these valves does not impact the ability to prevent overpressurization of low pressure piping. Therefore, the proposed change continues to provide adequate assurance that overpressurization of low pressure piping will not occur.

ITS 3.4.14 Required Actions A.1 and A.2 require isolation of the high pressure portion of the piping from the low pressure portion of the piping by the use of two valves. These requirements provide adequate assurance that the high pressure portion of the piping will remain isolated from the low pressure portion of the piping. Therefore, in the event of an inoperable RCS PIV, the CTS requirement to record the position of the other closed valve located in the high pressure piping daily (i.e., verification of compliance with Technical Specifications Actions) is deleted. This change is considered to be acceptable based on 1) the administrative controls governing valve operation, 2) the low probability of misalignment of these valves, and 3) the fact that ITS 3.4.14 Required Actions A.1 and A.2 require isolation of the high pressure portion of the piping from the low pressure portion of the piping by the use of two valves versus the CTS requirement to isolate the high pressure portion of the piping by the use of one valve. The CTS verification is an implicit part of using Technical Specifications and determining the appropriate Conditions to enter and Actions to take in the event of inoperability of Technical Specification equipment and the failure to comply with Technical Specification Actions. In addition, plant and equipment status is continuously monitored by control room personnel. The results of this monitoring process are documented in records/logs maintained by control room personnel. The continuous monitoring process includes re-evaluating the status of

compliance with Technical Specification requirements, when Technical Specification equipment status changes, using the control room records/logs as aids. Therefore, the explicit requirement to periodically record/verify, in the event of an inoperable RCS PIV, the position of the other closed valve located in the high pressure piping is considered to be unnecessary for ensuring compliance with the applicable Technical Specification Actions.

RELOCATED SPECIFICATIONS

R1	3.1.2.2	Steam Generator Pressure
	3.1.2.3	Pressurizer Heatup and Cooldown
	Table 4.1-2 Item 1	Oxygen and chloride concentration in the RCS

These Specifications, or Limiting Conditions for Operation (Chapter 3.0), are not retained in the ITS because they have been reviewed against, and determined not to satisfy, the selection criteria for Technical Specifications provided in 10 CFR 50.36. The selection criteria were established to ensure that the Technical Specifications are reserved for those conditions or limitations on plant operation considered necessary to limit the possibility of an abnormal situation or event that could result in an immediate threat to the health and safety of the public. The rationale for relocation of each of these Specifications is provided in the report, "Application of Selection Criteria to the H. B. Robinson Steam Electric Plant Unit No. 2 Technical Specifications."

These Limiting Conditions for Operation, and their associated Surveillance Requirements (Chapter 4.0), are relocated to licensee controlled documents. Relocation of the specific requirements for systems or variables contained in these Specifications to licensee documents will have no impact on the operability or maintenance of those systems or variables. The licensee will initially continue to meet the requirements contained in the relocated Specifications. The licensee is allowed to make changes to these requirements in accordance with the provisions of 10 CFR 50.59. Such changes can be made without prior NRC approval, if the change does not involve an unreviewed safety question, as defined in 10 CFR 50.59. These controls are considered adequate for assuring that structures, systems, and components in the relocated Specifications are maintained operable, and variables are maintained within limits. This change is consistent with the NRC Final Policy Statement on Technical Specification Improvements.

components. The probability of an occurrence of an accident, including a loss of RCP seal injection, is increased by the increase in length of allowed outage time for one inoperable pathway. However, continuous operation with one pathway inoperable such that a single failure would preclude the pathways from fulfilling their required function is not allowed. Therefore, the significance of the increase in probability is small. The consequences of an accident occurring during the additional interval permitted in the allowed outage time for one pathway are the same as the consequences during the currently permitted 24 hours for an inoperable pathway. Therefore, the proposed change does not involve a significant increase in the probability or an increase in the 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 or changes in parameters governing normal plant operation. An increase in the allowed outage time for a Makeup Water Pathway does not create the possibility of any new or different kind of accident from any accident previously evaluated. Therefore, the possibility of a new or different kind of accident from any accident previously evaluated is not created.

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

   The 72 hour allowed outage time for an inoperable Makeup Water Pathway is reasonable based on the redundant capabilities afforded by the OPERABLE pathway and the low probability of a loss of RCP seal injection occurring during this time period. However, the overall reliability is reduced because a single failure of the OPERABLE pathway could result in a loss of function. As a result, any reduction in the margin of safety is small and is at least partially offset by a reduction in the risk associated with averted plant shutdowns and associated shutdown transients. Therefore, the proposed change does not involve a significant reduction in a margin of safety.

LESS RESTRICTIVE CHANGES
("L24" Labeled Comments/Discussions)

Carolina Power & Light Company has evaluated the proposed Technical Specification change and has concluded 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 removes an unnecessary additional performance of a Surveillance which has been performed within its normally required Frequency. Not performing the Surveillance will not affect any equipment which is assumed as an initiator of any analyzed event. Furthermore, since the Surveillance continues to be performed on its normal Frequency, there is no impact on the capability of the component to perform its required safety function. This change also deletes the requirement to record/verify the position of a valve closed to provide isolation in accordance with Technical Specification Actions. This verification is not considered in the initiation of any previously analyzed accident. Therefore, this change does not significantly increase the frequency of such accidents. This verification is an implicit part of using Technical Specifications and determining the appropriate Conditions to enter and Actions to take in the event of inoperability of Technical Specification equipment and failure to comply with Technical Specification Actions. In addition, plant and equipment status is continuously monitored by control room personnel. The results of this monitoring process are documented in records/logs maintained by control room personnel. The continuous monitoring process includes re-evaluating the status of compliance with Technical Specification requirements, when Technical Specification equipment status changes, using the control room records/logs as aids. Therefore, the explicit requirement to periodically record/verify, in the event of an inoperable RCS PIV, the position of the other closed valve located in the high pressure piping is considered to be unnecessary for ensuring compliance with the applicable Technical Specification Actions. The status of the plant and equipment will continue to be monitored to assure the potential consequences are not significantly increased. Therefore, this change does not involve an increase in the probability or consequences of an accident previously evaluated.

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

   The proposed change does not involve any physical alteration of plant systems, structures or components, nor does it alter parameters governing normal plant operation. The normal periodic surveillance frequency (ITS SR 3.4.14.1) for RCS PIV leakage testing provides adequate assurance of PIV OPERABILITY. In addition, the status of the plant and equipment will continue to be monitored to assure the possibility for a new or different kind of accident is not created. The proposed change does not introduce a new mode of operation or alter the method of normal plant operation. Therefore, the possibility of a new or different kind of accident from any accident previously evaluated is not created.

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

   The proposed change does not involve a significant reduction in a margin of safety since the normal periodic Frequency is adequate for assuring the requirements of the Actions are maintained. If the Surveillance is not performed within the normal surveillance interval, compliance with the

requirements of the Note to ITS 3.4.14 Required Actions A.1 and A.2 would not be satisfied for these valves and a shutdown per ITS 3.4.14 Required Actions C.1 and C.2 would be required (i.e., action taken to exit the Applicability of the LCO). If at any time it is discovered that the valves used to comply with ITS 3.4.14 Required Actions A.1 and A.2 did not satisfy the requirements of SR 3.4.14.1, ITS 3.4.14 Condition C must be immediately entered and ITS 3.14.4 Required Actions C.1 and C.2 taken. The change associated with deletion of the explicit requirement to periodically verify the position of the valve used to isolate the affected high pressure line is considered to be acceptable based on 1) the administrative controls governing valve operation, 2) the low probability of misalignment of these valves, and 3) the fact that ITS 3.4.14 Required Actions A.1 and A.2 require isolation of the high pressure portion of the piping from the low pressure portion of the piping by the use of two valves versus the CTS requirement to isolate the high pressure portion of the piping by the use of one valve. In addition, the verification of the equipment status is an implicit part of using Technical Specifications and determining the appropriate Conditions to enter and Actions to take in the event of inoperability of Technical Specification equipment and failure to comply with Technical Specification Actions. Plant and equipment status is continuously monitored by control room personnel. The results of this monitoring process are documented in records/logs maintained by control room personnel. The continuous monitoring process includes re-evaluating the status of compliance with Technical Specification requirements, when Technical Specification equipment status changes, using the control room records/logs as aids. Therefore, the explicit requirement to periodically record/verify, in the event of an inoperable RCS PIV, the position of the other closed valve located in the high pressure piping is considered to be unnecessary for ensuring compliance with the applicable Technical Specification Actions. The status of the plant and equipment will continue to be monitored to assure appropriate previously approved actions are taken in the event of failure to comply with Technical Specification Actions. Therefore, this change does not involve a significant reduction in a margin of safety.

RELOCATED CHANGES
("R" Labeled Comments/Discussions)

Relocating Requirements which do not meet the Technical Specification criteria to documents with an established control program allows the Technical Specifications to be reserved only for those conditions or limitations upon reactor operation which are necessary to adequately limit the possibility of an abnormal situation or event giving rise to an immediate threat to the public health and safety, thereby focusing the scope of Technical Specifications.

Therefore, requirements which do not meet the Technical Specification criteria in the NRC Final Policy Statement on Technical Specification Improvement for Nuclear Power Reactors (58 FR 39132, dated 7/22/93) have been relocated to licensee controlled documents. This policy statement addresses the scope and purpose of Technical Specifications. In doing so, it establishes a specific set of objective criteria for determining which regulatory requirements and operating restrictions should be included in Technical Specifications. These criteria are as follows:

Criterion 1: Installed instrumentation that is used to detect and indicate in the control room, a significant abnormal degradation of the reactor coolant pressure boundary;

Criterion 2: A process variable that is an initial condition of a design basis accident (DBA) or transient analyses that either assumes the failure of or presents a challenge to the integrity of a fission product barrier;

Criterion 3: A structure, system or component that is part of the primary success path and which functions or actuates to mitigate a design basis accident or transient that either assumes the failure of or presents a challenge to the integrity of a fission barrier;

Criterion 4: A structure, system or component which operating experience or probabilistic safety assessment has shown to be significant to public health and safety.

The application of these criteria is provided in the "Application of Screening Criteria to the HBRSEP Unit No. 2 Technical Specifications." Requirements which met the criteria have been included in the proposed improved Technical Specifications. Carolina Power & Light (CP&L) proposes to remove the requirements which do not meet the criteria from the Technical Specifications and relocate the requirements to a suitable owner controlled document. The requirements in the relocated Specifications are not affected by this Technical Specification change. CP&L will initially continue to perform the required operation and maintenance to assure that the requirements are satisfied. Relocating specific requirements for systems or variables has no impact on the system's operability or the variable's maintenance, as applicable.

Licensee controlled programs will be utilized as the control mechanism for the relocated Specifications as they will be placed in plant procedures or other licensee controlled documents. CP&L is allowed to make changes to these requirements, without prior NRC approval, if the change does not involve an unreviewed safety question. These controls are considered adequate for assuring structures, systems and components in the relocated Specifications are maintained operable and variables in the relocated Specifications are maintained within limits.

Carolina Power & Light Company has evaluated each of the proposed Technical Specification changes identified as "Relocated" and has concluded that they do 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 proposed changes do 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 relocates requirements and surveillances for structures, systems, components or variables which did not meet the criteria for inclusion in Technical Specifications as identified in the "Application of Selection Criteria to the HBRSEP Unit No. 2 Technical Specifications." The affected structures, systems, components or variables are not assumed to be initiators of analyzed events and are not assumed to mitigate accident or transient events. The requirements and surveillances for these affected structures, systems, components or variables will be relocated from the Technical Specifications to an appropriate administratively controlled document under licensee control. Therefore, this change does not involve an increase in the probability or consequences of an accident previously evaluated.

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

   The proposed change does not necessitate a physical alteration of the plant (no new or different type of equipment will be installed) or change in parameters governing normal plant operation. The proposed change will not impose any different requirements and adequate control of information will be maintained. Thus, this change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

   The proposed change will not reduce a margin of safety because it has no impact on any safety analysis assumptions. In addition, the affected requirement will be relocated to an owner controlled document for which future changes will be evaluated pursuant to the requirements of licensee controlled programs. Therefore, this change does not involve a reduction in a margin of safety.

B 3.4 REACTOR COOLANT SYSTEM (RCS)

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

BASES

BACKGROUND

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

Figures 3.4.3-1 and 3.4.3-2

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

INSERT B.3.4.3-4

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

INSERT B.3.4.3-1

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

10 CFR 50, Appendix G (Ref. 8), requires the establishment of P/T limits for specific material fracture toughness requirements of the RCPB materials. Reference 8 requires an adequate margin to brittle failure during normal operation, anticipated operational occurrences, and system hydrostatic tests. It mandates the use of the American Society of Mechanical Engineers (ASME) Code, Section III, Appendix G (Ref. 8).

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

INSERT B.3.4.3-2

(continued)

The following limitations apply to these figures:

a. Over the temperature range from COLD SHUTDOWN to hot operating conditions, the heatup rate shall not exceed 60°F/hr in any one hour period.

b. Allowable combinations of pressure and temperature for a specific cooldown rate are below and to the right of the limit lines for that rate as shown in Figure 3.4.3-2. This rate shall not exceed 100°F/hr in any one hour period. The limit lines for cooling rates between those shown in Figure 3.4.3-2 may be obtained by interpolation.

c. Primary system hydrostatic leak tests may be performed as necessary provided the test temperature limitation as noted on Figure 3.4.3-1 is not violated. The maximum hydrostatic test pressure should remain below 2335 psig.

the accumulators are capable of supplying sufficient nitrogen to operate the PORVs if they are needed for RCS pressure control, and normal nitrogen and the backup instrument air systems are not available. Backup instrument air is supplied when the accumulator reaches its low pressure setpoint. This SR must be performed by isolating the normal air and nitrogen supplies from the PORVs.

BASES

BACKGROUND (continued)

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

a. Residual Heat Removal (RHR) System;

b. Safety Injection System; and

c. Chemical and Volume Control System.

The PIVs are listed in the FSAR, Section [ ] (Ref. 6).

Table B 3.4.14-1

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.

APPLICABLE SAFETY ANALYSES

Reference 4 identified potential intersystem LOCAs as a significant contributor to the risk of core melt. The dominant accident sequence in the intersystem LOCA category is the failure of the low pressure portion of the RHR System outside of containment. The accident is the result of a postulated failure of the PIVs, which are part of the RCPB, and the subsequent pressurization of the RHR System downstream of the PIVs from the RCS. Because the low pressure portion of the RHR System is typically designed for 600 psig, overpressurization failure of the RHR low pressure line would result in a LOCA outside containment and subsequent risk of core melt.

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.

RCS PIV leakage satisfies Criterion 2 of the NRC Policy Statement.

LCO

INSERT B 3.4.14-3

RCS PIV leakage is identified LEAKAGE into closed systems connected to the RCS. Isolation valve leakage is usually on the order of drops per minute. Leakage that increases

(continued)

ITS Insert B.3.4.14-3 (RCS Pressure Isolation Valves (PIVs))

OPERABILITY of the PIVs is primarily based on meeting acceptable leakage criteria and the assurance that the RHR System PIVs cannot be opened when the RCS is pressurized greater than the RHR System piping design pressure. For a PIV to be considered OPERABLE, it must be functional as a pressure isolation device and the PIV leakage must be within limits of SR 3.4.14.1. Additionally, the RHR System interlock must be OPERABLE.

BASES

LCO (continued)

significantly suggests that something is operationally wrong and corrective action must be taken.

INSERT B3.4.14-1

~~The LCO PIV leakage limit is 0.5 gpm per nominal inch of valve size with a maximum limit of 5 gpm. The previous criterion of 1 gpm for all valve sizes imposed an unjustified penalty on the larger valves without providing information on potential valve degradation and resulted in higher personnel radiation exposures. A study concluded a leakage rate limit based on valve size was superior to a single allowable value.~~

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) in those types of valves in which the higher service pressure will tend to diminish the overall leakage channel opening. In such cases, 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.

APPLICABILITY

In MODES 1, 2, 3, and 4, this LCO applies because the PIV leakage potential is greatest when the RCS is pressurized. In MODE 4, valves in the RHR flow path are not required to meet the requirements of this LCO when in, or during the transition to or from, the RHR mode of operation.

In MODES 5 and 6, 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.

ACTIONS

The Actions are modified by two Notes. Note 1 provides clarification that each flow path allows separate entry into a Condition. This is allowed based upon the functional independence of the 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

(continued)

BASES

ACTIONS (continued) - degraded the ability of the interconnected system to perform its safety function.

When using a manual valve to isolate the affected system, the manual valve shall be closed. As an additional measure to ensure the manual valve remains closed, the valve shall be locked in the closed position. Deactivating an automatic valve includes deenergizing the associated power supply.

A.1 and A.2

The flow path must be isolated by two valves. Required Actions A.1 and A.2 are modified by a Note that the valves used for isolation must meet the same leakage requirements as the PIVs and must be within the RCPB for the high pressure portion of the system.

Required Action A.1 requires that the isolation of the affected system with one valve must be performed within 4 hours. Four hours provides time to reduce leakage in excess of the allowable limit and to isolate the affected system if leakage cannot be reduced. The 4 hour Completion Time allows the actions and restricts the operation with leaking isolation valves.

Required Action A.2 specifies that the double isolation barrier of two valves be restored by closing some other valve qualified for isolation or restoring one leaking PIV. The 72 hour Completion Time after exceeding the limit considers the time required to complete the Action and the low probability of a second valve failing during this time period.

or

The 72 hour Completion Time after exceeding the limit allows for the restoration of the leaking PIV to OPERABLE status. This timeframe considers the time required to complete this Action and the low probability of a second valve failing during this period. (Reviewer Note: Two options are provided for Required Action A.2. The second option (72 hour restoration) is appropriate if isolation of a second valve would place the unit in an unanalyzed condition.)

If the Required Actions and Completion Times of Condition A or B are not met,

C.1 and C.2

If leakage cannot be reduced, [the system isolated,] or the other Required Actions accomplished, the plant must be brought to a MODE in which the requirement does not apply. To achieve this status, the plant must be brought to MODE 3

(continued)

BASES

ACTIONS

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

within 6 hours and MODE 5 within 36 hours. This Action may reduce the leakage and also reduces the potential for a LOCA outside the containment. 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.

B ~~C.1~~

The inoperability of the RHR autoclosure interlock renders the RHR suction isolation valves ~~not~~ capable of isolating in response to a high pressure condition and preventing inadvertent opening of the valves at RCS pressures in excess of the RHR systems design pressure. If the RHR autoclosure interlock is inoperable, operation may continue as long as the affected RHR suction penetration is closed by at least one closed manual or deactivated automatic valve within 4 hours. This Action accomplishes the purpose of the autoclosure function. interlock

SURVEILLANCE REQUIREMENTS

SR 3.4.14.1

Performance of leakage testing on each RCS PIV or isolation valve used to satisfy Required Action A.1 and Required Action A.2 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. listed in Table B 3.4.14-1

INSERT B.3.4.14.4

For the two PIVs in series, the leakage 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.

More than one valve may be tested in parallel. The combined leakage must be within the limits of this SR. In addition, the minimum differential pressure when performing the SR shall be < 150 psid

Testing is to be performed every ~~18~~ months, a typical refueling cycle. ~~If the plant does not go into MODE 5 for at least X days.~~ The ~~18 month~~ Frequency is consistent with

INSERT B 3.4 14-2

(continued)

To satisfy ALARA requirements, leakage may be measured indirectly (as from the performance of pressure indicators) if accomplished in accordance with approved procedures and supported by computations showing that the method is capable of demonstrating valve compliance with the leakage criteria. Leakage rates > 1.0 gpm and ≤ 5.0 gpm are considered unacceptable if the latest measured rate exceeds the rate determined by the previous test by an amount that reduces the margin between measured leakage rate and the 5.0 gpm limit by ≥ 50%. Leakage rates > 5.0 gpm are considered to be unacceptable.

BASES

SURVEILLANCE REQUIREMENTS

SR 3.4.14.1 (continued)

~~10 CFR 50.55a(g) (Ref. 8) as contained in the Inservice Testing Program, is within frequency allowed by the American Society of Mechanical Engineers (ASME) Code, Section XI (Ref. 9), and is based on the need to perform such surveillances under the conditions that apply during an outage and the potential for an unplanned transient if the Surveillance were performed with the reactor at power.~~

In addition, testing must be performed once after the valve has been opened by flow or exercised to ensure tight reseating. PIVs disturbed in the performance of this Surveillance should also be tested unless ~~documentation shows~~ that an infinite testing loop cannot practically be avoided. Testing must be performed within 24 hours after the valve has been reseated. Within 24 hours is a reasonable and practical time limit for performing this test after opening or reseating a valve.

The leakage limit is to be met at the RCS pressure associated with MODES 1 and 2. This permits leakage testing at high differential pressures with stable conditions not possible in the MODES with lower pressures.

Entry into MODES 3 and 4 is allowed to establish the necessary differential pressures and stable conditions to allow for performance of this Surveillance. The Note that allows this provision is complementary to the Frequency of prior to entry into MODE 2 whenever the unit has been in MODE 5 for 7 days or more, if leakage testing has not been performed in the previous 9 months. In addition, this Surveillance is not required to be performed on the RHR System when the RHR System is aligned to the RCS in the shutdown cooling mode of operation. PIVs contained in the RHR shutdown cooling flow path must be leakage rate tested after RHR is secured and stable unit conditions and the necessary differential pressures are established.

~~SR 3.4.14.2 and SR 3.4.14.3~~

Verifying that the RHR ~~autoclosure~~ interlocks are OPERABLE ensures that RCS pressure will not pressurize the RHR system beyond 125% of its design pressure of 600 psig. The interlock setpoint ~~that~~ prevents the valves from being

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.4.14.2 ~~and SR 3.4.14.3~~ (continued)

opened is set so the actual RCS pressure must be ≤ [425] psig to open the valves. This setpoint ensures the RHR design pressure will not be exceeded and the RHR relief valves will not lift. The [18] month Frequency is based on the need to perform the Surveillance under conditions that apply during a plant outage. The [18] month Frequency is also acceptable based on consideration of the design reliability (and confirming operating experience) of the equipment.

~~These SRs are modified by Notes allowing the RHR autoclosure function to be disabled when using the RHR System suction relief valves for cold overpressure protection in accordance with SR 3.4.12.7.~~

REFERENCES

1. 10 CFR 50.2.
2. 10 CFR 50.55a(c).
3. ~~10 CFR 50, Appendix A, Section V, GDC 55.~~ UFSAR, Section 3.1
4. WASH-1400 (NUREG-75/014), Appendix V, October 1975.
5. NUREG-0677, May 1980.
6. ~~[Document containing list of PNVs.]~~
7. ASME, Boiler and Pressure Vessel Code, Section XI.
8. ~~10 CFR 50.55a(g).~~

INSERT B.3.14.4-5

Table B 3.4.14-1 (page 1 of 1)
Reactor Coolant System Pressure Isolation Valves

SYSTEM	VALVE NUMBER
1. Low Pressure Safety Injection/Residual Heat Removal	
a. Loop 1, Cold Leg	875A 876A
b. Loop 2, Cold Leg	875B 876B
c. Loop 3, Cold Leg	875C 876C
2. High Pressure Safety Injection	
a. Loop 2, Hot Leg	874B
b. Loop 3, Hot Leg	874A

BASES

ACTIONS (continued)

C.1

If a Required Action and the associated Completion Time of Condition A is not met or if the DOSE EQUIVALENT I-131 is in the unacceptable region of Figure 3.4.16-1, the reactor must be brought to MODE 3 with RCS average temperature < 500°F within 6 hours. The Completion Time of 6 hours is reasonable, based on operating experience, to reach MODE 3 below 500°F from full power conditions in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.4.16.1

SR 3.4.16.1 requires performing a gamma isotopic analysis as a measure of the gross specific activity of the reactor coolant at least once every 7 days. While basically a quantitative measure of radionuclides with half lives longer than 15 minutes, excluding iodines, this measurement is the sum of the degassed gamma activities and the gaseous gamma activities in the sample taken. This Surveillance provides an indication of any increase in gross specific activity.

The analysis shall consist of a quantitative measurement of the total radioactivity of the primary coolant in units of Ci/gm.

Trending the results of this Surveillance allows proper remedial action to be taken before reaching the LCO limit under normal operating conditions. The Surveillance is applicable in MODES 1 and 2, and in MODE 3 with T_{avg} at least 500°F. The 7 day Frequency considers the unlikelihood of a gross fuel failure during the time.

SR 3.4.16.2

This Surveillance is performed in MODE 1 only to ensure iodine remains within limit during normal operation and following fast power changes when fuel failure is more apt to occur. The 14 day Frequency is adequate to trend changes in the iodine activity level, considering gross activity is monitored every 7 days. The Frequency, between 2 and 6 hours after a power change ≥ 15% RTP within a 1 hour period, is established because the iodine levels peak during this time following fuel failure; samples at other times would provide inaccurate results.

(continued)

1 In the conversion of the HBRSEP current Technical Specifications (CTS) to the proposed plant specific Improved Technical Specifications (ITS), certain wording preferences or conventions are adopted which do not result in technical changes (either actual or interpretational). Editorial changes or clarifications which involve the insertion of plant specific terms, parameters, or descriptions are used to preserve consistency with the CTS and licensing basis.

2 Bases for ITS 3.4.1 are modified by incorporating plant specific DNBR safety limits and the appropriate Reference document.

3 Bases text presentation is modified to improve clarity, or to correct a typographical or grammatical error.

4 Not used.

5 The Note and the Frequency discussions in the Bases of ITS SR 3.4.2.1 are modified to clarify that entry into the applicable condition without first performing the Surveillance does not result in non-compliance with the LCO and that entry into the applicable condition of the LCO requires the Surveillance to be met. The plant design incorporates monitoring of Tavg and an automatic alarm as Tavg approaches its limit. As a result, the Surveillance is met by the monitoring of the automatic alarm status. The intent of the Frequency specified in SR 3.4.2.1 is to require verification during the time that the monitoring instrumentation would be in alarm.

6 Bases for ITS 3.4.3 and 3.4.12 are modified by removing references to the Pressure and Temperature Limits Report (PTLR), and retaining CTS Figures 3.1-1 and 3.1-2, which provide RCS heatup and cooldown limitations, respectively, consistent with current licensing basis. The curves depicted in these figures were updated in 1994 to cover operation up to 24 effective full power years (EFPY).

7 Bases are modified to incorporate plant specific safety analyses and/or Bases information or to reflect changes made to the Specifications.

8 The Bases for ITS 3.4.5, 3.4.6, and 3.4.7 contain a Note, permitting RCPs and RHR pumps to be de-energized for ≤1 hour per 8 hour period. This Note is modified by changing the phrase, "per 8 hour period," to "in any 8 hour period," to eliminate any interpretation that these pumps can be de-energized for consecutive 1 hour periods in two 8 hour

B 3.4 REACTOR COOLANT SYSTEM (RCS)

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

BASES

BACKGROUND

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

Figures 3.4.3-1 and 3.4.3-2 contain P/T limit curves for heatup, cooldown, inservice leak and hydrostatic (ISLH) testing, and data for the maximum rate of change of reactor coolant temperature.

The following limitations apply to these figures:

a. Over the temperature range from COLD SHUTDOWN to hot operating conditions, the heatup rate shall not exceed 60°F/hr in any one hour period.

b. Allowable combinations of pressure and temperature for a specific cooldown rate are below and to the right of the limit lines for that rate as shown in Figure 3.4.3-2. This rate shall not exceed 100°F/hr in any one hour period. The limit lines for cooling rates between those shown in Figure 3.4.3-2 may be obtained by interpolation.

c. Primary system hydrostatic leak tests may be performed as necessary provided the test temperature limitation as noted on Figure 3.4.3-1 is not violated. The maximum hydrostatic test pressure should remain below 2335 psig.

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

(continued)

BASES

BACKGROUND (continued)

The ability of the large steel pressure vessel that contains the reactor core and its primary coolant to resist fracture constitutes and important factor in ensuring safety in the nuclear industry. The LCO establishes operating limits that provide a margin to brittle failure of the reactor vessel and piping of the reactor coolant pressure boundary (RCPB). The vessel is the component most subject to brittle failure, and the LCO limits apply mainly to the vessel. The limits do not apply to the pressurizer, which has different design characteristics and operating functions.

10 CFR 50, Appendix G (Ref. 1), requires the establishment of P/T limits for specific 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 hydrostatic tests. It mandates the use of the American Society of Mechanical Engineers (ASME) Code, Section III, Appendix G (Ref. 2).

The beltline region of the reactor pressure vessel is the most critical region of the vessel because it is subjected to neutron bombardment. The overall effects of fast neutron irradiation on the mechanical properties of low alloy ferritic pressure vessel steels, such as the ASTM A302 Grade B parent material of the HBRSEP Unit No. 2 reactor pressure vessel, are well documented in the literature. Generally, low alloy ferritic materials show an increase in hardness and other strength properties and a decrease in ductility and impact toughness under certain conditions of irradiation. Accompanying a decrease in impact strength is an increase in the temperature for the transition from brittle to ductile fracture.

A method for guarding against fast fracture in reactor pressure vessels is presented in Reference 2. The method utilizes fracture mechanics concepts and is based on the reference nil-ductility temperature, RT_{NDT}.

RT_{NDT} is defined as the greater of:

1. The drop weight nil-ductility transition temperature (NDTT, per ASTM E-208), or

2. The temperature 60°F less than the 50 ft-lb (and 35 mils lateral expansion) temperature as determined from

(continued)

BASES

BACKGROUND (continued)

Charpy specimens oriented in a direction normal to the major working direction of the material.

The RT_{NDT} of a given material is used to index that material to a reference stress intensity curve (K_{IR} curve), which appears in Reference 2. The K_{IR} curve is a lower bound of dynamic, crack arrest, and static fracture toughness results obtained from several heats of pressure vessel steel. When a given material is indexed to the K_{IR} curve, allowable stress intensity factors can be obtained for this material as a function of temperature. Allowable operating limits can then be determined utilizing these allowable stress intensity factors.

The Certified Material Test Reports (CMTR) for the original steam generators provided records of Charpy V-notch tests performed at +10°F. Acceptable Charpy V-notch tests of +10°F indicate RT_{NDT} is at or below this temperature. The steam generator lower assemblies were replaced in 1984 and the material test results indicate the highest RT_{NDT} is 60°F or below. The ASME Code recommends that hydrostatic tests be performed at a temperature not lower than RT_{NDT} plus 60°F, thus the pressurizing temperature for the steam generator shell is established at 120°F to provide protection against nonductile failure at the test pressure. The value of RT_{NDT}, and in turn the operating limits of nuclear power plants, can be adjusted to account for the effects of radiation on reactor vessel material properties. The radiation embrittlement or changes in mechanical properties of a given reactor pressure vessel still can be monitored by a surveillance program such as the HBRSEP Unit No. 2 Reactor Vessel Radiation Surveillance Program (Ref. 3), where a surveillance capsule is periodically removed from the reactor pressure vessel and the encapsulated specimens tested. These data are compared to data from pertinent radiation effects studies and an increase in the Charpy V-notch 30 ft-lb temperature (ΔRT_{NDT}) due to irradiation is added to the original ΔRT_{NDT} to adjust the RT_{NDT} for radiation embrittlement. This adjusted RT_{NDT} (RT_{NDT} initial + ΔRT_{NDT}) is utilized to index the material to the K_{IR} curve and in turn to set operating limits which take into account the effects of irradiation on the reactor pressure vessel materials. Allowable pressure - temperature relationships for various heatup and cooldown rates are calculated using methods (Ref. 4) derived from Appendix G to Section III of the ASME Boiler and Pressure Vessel Code. The approach

(continued)

BASES

BACKGROUND (continued)

specifies that the allowable total stress intensity factor, K_I, at any time during heatup or cooldown cannot be greater than that shown on the K_{IR} curve in Appendix G for the metal temperature at that time. Furthermore, the approach applies an explicit safety factor of 2.0 on the stress intensity factor induced by pressure gradients.

Following the generation of pressure - temperature curves for both the steady state and finite heatup rate situations. The final limit curves are produced in the following fashion. First, a composite curve is constructed based on a point-by-point comparison of the steady state and finite heatup rate data. At any given temperature, the allowable pressure is taken to be the lesser of the two values taken from the curves under consideration. The composite curve is then adjusted to allow for possible errors in the pressure and temperature sensing instruments.

The use of the composite curve is mandatory in setting heatup limitations because it is possible for conditions to exist such that over the course of the heatup ramp the controlling analysis switches from the outside diameter (OD) to the inside diameter (ID) location; and the pressure limit must, at all times, be based on the most conservative case. The cooldown analysis proceeds in the same fashion as that for heatup, with the exception that the controlling location is always at the ID position. The thermal gradients induced during cooldown tend to produce tensile stresses at the ID location, and compressive stresses at the OD position. Thus, the ID flaw is clearly the worst case.

As in the case of heatup, allowable pressure - temperature relationships are generated for both steady state and finite cooldown rate situations. Composite limit curves are then constructed for each cooldown rate of interest. Adjustments are made to account for pressure and temperature instrumentation error.

The criticality limit curve includes the Reference 1 requirement that it be ≥ 40°F above the heatup curve or the cooldown curve, and not less than the minimum permissible temperature for ISLH testing. However, the criticality curve is not operationally limiting; a more restrictive limit exists in LCO 3.4.2, "RCS Minimum Temperature for Criticality."

(continued)

BASES (continued)

APPLICABLE SAFETY ANALYSES

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

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

LCO

The two elements of this LCO are:

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

b. Limits on the rate of change of temperature.

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

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

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

a. The severity of the departure from the allowable operating P/T 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

(continued)

BASES

LCO (continued)

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

APPLICABILITY

The RCS P/T limits LCO provides a definition of acceptable operation for prevention of nonductile failure in accordance with 10 CFR 50, Appendix G (Ref. 1). Although the P/T limits were developed to provide guidance for operation during heatup or cooldown (MODES 3, 4, and 5) or ISLH testing, their Applicability is at all times in keeping with the concern for nonductile failure. The limits do not apply to the pressurizer.

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

ACTIONS

A.1 and A.2

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

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

Besides restoring operation within limits, an evaluation is required to determine if RCS operation can continue. The evaluation must verify the RCPB integrity remains acceptable and must be completed before continuing operation. Several methods may be used, including comparison with pre-analyzed

(continued)

BASES

ACTIONS

A.1 and A.2 (continued)

transients in the stress analyses, new analyses, or inspection of the components.

ASME Code, Section XI, Appendix E (Ref. 5), may be used to support the evaluation. However, its use is restricted to evaluation of the vessel beltline.

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

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

B.1 and B.2

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

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

(continued)

BASES

ACTIONS

B.1 and B.2 (continued)

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

Pressure and temperature are reduced by bringing the plant to MODE 3 within 6 hours and to MODE 5 with RCS pressure < 400 psig within 36 hours.

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

C.1 and C.2

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

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

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

ASME Code, Section XI, Appendix E (Ref. 5), may be used to support the evaluation. However, its use is restricted to evaluation of the vessel beltline.

(continued)

BASES

ACTIONS

C.1 and C.2 (continued)

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

SURVEILLANCE REQUIREMENTS

SR 3.4.3.1

Verification that operation is within the limits of Figures 3.4.3-1 and 3.4.3-2 is required every 30 minutes when RCS pressure and temperature conditions are undergoing planned changes. This Frequency is considered reasonable in view of the control room indication available to monitor RCS status. Also, since temperature rate of change limits are specified in hourly increments, 30 minutes permits assessment and correction for minor deviations within a reasonable time.

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

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

REFERENCES

1. 10 CFR 50, Appendix G.

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

3. Yanichko, S. E., "Carolina Power & Light Company, H. B. Robinson Unit No. 2 Reactor Vessel Radiation Surveillance Program," Westinghouse Nuclear Energy Systems, WCAP-7373, January, 1970.

4. Norris, E. B., "Reactor Vessel Material Surveillance Program for H. B. Robinson Unit No. 2, Analysis of

(continued)

BASES

REFERENCES (continued)

Capsule V," Southwest Research Institute, Final Report, SWRI Project No. 02-4397.

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

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.4.11.3

Operating the solenoid air control valves and check valves on the nitrogen accumulators ensures the PORV control system actuates properly when called upon. The Frequency of 18 months is based on a typical refueling cycle and the Frequency of the other Surveillances used to demonstrate PORV OPERABILITY.

SR 3.4.11.4

The Surveillance demonstrates that the accumulators are capable of supplying sufficient nitrogen to operate the PORVs if they are needed for RCS pressure control, and normal nitrogen and the backup instrument air systems are not available. Backup instrument air is supplied when the accumulator reaches its low pressure setpoint. This SR must be performed by isolating the normal air and nitrogen supplies from the PORVs. The Frequency of 18 months is based on a typical refueling cycle and industry accepted practice.

REFERENCES

1. UFSAR, Section 15.6.

2. Generic Letter 90-06, "Resolution of Generic Issue 70, 'Power-Operated Relief Valve and Block Valve Reliability,' and Generic Issue 94, 'Additional Low-Temperature Overpressure Protection for Light-Water Reactors,' Pursuant to 10 CFR 50.54(f)," dated June 25, 1990.

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

BASES

BACKGROUND (continued)

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

a. Residual Heat Removal (RHR) System;

b. Safety Injection System; and

c. Chemical and Volume Control System.

The PIVs are listed in Table B 3.4.14-1.

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.

APPLICABLE SAFETY ANALYSES

Reference 4 identified potential intersystem LOCAs as a significant contributor to the risk of core melt. The dominant accident sequence in the intersystem LOCA category is the failure of the low pressure portion of the RHR System outside of containment. The accident is the result of a postulated failure of the PIVs, which are part of the RCPB, and the subsequent pressurization of the RHR System downstream of the PIVs from the RCS. Because the low pressure portion of the RHR System is designed for 600 psig, overpressurization failure of the RHR low pressure line would result in a LOCA outside containment and subsequent risk of core melt.

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.

RCS PIV leakage satisfies Criterion 2 of the NRC Policy Statement.

(continued)

BASES (continued)

LCO

OPERABILITY of the PIVs is primarily based on meeting acceptable leakage criteria and the assurance that the RHR System PIVs cannot be opened when the RCS is pressurized greater than the RHR System piping design pressure. For a PIV to be considered OPERABLE, it must be functional as a pressure isolation device and the PIV leakage must be within limits of SR 3.4.14.1. Additionally, the RHR System interlock must be OPERABLE.

RCS PIV leakage is identified 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.

The LCO PIV leakage limit is administratively controlled to 1.0 gpm at the first test of each valve with an increasing limit based on the previous leakage rate and maximum limit of 5 gpm for subsequent tests. Leakage rates ≤ 5.0 gpm are acceptable if the latest measured leakage rate has not exceeded the rate determined by the previous test by an amount that reduces the margin between the previous measured leakage rate and the maximum leakage rate of 5.0 gpm by > 50%. Leakage rates ≤ 5.0 gpm which are increasing at rates which reduce the margin ≤ 50% between tests provide reasonable assurance that the leakage rate will not increase beyond 5.0 gpm before the next scheduled leak test. Leakage rates < 5.0 gpm ensure the leakage will be within the capabilities of the low pressure system relief valve capacity (with some margin) and prevent overpressurization.

Reference 6 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) in those types of valves in which the higher service pressure will tend to diminish the overall leakage channel opening. In such cases, 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.

APPLICABILITY

In MODES 1, 2, 3, and 4, this LCO applies because the PIV leakage potential is greatest when the RCS is pressurized.

(continued)

BASES

APPLICABILITY (continued)

In MODE 4, valves in the RHR flow path are not required to meet the requirements of this LCO when in, or during the transition to or from, the RHR mode of operation.

In MODES 5 and 6, 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.

ACTIONS

The Actions are modified by two Notes. Note 1 provides clarification that each flow path allows separate entry into a Condition. This is allowed based upon the functional independence of the 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.

A.1 and A.2

The flow path must be isolated by two valves. Required Actions A.1 and A.2 are modified by a Note that the valves used for isolation must meet the same leakage requirements as the PIVs and must be within the RCPB or the high pressure portion of the system.

Required Action A.1 requires that the isolation of the affected system with one valve must be performed within 4 hours. When using a manual valve to isolate the affected system, the manual valve shall be closed. As an additional measure to ensure the manual valve remains closed, the valve shall be locked in the closed position. Deactivating an automatic valve includes deenergizing the associated power supply. Four hours provides time to reduce leakage in excess of the allowable limit and to isolate the affected system if leakage cannot be reduced. The 4 hour Completion Time allows the actions and restricts the operation with leaking isolation valves.

Required Action A.2 specifies that the double isolation barrier of two valves be restored by closing some other valve qualified for isolation or restoring one leaking PIV.

(continued)

BASES

ACTION

A.1 (continued)

The 72 hour Completion Time after exceeding the limit considers the time required to complete the Action and the low probability of a second valve failing during this time period.

B.1

The inoperability of the RHR interlock renders the RHR suction isolation valves capable of inadvertent opening at RCS pressures in excess of the RHR systems design pressure. If the RHR interlock is inoperable, operation may continue as long as the affected RHR suction penetration is closed by at least one closed manual or deactivated automatic valve within 4 hours. This Action accomplishes the purpose of the interlock function.

C.1 and C.2

If the Required Actions and Completion Times of Condition A or B are not met, the plant must be brought to a MODE in which the requirement does not apply. To achieve this status, the plant must be brought to MODE 3 within 6 hours and MODE 5 within 36 hours. This Action may reduce the leakage and also reduces the potential for a LOCA outside the containment. The allowed Completion Times are reasonable based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.4.14.1

Performance of leakage testing on each RCS PIV or isolation valve used to satisfy Required Action A.1 and Required Action A.2 is required to verify that leakage is below the specified limit and to identify each leaking valve. The leakage limit applies to each valve listed in Table B 3.4.14-1. Leakage testing requires a stable pressure condition.

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.4.14.1 (continued)

To satisfy ALARA requirements, leakage may be measured indirectly (as from the performance of pressure indicators) if accomplished in accordance with approved procedures and supported by computations showing that the method is capable of demonstrating valve compliance with the leakage criteria. Leakage rates > 1.0 gpm and ≤ 5.0 gpm are considered unacceptable if the latest measured rate exceeds the rate determined by the previous test by an amount that reduces the margin between measured leakage rate and the 5.0 gpm limit by ≥ 50%. Leakage rates > 5.0 gpm are considered to be unacceptable.

More than one valve may be tested in parallel. The combined leakage must be within the limits of this SR. In addition, the minimum differential pressure when performing the SR shall not be < 150 psid. For two PIVs in series, the leakage 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.

Testing is to be performed every 18 months, a typical refueling cycle. Testing must also be performed once prior to entering MODE 2 whenever the unit has been in MODE 5 for at least 7 days if leakage testing has not been performed in the previous 9 months. The 18 month Frequency is consistent with the frequency allowed by the American Society of Mechanical Engineers (ASME) Code, Section XI (Ref. 6).

In addition, testing must be performed once after the valve has been opened by flow or exercised to ensure tight reseating. PIVs disturbed in the performance of this Surveillance should also be tested unless it has been established per Note 3 that an infinite testing loop cannot practically be avoided. Testing must be performed within 24 hours after the valve has been reseated if in MODES 1 or 2, or prior to entry into MODE 2 if not in MODES 1 or 2 at the end of the 24 hour period. Within 24 hours is a reasonable and practical time limit for performing this test after opening or reseating a valve.

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.4.14.1 (continued)

The leakage limit is to be met at the RCS pressure associated with MODES 1 and 2. This permits leakage testing at high differential pressures with stable conditions not possible in the MODES with lower pressures.

Entry into MODES 3 and 4 is allowed to establish the necessary differential pressures and stable conditions to allow for performance of this Surveillance. The Note that allows this provision is complementary to the Frequency of prior to entry into MODE 2 whenever the unit has been in MODE 5 for 7 days or more, if leakage testing has not been performed in the previous 9 months. In addition, this Surveillance is not required to be performed on the RHR System when the RHR System is aligned to the RCS in the shutdown cooling mode of operation. PIVs contained in the RHR shutdown cooling flow path must be leakage rate tested after RHR is secured and stable unit conditions and the necessary differential pressures are established.

SR 3.4.14.2

Verifying that the RHR interlock is OPERABLE ensures that RCS pressure will not pressurize the RHR system beyond 125% of its design pressure of 600 psig. The interlock setpoint prevents the valves from being opened and is set so the actual RCS pressure must be < 465 psig to open the valves. This setpoint ensures the RHR design pressure will not be exceeded and the RHR relief valves will not lift. The 18 month Frequency is based on the need to perform the Surveillance under conditions that apply during a plant outage. The 18 month Frequency is also acceptable based on consideration of the design reliability (and confirming operating experience) of the equipment.

REFERENCES

1. 10 CFR 50.2.
2. 10 CFR 50.55a(c).
3. UFSAR, Section 3.1.
4. WASH-1400 (NUREG-75/014), Appendix V, October 1975.

(continued)

BASES

REFERENCES (continued)

5. NUREG-0677, May 1980.

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

BASES

Table B 3.4.14-1 (page 1 of 1)
Reactor Coolant System Pressure Isolation Valves

SYSTEM	VALVE NUMBER
1. Low Pressure Safety Injection/Residual Heat Removal	
a. Loop 1, Cold Leg	875A 876A
b. Loop 2, Cold Leg	875B 876B
c. Loop 3, Cold Leg	875C 876C
2. High Pressure Safety Injection	
a. Loop 2, Hot Leg	874B
b. Loop 3, Hot Leg	874A

BASES

ACTIONS (continued)

C.1

If a Required Action and the associated Completion Time of Condition A is not met or if the DOSE EQUIVALENT I-131 is in the unacceptable region of Figure 3.4.16-1, the reactor must be brought to MODE 3 with RCS average temperature < 500°F within 6 hours. The Completion Time of 6 hours is reasonable, based on operating experience, to reach MODE 3 below 500°F from full power conditions in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.4.16.1

SR 3.4.16.1 requires performing a gamma isotopic analysis as a measure of the gross specific activity of the reactor coolant at least once every 7 days. The analysis shall consist of a qualitative measurement of the total radioactivity of the primary coolant in units of Ci/gm. While basically a quantitative measure of radionuclides with half lives longer than 15 minutes, excluding iodines, this measurement is the sum of the degassed gamma activities and the gaseous gamma activities in the sample taken. This Surveillance provides an indication of any increase in gross specific activity.

Trending the results of this Surveillance allows proper remedial action to be taken before reaching the LCO limit under normal operating conditions. The Surveillance is applicable in MODES 1 and 2, and in MODE 3 with T_{avg} at least 500°F. The 7 day Frequency considers the unlikelihood of a gross fuel failure during the time.

SR 3.4.16.2

This Surveillance is performed in MODE 1 only to ensure iodine remains within limit during normal operation and following fast power changes when fuel failure is more apt to occur. The 14 day Frequency is adequate to trend changes in the iodine activity level, considering gross activity is monitored every 7 days. The Frequency, between 2 and 6 hours after a power change ≥ 15% RTP within a 1 hour period, is established because the iodine levels peak during

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.4.16.2 (continued)

this time following fuel failure; samples at other times would provide inaccurate results.

SR 3.4.16.3

A radiochemical analysis for $\bar{E}$ determination is required every 184 days (6 months) with the plant operating in MODE 1 equilibrium conditions. The $\bar{E}$ determination directly relates to the LCO and is required to verify plant operation within the specified gross activity LCO limit. The analysis for $\bar{E}$ is a measurement of the average energies per disintegration for isotopes with half lives longer than 15 minutes, excluding iodines. The Frequency of 184 days recognizes $\bar{E}$ does not change rapidly.

This SR has been modified by a Note that indicates sampling is required to be performed within 31 days after a minimum of 2 effective full power days and 20 days of MODE 1 operation have elapsed since the reactor was last subcritical for at least 48 hours. This ensures that the radioactive materials are at equilibrium so the analysis for $\bar{E}$ is representative and not skewed by a crud burst or other similar abnormal event.

REFERENCES

1. 10 CFR 100.11, 1973.

2. UFSAR, Section 15.6.3.

SUPPLEMENT 5
CONVERSION PACKAGE SECTION 3.5
PAGE INSERTION INSTRUCTIONS

Remove and insert the following pages into Enclosure 13 to Serial: RNP-RA/96-0141.

	Remove Page	Insert Page
a.	Part 1, "Markup of Current Technical Specifications (CTS)"	
	3.3-3, 3.3-4, 3.3-2, 3.3-3, 3.3-4	3.3-3, 3.3-4, 3.3-2, 3.3-3, 3.3-4
	3.3-1, 3.3-2	3.3-1, 3.3-2
b.	Part 2, "Discussion of Changes (DOCs) for CTS Markup"	
	1 through 9	1 through 9
c.	Part 3, " No Significant Hazards Consideration (NSHC), And Basis for Categorical Exclusion from 10 CFR 51.22	
	NA	
d.	Part 4, "Markup of NUREG-4131, Revision 1, Standard Technical Specifications-Westinghouse Plants, (ISTS)"	
	3.5-1	3.5-1
	-	3.5-1a
	3.5-2, 3.5-3, 3.5-3a, 3.5-4, 3.5-4a, 3.5-5	3.5-2, 3.5-3, 3.5-3a, 3.5-4, 3.5-4a, 3.5-5
	3.5-5a, 3.5-6, 3.5-6a, 3.5-8	3.5-5a, 3.5-6, 3.5-6a, 3.5-8
e.	Part 5, "Justification of Differences (JFDs) to ISTS"	
	1 & 2	1 through 4
f.	Part 6, "Markup of ISTS Bases"	
	B 3.5-6, B 3.5-7	B 3.5-6, B 3.5-7
	-	B 3.5-6a & B 3.5-7a
	B 3.5-8	B 3.5-8
	Insert B 3.5.1-3 (no page number)	B 3.5-8a
	B 3.5-15, B 3.5-16	B 3.5-15, B 3.5-16
	Insert B 3.5.2-4 (no page number)	B 3.5-16a
	B 3.5-17	B 3.5-17
	Insert B 3.5.2-5 (no page number)	B 3.5-17a
	B 3.5-18	B 3.5-18
	-	B 3.5-18a
	Insert B 3.5.2-6 (no page number)	B 3.5-19a
	B 3.5-22	B 3.5-22

SUPPLEMENT 5
CONVERSION PACKAGE SECTION 3.5
PAGE INSERTION INSTRUCTIONS

Remove and insert the following pages into Enclosure 13 to Serial: RNP-RA/96-0141.

	Remove Page	Insert Page
g.	Part 7, "Justification for Differences (JFDs) to ISTS Bases"	
	1 through 3	1 through 4
h.	Part 8, "Proposed HBRSEP, Unit No. 2 ITS"	
	3.5-1 through 3.5-7	3.5-1 through 3.5-7
	3.5-9	3.5-9
i.	Part 9. "Proposed Bases to HBRSEP, Unit No. 2 ITS Bases"	
	B 3.5-6 through B 3.5-8	B 3.5-6 through B 3.5-8
	-	B 3.5-8a
	B 3.5-14 through B 3.5-19	B 3.5-14 through B 3.5-19
	B 3.5-21	B 3.5-21
j.	Part 10. "ISTS Generic Changes"	
	NA	

Specification 3.5.1

ITS

A1

See 3.4.4

A4

i. Power operation with less than three loops in service is prohibited.

3.3.1.2 During power operation, the requirements of 3.3.1.1 may be modified to allow any one of the following components to be inoperable. If the system is not restored to meet the requirements of 3.3.1.1 within the time period specified, the reactor shall be placed in the hot shutdown condition utilizing normal operating procedures. If the requirements of 3.3.1.1 are not satisfied within an additional 48 hours, the reactor shall be placed in the cold shutdown condition utilizing normal operating procedures.

Require Action and associated completion times of Condition A or B not met

M4

MODE 3 within 6 hours

[RA D.1]

[RA D.2]

12

M5

Reduce pressurizer pressure to ≤ 1000 psig

M5

a. One accumulator may be isolated or otherwise inoperable relative to the requirements of 3.3.1.1.b for a period not to exceed four hours.

[RA C.1]

for reasons other than Condition A

b. If one safety injection pump becomes inoperable during normal reactor operation, the reactor may remain in operation for a period not to exceed 24 hours.

c. If one residual heat removal pump becomes inoperable during normal reactor operation, the reactor may remain in operation for a period not to exceed 24 hours.

See 3.5.2

Add RA A.1

L1

Add RA E.1

M7

Amendment No. ~~97~~, ~~115~~, ~~119~~, ~~146~~, 153

Supplement 5

Specification 3.5.1

A1

d. If one residual heat exchanger becomes inoperable during normal reactor operation, the reactor may remain in operation for a period not to exceed 24 hours.

e. If any one flow path including valves of the safety injection or residual heat removal system is found to be inoperable during normal reactor operation, the reactor may remain in operation for a period not to exceed 24 hours. The hot leg injection paths of the Safety Injection System, including valves, are not subject to the requirements of this specification.

See 3.5.2 and 3.5.3

~~f. Power or air supply may be restored to any valve referenced in 3.3.1.1.g. and 3.3.1.1.h. for the purpose of valve testing or maintenance providing no more than one valve has power restored and provided that testing and maintenance is completed and power removed within 24 hours except for accumulator isolation valves (MOV 865 A,B,&C) which will have this time period limited to four hours.~~

A10

Add RA B.1
B.2

Supplement 5

ITS

Specification 3.5.2

A1

b. Each accumulator is pressurized to at least 600 psig and contains at least 825 ft^3 and no more than 841 ft^3 of water with a boron concentration of at least 1950 ppm. No accumulator may be isolated. — See 3.5.1

[LCO 3.5.2] c. Two ~~safety injection~~ pumps are operable, ~~each capable of automatic initiation from a separate emergency bus.~~ ECCS trains shall be — A2

d. Two residual heat removal pumps are operable.

e. Two residual heat exchangers are operable.

f. All essential features including valves, interlocks, and piping associated with the above components are operable. — LA2

[SR 3.5.2.1] g. During conditions of operation with reactor coolant pressure in excess of 1000 psig the A.C. control power shall be removed from the following motor operated valves with the valve in the specified position: — M22

Valves	Position
MOV 862 A&B	Open
MOV 864 A&B	Open
MOV 865 A,B,&C	Open
MOV 878 A&B	Open
MOV 863 A&B	Closed
MOV 866 A&B	Closed

See 3.5.1

M22

[SR 3.5.2.7] h. During conditions of operation with reactor coolant pressure in excess of 1000 psig, the air supply to air operated valves 605 and 758 shall be shut off with valves in the closed position. — M22

Add SR 3.5.2.3 — A6

Add SR 3.5.2.6 — M11

Add SR 3.5.2.8 — M12

Supplements

ITS

Specification 3.5.2

A1

d. If one residual heat exchanger becomes inoperable during normal reactor operation, the reactor may remain in operation for a period not to exceed 24 hours.

L2

e. If any one flow path including valves of the safety injection or residual heat removal system is found to be inoperable during normal reactor operation, the reactor may remain in operation for a period ~~not to exceed 24 hours.~~ The hot leg injection paths of the Safety Injection System including valves, are not subject to the requirements of this specification.

LA1

f. Power or air supply may be restored to any valve referenced in 3.3.1.1.g. and 3.3.1.1.h. for the purpose of valve testing or maintenance providing no more than one valve has power restored and provided that testing and maintenance is completed and power removed within 24 hours ~~except for accumulator isolation valves (MOV 865 A,B,&C)~~ which will have this time period limited to four hours.

A10

See 3.5.1

Add RA B.1 B.2

A10

Add Applicability Note 1

M23

Add Applicability Note 2

L8

Supplement 5

Specification 3.5.3

ITS

A1

3.3 EMERGENCY CORE COOLING SYSTEM, AUXILIARY COOLING SYSTEMS, AIR RECIRCULATION FAN COOLERS, CONTAINMENT SPRAY, POST ACCIDENT CONTAINMENT VENTING SYSTEM, AND ISOLATION SEAL WATER SYSTEM

~~Applicability~~

~~Applies to the operating status of the Emergency Core Cooling System, Auxiliary Cooling Systems, Air Recirculation Fan Coolers, Containment Spray, Post Accident Containment Venting System, and Isolation Seal Water System.~~

~~Objective~~

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

A4

Specification

[Applicability]

3.3.1 Safety Injection and Residual Heat Removal Systems

IN MODE 4

3.3.1.1 ~~The reactor shall not be made critical unless~~ the following conditions are met:

a. The refueling water tank contains not less than 300,000 gallons of water with a boron concentration of at least 1950 ppm.

See 3.5.4

Add LCO Note

A12

M11

Add SR 3.5.3.1

Amendment No. 106

Supplement 5

ITS

L6

A1

b. Each accumulator is pressurized to at least 600 psig and contains at least 825 ft^3 and no more than 841 ft^3 of water with a boron concentration of at least 1950 ppm. No accumulator may be isolated. — See 3.5.1

ONE

ECCS train shall be — A2

[LCO 3.5.3] c. ~~Two safety injections pumps are~~ operable, ~~each capable of automatic initiation from a separate emergency bus.~~

d. ~~Two residual heat removal pumps are operable.~~

e. ~~Two residual heat exchangers are operable.~~

f. ~~All essential features including valves, interlocks, and piping associated with the above components are operable.~~

LA2

g. During conditions of operation with reactor coolant pressure in excess of 1000 psig the A.C. control power shall be removed from the following motor operated valves with the valve in the specified position:

Valves	Position
MOV 862 A&B	Open
MOV 864 A&B	Open
MOV 865 A,B,&C	Open
MOV 878 A&B	Open
MOV 863 A&B	Closed
MOV 866 A&B	Closed

See 3.5.2

See 3.5.1

see 3.5.2

h. During conditions of operation with reactor coolant pressure in excess of 1000 psig, the air supply to air operated valves 605 and 758 shall be shut off with valves in the closed position. — See 3.5.2

Supplement 5

ADMINISTRATIVE CHANGES

A1 In the conversion of the H.B. Robinson Steam Electric Plant (HBRSEP), Unit No.2 Current Technical Specification (CTS) to the proposed plant specific Improved Technical Specifications (ITS) certain wording preferences or conventions are adopted which do not result in technical changes (either actual or interpretational). Editorial changes, reformatting, and revised numbering are adopted to make ITS consistent with the conventions in the - Standard Technical Specifications, Westinghouse Plants NUREG-1431, Rev 1 (i.e., Improved Standard Technical Specifications (ISTS)).

A2 Some CTS specifications include listings of components, features, attributes, etc. associated with OPERABILITY of CTS equipment. The ITS does not retain the specific listings since they are generically encompassed within the definition of OPERABLE specified in ITS Section 1.1, Definitions. Therefore, this is an administrative change and is consistent with ISTS.

A3 CTS 4.5.2.1 does not include information regarding the ECCS function (i.e., Low Head Safety Injection (LHSI) or High head Safety Injection (HHSI)) associated with the specified valves. This functional information is included within a column of SR 3.5.2.1. Since this column merely provides functional information and does not affect technical requirements, this is an administrative change and is consistent with ISTS.

A4 CTS 3.3.1.1, 3.3.1.2 and 3.3.1.3 collectively impose requirements for ECCS with the reactor critical and in hot shutdown. These CTS operating conditions encompass ITS MODES 1, 2, 3 and 4. ITS 3.5.2 specifies requirements for ECCS in MODES 1, 2 and 3. ITS 3.5.3 specify requirements in MODE 4. Therefore, this is an administrative change and is consistent with ISTS.

A5 Not used.

A6 Although, CTS does not explicitly specify inservice testing of the ECCS pumps, CTS 4.0.1.a requires testing the ECCS pumps in accordance with the inservice testing program. ITS SR 3.5.2.3 specifically requires testing the ECCS pumps in accordance with the inservice testing program. Therefore, this change is administrative and is consistent with NUREG-1431.

A7 CTS 4.5.1.1 specifies the Safety Injection System tests be initiated by a "test safety injection signal." ITS SR 3.5.2.4 specifies an "actual or simulated actuation signal." This change is administrative and is consistent with ISTS.

A8 CTS 4.5.1.2 requires verification that all automatic valves have completed their travel. ITS SR 3.5.2.5 requires verification that valves in the flowpath, not locked sealed or otherwise secured in position, actuate to the correct position. There are no Safety Injection System valves not in the flow path which actuate on an SI signal. Valves which are locked, sealed or otherwise secured in position are maintained in the correct position and thus do not have an actuation requirement. This change is administrative and is consistent with ISTS.

A9 The CTS bases are not retained in the ITS, but are replaced in their entirety. The ITS includes significantly expanded and improved bases. The bases do not define or impose any specific requirements but serve to explain, clarify and document the reasons (i.e., bases) for the associated specification. The bases are not part of the Technical Specifications required by 10 CFR 50.36. Therefore, this is an administrative change and is consistent with ISTS.

A10 CTS 3.3.1.2.f permits power or air supply to be restored to any valve referenced in CTS 3.3.1.1.g and 3.3.1.1.h for testing or maintenance provided no more than one of all the valves identified has power or air restored. This provision in CTS is replaced with Condition B in LCOs 3.5.1 and 3.5.2, which provided required actions in the event one valve has power or air restored. ITS Required Action B.1 requires immediate verification that only one valve among the referenced valves in SRs 3.5.1.5, 3.5.2.1, and 3.5.2.7 has control power or air restored. The Completion Times for LCO 3.5.1 B.2 and LCO 3.5.2 B.2 are consistent with CTS 3.3.1.2.f. Therefore, this change is administrative and has no adverse impact on safey.

A11 Not used.

A12 CTS 3.3.1.3 requires two ECCS RHR subsystems to be OPERABLE when in Hot Shutdown. CTS Hot Shutdown encompasses ITS MODES 3 and 4. Although CTS does not explicitly provide for ECCS RHR subsystems to remain OPERABLE in Hot Shutdown when aligned for decay heat removal, it is inherently required that RHR be aligned for decay heat removal in order to achieve Cold Shutdown. ITS SR 3.5.3.1 Note 1 clarifies the acceptability of the current practice and interpretation of allowing credit to be taken for portions of an ECCS subsystem as an OPERABLE subsystem although they may be manually aligned to function for decay heat removal. The addition of this clarification to ITS SR 3.5.3.1 as a Note, is therefore an editorial change with no effect on the technical specification implementation requirement. Therefore, this change is administrative and is consistent with ISTS.

A13 CTS 3.3.1.1 and 3.3.1.3 collectively impose requirements for ECCS with the reactor critical and in hot shutdown. These CTS operating

conditions encompass ITS MODES 1, 2, 3 and 4. ITS 3.5.4 specifies requirements for ECCS in MODES 1, 2, 3 and 4. Therefore, this is an administrative change and is consistent with ISTS.

A14 Not used.

TECHNICAL CHANGES - MORE RESTRICTIVE

M1 CTS 3.3.1.1 requires the accumulators be OPERABLE whenever the reactor is critical. ITS 3.5.1 is applicable in MODES 1, 2 and MODE 3 with pressurizer pressure greater 1000 psig. In MODES 1 and 2, and in MODE 3 with RCS pressure > 1000 psig, the accumulator OPERABILITY requirements are based on full power operation. Although cooling requirements decrease as power decreases, the accumulators provide core cooling as long as elevated RCS pressures are greater than 1000 psig and temperatures exist. This change is an additional restriction on plant operation and is consistent with NUREG-1431.

M2 CTS 3.3.1.1.g specifies the requirement that motor operated valves (MOV) 865A, B and C have their power removed with the valves in the specified (open) position but does not require a periodic verification. ITS SR 3.5.1.1 requires a verification of valve position once prior to removing power from the valve. This verification ensures that the accumulators are available for injection. If an isolation valve is not fully open, the rate of injection to the RCS would be reduced. Although a motor operated valve position should not change with power removed, a closed valve could result in not meeting accident analyses assumptions. This Frequency is considered reasonable in view of other administrative controls that ensure a mispositioned isolation valve is unlikely. SR 3.5.1.5 requires verifying that power is removed every 31 days. Verification every 31 days that control power is removed from each accumulator isolation valve operator ensures that an active failure could not result in the undetected closure of an accumulator motor operated isolation valve. If this were to occur, only one accumulator would be available for injection given a single failure coincident with a LOCA. Since power is removed under administrative control, the 31 day Frequency provides adequate assurance that power is removed. These changes are additional restrictions on plant operation and are consistent with NUREG-1431.

M3 CTS 3.3.1.1.b specifies the requirements for a minimum accumulator cover pressure and contained borated water volume but does not require a periodic verification. ITS SR 3.5.1.2 and SR 3.5.1.3 require a verification of these parameters every 12 hours. This Frequency is sufficient to ensure adequate injection during a LOCA. Because of the static design of the accumulator, a 12 hour Frequency usually allows the operator to identify changes before limits are reached. Operating experience has shown this Frequency to be appropriate for early

detection and correction of off normal trends. These changes are an additional restriction on plant operation and are consistent with NUREG-1431.

M4 With one accumulator inoperable for greater than four hours, CTS 3.3.1.2 requires the unit be placed in hot shutdown but does not explicitly specify a time period. ITS 3.5.1 RA D.1 requires the unit be placed in MODE 3 within six 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. This change is an additional restriction on plant operation and is consistent with NUREG-1431.

M5 Although CTS 3.3.1.2 specifies the unit be placed in cold shutdown after achieving hot shutdown, this requirement is not applicable. Once the applicability of specification 3.3.1.1 (reactor critical) is exited, no further action is required. ITS 3.5.1 RA D.2 requires depressurization to ≤ 1000 psig consistent with the increased overall applicability of specification 3.5.1 (see DOC M1). Although cooling requirements decrease as power decreases, the accumulators provide core cooling as long as elevated RCS pressures are greater than 1000 psig and temperatures exist. This change is an additional restriction on plant operation and is consistent with NUREG-1431.

M6 Not used.

M7 With two or more accumulators inoperable, CTS 3.0 requires the plant to be placed in hot shutdown within 8 hours. ITS 3.5.1 RA E.1 requires an immediate entry into LCO 3.0.3. ITS LCO 3.0.3 requires the plant be in MODE 3 within 7 hours If more than one accumulator is inoperable, the plant is in a condition outside the accident analyses; therefore, LCO 3.0.3 must be entered immediately. This change is an additional restriction on plant operation and is consistent with NUREG-1431.

M8 CTS Table 4.1.2, Item 6 requires a sample of accumulator boron concentration every month. ITS SR 3.5.1.4 requires the surveillance to be performed monthly and once within six hours after a solution volume increase of ≥ 70 gallons that is not from the RWST. Sampling the affected accumulator within 6 hours after a ≥ 70 gallon volume increase will identify whether in leakage has caused a reduction in boron concentration. The 70 gallon volume increase and time limit of 6 hours is based on preventing a reduction in boron concentration in an accumulator below 1950 ppm with an initial boron concentration of 2000 ppm assuming in-leakage of 70 gallons pure water at a maximum in-leakage rate of 0.2 gpm. This event specific surveillance requirement is an additional restriction on plant operation and is consistent with NUREG-1431.

M9 CTS Table 4.1.2, Item 6 requires sampling of boron concentration. CTS 3.3.1.1.b specifies the lower limit on boron concentration but does not include an upper limit. ITS SR 3.5.1.4 requires verification that boron concentration is above the lower limit and below the upper limit. The maximum boron concentration is important since it is an assumption used in determining the cold leg to hot leg recirculation injection switch over time and minimum sump pH. This change is an additional restriction on plant operation and is consistent with NUREG-1431.

M10 CTS Table 4.1.2, Item 6 permits a maximum time between tests of 45 days. ITS SR 3.5.1.4 has a maximum interval of ≈ 39 days (monthly x 1.25). The 31 day Frequency is adequate to identify changes that could occur from mechanisms such as stratification or in leakage. The slight reduction in maximum surveillance interval does not impose a significant impact upon HBR operation. This change is an additional restriction on plant operation and is consistent with NUREG-1431.

M11 CTS does not include a surveillance comparable to ITS SR 3.5.2.6 (required on MODES 1, 2 and 3). Additionally, ITS SR 3.5.3.1 requires performance of ITS SR 3.5.2.6 in MODE 4. Periodic inspections of the containment sump suction inlet ensure that it is unrestricted and stays in proper operating condition. The 18 month Frequency is based on the need to perform this Surveillance under the conditions that apply during a plant outage, on the need to have access to the location, and because of the potential for an unplanned transient if the Surveillance were performed with the reactor at power. This Frequency has been found to be sufficient to detect abnormal degradation and is confirmed by operating experience. These changes are additional restrictions on plant operation and are consistent with NUREG-1431.

M12 CTS does not include a surveillance comparable to ITS 3.5.2.7. Verification of proper valve position ensures the proper flow path is established for the LHSI system following operation in RHR mode. The Frequency of 31 days is commensurate with the accessibility and radiation levels involved in performing the surveillance. This change is an additional restriction on plant operation and is consistent with NUREG-1431.

M13 CTS 4.5.1.1 requires performance of the safety injection tests at each reactor refueling interval. CTS does not explicitly limit the refueling interval to a finite time period. ITS SR 3.5.2.4 requires performance at an 18 month interval. The 18 month Frequency is based on the need to perform these Surveillances under the conditions that apply during a plant outage and the potential for unplanned plant transients if the Surveillances were performed with the reactor at power. The 18 month Frequency is also acceptable based on consideration of the design reliability (and confirming operating experience) of the equipment.

This change is an additional restriction on plant operation and is consistent with NUREG-1431.

M14 CTS 4.5.1.2 requires the Safety Injection System tests verify the pump breakers close. ITS SR 3.5.2.5 requires verification the pumps start. Verification of pump starting is important to properly test the train. This change is an additional restriction on plant operation and is consistent with NUREG-1431.

M15 Although CTS 4.5.2.1 does not include MOVs SI-878A and SI-878B in the periodic surveillance requirement, these valves are required to be in the position specified in CTS 3.3.1.1.g with AC control power removed. These valves are included in ITS SR 3.5.2.1, since these valves are similarly required to be de-energized in the specified position for the ECCS trains to be OPERABLE. This change is an additional restriction on plant operation and is consistent with NUREG-1431.

M16 Not used.

M17 CTS actions comparable to ITS 3.5.3 RA B.1 and RA C.1 do not exist. With less than one ECCS train OPERABLE, entry into CTS 3.0 is required which requires the unit be placed in Cold Shutdown within 30 hours. With the required ECCS high head subsystem inoperable, ITS 3.5.3 RA B.1 requires restoring one subsystem to OPERABLE status within one hour. With no ECCS high head subsystem OPERABLE, due to the inoperability of the safety injection train or flow path from the RWST, the plant is not prepared to provide high pressure response to Design Basis Events requiring SI. The 1 hour Completion Time to restore at least one ECCS high head subsystem to OPERABLE status ensures that prompt action is taken to provide the required cooling capacity or to initiate actions to place the plant in MODE 5, where an ECCS train is not required. If the RA and associated completion time of RA B.1 are not met, ITS 3.5.3 RA C.1 requires the unit be placed in MODE 5 within 24 hours. When the Required Actions of Condition B cannot be completed within the required Completion Time, a controlled shutdown must be initiated to place the Unit in a Condition outside the Applicability for the Specification . Twenty-four hours is a reasonable time, based on operating experience, to reach MODE 5 in an orderly manner and without challenging plant systems or operators. These change are an additional restriction on plant operation and are consistent with NUREG-1431.

M18 With the RWST not within limits, CTS required action is specified in 3.0. CTS 3.0 requires achieving hot shutdown within eight hours, followed by cold shutdown within an additional 30 hours. ITS 3.5.4 RA B.1 requires restoring RWST to OPERABLE status within one hour. In this Condition, neither the ECCS nor the Containment Spray System can perform its design function. Therefore, prompt action must be taken to restore the tank to OPERABLE status or to place the plant in a MODE in

which the RWST is not required. The short time limit of 1 hour to restore the RWST to OPERABLE status is based on this condition simultaneously affecting redundant trains. With required action and associated completion time not met, ITS RA C.1 and C.2 requires achieving MODE 3 within 6 hours, and MODE 5 within 36 hours. If the RWST cannot be returned to OPERABLE status within the associated Completion Time, the plant must be brought to a MODE in which the LCO does not apply. 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. This change is an additional restriction on plant operation and is consistent with NUREG-1431.

M19 A CTS surveillance requirement comparable to ITS SR 3.5.4.1 does not exist nor does CTS include limits on RWST temperature. ITS SR 3.5.4.1 requires periodic verification that the RWST is within specified temperature limits. The RWST borated water temperature is verified every 24 hours to be within the limits assumed in the accident analyses band. This Frequency is sufficient to identify a temperature change that would approach either limit and is acceptable based on operating experience. ITS 3.5.4 Condition A with the associated Required Action and Completion time impose restrictions on operation with the RWST outside the specified limits. With RWST boron concentration or borated water temperature not within limits, they must be returned to within limits within 8 hours. Under these conditions neither the ECCS nor the Containment Spray System can perform its design function. Therefore, prompt action must be taken to restore the tank to OPERABLE condition. The 8 hour limit to restore the RWST temperature or boron concentration to within limits considers the time required to change either the boron concentration or temperature and the fact that the contents of the tank are still available for injection.

CTS 3.3.1.1.a specifies the requirements for RWST contained borated water volume but does not require a periodic verification. ITS SR 3.5.4.2 requires a verification of this parameter every 7 days. The RWST water volume should be verified every 7 days to be above the required minimum level in order to ensure that a sufficient initial supply is available for injection and to support continued ECCS and Containment Spray System pump operation on recirculation. Since the RWST volume is normally stable and is protected by an alarm, a 7 day Frequency is appropriate and is acceptable based upon operating experience. These changes are additional restrictions on plant operation and are consistent with NUREG-1431.

M20 CTS Table 4.1.2, Item 3 permits a maximum interval between test of 10 days. ITS SR 3.5.4.3 has a maximum interval of ≈ 9 days (7 days x 1.25). The ITS maximum SR interval is not a significant impact on plant operations and reflects a consistent approach to maximum

SR intervals. This change is an additional restriction on plant operation and is consistent with NUREG-1431.

M21 CTS does not currently place a requirement on the maximum boron concentration in the RWST. ITS SR 3.5.4.3 imposes an upper limit. The RWST upper limit assures that the resulting sump pH will be maintained in an acceptable range so that boron precipitation in the core will not occur and the effect of chloride and caustic stress corrosion on mechanical systems and components will be minimized. This change is an additional restriction on plant operation and is consistent with NUREG-1431.

M22 CTS 3.3.1.1.g requires that control power be removed from the specified valves at > 1000 psig. CTS 3.3.1.1.h requires that air be removed from the specified valves at > 1000 psig. ITS SR 3.5.2.1 and ITS SR 3.5.2.7 require motive power be removed from the valves in MODES 1, 2 and 3. Although not directly comparable, the CTS specified applicability of > 1000 psig normally occurs significantly above the MODE 3 lower temperature limits. Consistent with NUREG-1431 construction, SRs are generally applicable when the Specification is applicable. In MODES 1, 2, and 3, the ECCS OPERABILITY requirements for the limiting Design Basis Accident, a large break LOCA, are based on full power operation. Although reduced power would not require the same level of performance, the accident analysis does not provide for reduced cooling requirements in the lower MODES. The SI pump performance requirements are based on a small break LOCA. MODE 2 and MODE 3 requirements are bounded by the MODE 1 analysis. These changes are additional restrictions on plant operation and are consistent with NUREG-1431.

M23 CTS 3.3.1.1.e allows any one flow path including valves of the safety injection or residual heat removal system to be inoperable for up to 24 hours. ITS LCO 3.5.2 Applicability Note 1 permits, in MODE 3, both safety injection (SI) pump flow paths to be isolated by closing the isolation valves for up to 12 hours to perform pressure isolation valve testing per SR 3.4.14.1. The procedure for performing pressure isolation valve testing is being revised to implement the ITS requirements. In so doing, portions of the test that previously caused inoperability of the accumulators will now be performed in a plant condition in which operability of the accumulators is not required (i.e., RCS pressure < 1000 psig). Therefore the required time to perform pressure isolation valve testing in plant configurations that result in portions of the ECCS being inoperable is reduced. Pressure isolation valve testing will require that each cold leg injection flow path be isolated for a period of time. Based upon previous plant experience in performing pressure isolation valve testing on the cold leg injection portion of the ECCS, the required time for Note 1 to ITS LCO 3.5.2 is 12 hours. This time is reasonable considering the time necessary to manipulate manual valves, establish stable pressure

conditions, and measure leakages for all three cold leg injection pressure isolation valves and includes a small margin of approximately 2 to 3 hours. This change is more restrictive, and has no adverse impact on safety.

TECHNICAL CHANGES - LESS RESTRICTIVE (GENERIC)

LA1 CTS 3.3.1.2.e explicitly excludes the SI hot leg pathways and valves from the requirements of the specification. This detail regarding applicability of the specification is relocated to the ITS bases.

The details associated with the involved Specifications are not required to be in the ITS to provide adequate protection of the public health and safety, since the ITS still retains the requirement for OPERABILITY of the ECCS. This approach provides an effective level of regulatory control and provides for a more appropriate change control process. The level of safety of facility operation is unaffected by the change because there is no change in the overall operational requirements. Furthermore, NRC and utility resources associated with processing license amendments to these requirements will be reduced. Therefore, relocation of these details is acceptable.

LA2 CTS 3.3.1.1 includes details regarding the equipment associated with OPERABLE ECCS trains. This requirement is relocated to the Bases for ITS 3.5.2 and ITS 3.5.3.

These details associated with the involved Specifications are not required to be in the ITS to provide adequate protection of the public health and safety, since the ITS still retains the requirement for OPERABILITY of the ECCS trains. This approach provides an effective level of regulatory control and provides for a more appropriate change control process. The level of safety of facility operation is unaffected by the change because there is no change in the overall operational requirements. Furthermore, NRC and utility resources associated with processing license amendments to these requirements will be reduced. Therefore, relocation of this information is acceptable.

TECHNICAL CHANGES - LESS RESTRICTIVE (SPECIFIC)

L1 During Power Operation CTS 3.3.1.2 permits one accumulator to be isolated or otherwise inoperable for up to four hours. ITS 3.5.1 RA A.1 permits one accumulator to be inoperable for boron concentration out of limits for 72 hours. Therefore, this is a less restrictive change and is consistent with NUREG-1431. The 72 hour Completion Time for restoration of the boron concentration to within limits is reasonable time to complete the Required Action including confirmatory sampling and analysis.

As stated in the bases, the boron in the accumulators contributes to the assumption that the combined ECCS water in the partially recovered core during the early reflooding phase of a large break LOCA is sufficient to keep that portion of the core subcritical. One accumulator below the minimum boron concentration limit, however, will have no effect on available ECCS water and an insignificant effect on core subcriticality during reflood. Boiling of ECCS water in the core during reflood concentrates boron in the saturated liquid that remains in the core. In addition, current analysis demonstrates that the accumulators do not discharge following a large main steam line break.

The magnitude of a potential boron reduction is limited because of the SR 3.5.1.4 requirement to sample boron concentration after a volume addition of 10% of the indicated tank level that is not from the RWST. This ensures that a boron reduction below the lower limit is promptly identified and the magnitude of the change is limited.

L2 CTS 3.3.1.2 permits 24 hours to restore specified components/flowpaths to operable status or to be in hot shutdown. If components are not restored within an additional 48 hours, CTS 3.3.1.3 requires the unit be placed in Cold Shutdown. ITS 3.5.2 RA A.1 permits an ECCS train to be inoperable for 72 hours in MODES 1, 2 and 3 provided at least 100% of the ECCS flow equivalent to a single OPERABLE ECCS train is available. If the required actions or completion times of ITS 3.5.2 RA A.1 are not met, ITS 3.5.2 RA C.1 and RA C.2 require the unit to be in MODE 3 within 6 hours and MODE 4 within 12 hours respectively.

The allowance for more than one train to be inoperable, provided an equivalent 100% ECCS flow capability exists, as well as the increase in the allowable out of service time from 24 hours to 72 hours are less restrictive requirements upon unit operation. Additionally, ITS 3.5.2 RA C.1 and C.2 provide 6 hours to be in MODE 3 and 12 hours to be in MODE 4 respectively, in addition to the 72 hours allowable out of service time. Therefore, these are less restrictive changes and are consistent with NUREG-1431.

Due to the redundancy of trains and the diversity of subsystems, the inoperability of one component in a train does not render the ECCS incapable of performing its function. Neither does the inoperability of two different components, each in a different train, necessarily result in a loss of function for the ECCS. The intent of this Condition is to maintain a combination of equipment such that 100% of the ECCS flow equivalent to a single OPERABLE ECCS train remains available. This allows increased flexibility in plant operations under circumstances when components in different trains are inoperable. The 72 hour Completion Time is based on an NRC reliability evaluation and is a reasonable time for repair of many ECCS components.

L3 CTS 4.5.1.1 requires the Safety Injection System tests be performed in such a manner to prevent injection into the reactor coolant system. This requirement is not retained in ITS. Although it is expected future testing will be consistent with current methodology which does not result in actual injection, testing which results in actual injection is acceptable since the test would still demonstrate acceptable system operation. Therefore, this is a less restrictive change and is consistent with NUREG-1431.

L4 CTS 4.5.1.2 specifies certain details regarding test method (e.g., control board indications and visual observation, proper sequence and timing, etc.) regarding acceptable Safety Injection System test results. These test method details are not retained in ITS. The ITS specified verification of pump starts and valve actuations is sufficient to demonstrate OPERABILITY. This change allows increased flexibility in testing methodology while still requiring verification of OPERABILITY. Therefore, this is a less restrictive change and is consistent with NUREG-1431.

L5 CTS 4.5.2.1 mandates a test method for the verification of the specified valve positions. Specifically, this specification requires verification ". . . from the RTGB indicators/controls . . ." that the specified valves are in the proper position with control power removed. This test method detail is not retained in ITS. The ITS specified verification that the valves are in their proper position with control power removed is sufficient to demonstrate OPERABILITY. This change allows increased flexibility in testing methodology while still requiring verification of OPERABILITY. Therefore, this is a less restrictive change and is consistent with NUREG-1431.

L6 In Hot Shutdown, CTS 3.3.1.3 imposes requirements for ECCS in accordance with CTS 3.3.1.1 and 3.3.1.2. CTS 3.3.1.1 requires two ECCS trains to be operable. CTS 3.3.1.2 permits one ECCS component (SI pump, RHR pump, RHR heat exchanger) to be inoperable for up to 24 hours. With less than one ECCS train OPERABLE, no specific action is provided. In this condition, entry into CTS 3.0 is required. CTS 3.0 requires the unit be placed in Cold Shutdown within 30 hours. ITS 3.5.3 requires only one ECCS train to be OPERABLE in MODE 4. With the required ECCS RHR subsystem inoperable, ITS 3.5.3 RA A.1 requires action be initiated immediately to restore one required ECCS RHR subsystem, to OPERABLE status. Therefore, this is a less restrictive change and is consistent with NUREG-1431.

Due to the stable conditions associated with operation in MODE 4 and the reduced probability of occurrence of a Design Basis Accident (DBA), the ECCS operational requirements are reduced. With both RHR pumps and heat exchangers inoperable, it would be unwise to require the plant to go to MODE 5, where the only available heat removal system is the RHR.

Therefore, the appropriate action is to initiate measures to restore one ECCS RHR subsystem and to continue the actions until the subsystem is restored to OPERABLE status.

L7 With the RWST boron concentration not within limits, CTS required action is specified in 3.0. CTS 3.0 requires achieving hot shutdown within 8 hours, followed by cold shutdown within an additional 30 hours. ITS 3.5.4 RA A.1 permits 8 hours to restore the RWST to OPERABLE status. With required action and associated completion time not met, ITS RA C.1 and C.2 requires achieving MODE 3 within 6 hours, and MODE 5 within 36 hours. Therefore, this aspect of the change is less restrictive change and is consistent with NUREG-1431.

The 8 hours to restore the boron concentration to within limits is acceptable based upon consideration of the time required to change the boron concentration and the fact that the contents of the tank are still available for injection. Permitting prompt corrective action to restore the boron concentration to within limits is preferable to requiring immediate plant shutdown, with its increased risk for shutdown transients.

L8 A CTS provision comparable to Note 2 to Applicability to ITS Specification 3.5.2 does not exist. This Note permits one ECCS train to be inoperable for up to four hours after entry into MODE 3 or until the RCS cold leg temperatures exceed 375°F, whichever comes first. Operation in MODE 3 with ECCS trains declared inoperable pursuant to LCO 3.4.12, "Low Temperature Overpressure Protection (LTOP) System," is necessary for plants with an LTOP arming temperature at or near the MODE 3 boundary temperature of 350°F. LCO 3.4.12 requires that certain pumps be rendered inoperable at and below the LTOP arming temperature. When this temperature is at or near the MODE 3 boundary temperature, time is needed to restore the inoperable pumps to OPERABLE status. This Note permits entry into MODE 3 without first meeting the LCO requirements. The limitations imposed on duration and cold leg temperatures are bounded by the 72 hours permitted by ITS 3.5.2 RA A.1 for one ECCS train being inoperable when in MODES 1, 2 and 3. Therefore, this is a less restrictive change on plant operation and is consistent with NUREG-1431.

TECHNICAL CHANGES - RELOCATED SPECIFICATIONS

None

3.5 EMERGENCY CORE COOLING SYSTEMS (ECCS)

CTS

3.5.1 Accumulators

[3.3.1.1.b]

LCO 3.5.1 Three ECCS accumulators shall be OPERABLE.

[3.3.1.1] APPLICABILITY: MODES 1 and 2.
MODE 3 with pressurizer pressure > 1000 psig.

ACTIONS

	CONDITION	REQUIRED ACTION	COMPLETION TIME
[L1]	A. One accumulator inoperable due to boron concentration not within limits.	A.1 Restore boron concentration to within limits	72 hours (Insert 3.5.1-1a)
[3.3.1.2.a]	B. C. One accumulator inoperable for reasons other than Condition A.	B.1 C.1 Restore accumulator to OPERABLE status	1 hour 4
[3.3.1.2]	D. Required Action and associated Completion Time of Condition A or B not met.	D.1 Be in MODE 3. AND D.2 Reduce pressurizer pressure to ≤ 1000 psig	6 hours 12 hours
[M7]	E. Two or more accumulators inoperable.	E.1 Enter LCO 3.0.3	Immediately

Insert 3.5.1-1a

B. One valve identified in SR 3.5.1.5 with control power restored.	B.1	Verify control power or air is removed to all valves identified in SR 3.5.2.1, and SR 3.5.2.7.	Immediately
[A12]	AND		
	B.2	Remove control power to valve.	4 hours

CTS

SURVEILLANCE REQUIREMENTS

	SURVEILLANCE		FREQUENCY
[3.3.1.1.b] [3.3.1.1.g]	SR 3.5.1.1	Verify each accumulator isolation valve is fully open.	~~12 hours~~ once prior to removing power from the valve operator (2)
[3.3.1.1.b]	SR 3.5.1.2	Verify borated water volume in each accumulator is ≥ [~~7863 gallons (1%) and ≤ 8177 gallons (1%)~~] 225 ft³ and ≤ 841 ft³	12 hours
[3.3.1.1.b]	SR 3.5.1.3	Verify nitrogen cover pressure in each accumulator is ≥ [~~[illegible]~~] 600 psig and ≤ [~~[illegible]~~] 660 psig. (14)	12 hours
[3.3.1.1.b] [T4.1.2 Item 6]	SR 3.5.1.4	Verify boron concentration in each accumulator is ≥ [~~1900~~] 1950 ppm and ≤ [~~2100~~] 2400 ppm. (15)	31 days AND -----NOTE----- Only required to be performed for affected accumulators ---------- Once within 6 hours after each solution volume increase of ≥ [~~[illegible]~~] 70 gallons ~~[1% of indicated level]~~ that is not the result of addition from the refueling water storage tank

(continued)

Supplement 5

CTS

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
[M2] SR 3.5.1.5	Verify power is removed from each accumulator isolation valve operator ~~when pressurizer pressure ≥ [2000] psig~~ control	31 days

5

2

Supplement-5

Insert 3.5.1-1

Not used.

CTS

3.5 EMERGENCY CORE COOLING SYSTEMS (ECCS)

3.5.2 ECCS – Operating

[3.3.1.1.c, d, e, f]

LCO 3.5.2 Two ECCS trains shall be OPERABLE. 16

[3.3.1.1]

APPLICABILITY: MODES 1, 2, and 3. 4

12

Entry and

NOTES

1. In MODE 3, both safety injection (SI) pump flow paths may be isolated by closing the isolation valves for up to 2 hours to perform pressure isolation valve testing per SR 3.4.14.1. 1

2. Operation in MODE 3 with ECCS pumps declared inoperable pursuant to LCO 3.4.12, "Low Temperature Overpressure Protection (LTOP) System," is allowed for up to 4 hours or until the temperature of all RCS cold legs exceeds 375°F, whichever comes first. X

one required SI

[L 8]

[3.3.1.2.f]

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
[3.3.1.2.b] A. One or more trains inoperable. [3.3.1.2.c] AND At least 100% of the ECCS flow equivalent to a single OPERABLE ECCS train available.	A.1 Restore train(s) to OPERABLE status.	72 hours
[3.3.1.2] B. C. Required Action and associated Completion Time not met.	B.1 C.1 Be in MODE 3. AND B.2 C.2 Be in MODE 4.	6 hours 12 hours

20

INSERT 3.5.2-1a

Supplements

Insert 3.5.2-1

Not used.

Insert 3.5.2-1a

CTS

[A10]

B. One valve identified in SR 3.5.2.1 and SR 3.5.2.7 with control power or air restored.	B.1	Verify control power is removed to all valves identified in SR 3.5.1.5.	Immediately
	AND		
	B.2	Remove control power or air to valve.	24 hours

CTS

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.5.2.1	Verify the following valves are in the listed position with power to the valve operator removed. (Control)	12 hours (Low Head Safety Injection (LHSI)) (HIGH HEAD Safety Injection (HHSI))

[4.5.2.1] [3.3.1.1.g]

Number	Position	Function
SI 862A & B [ ]	[OPEN]	[ ]
SI 863A & B [ ]	[CLOSED]	[LHSI]
SI 864A & B	OPEN	LHSI,
SI 866A & B	CLOSED	HHSI
SI 878A & B [ ]	[OPEN]	[HHSI]

SR 3.5.2.2 [4.5.2.2]	Verify each ECCS manual, power operated, and automatic valve in the flow path, that is not locked, sealed, or otherwise secured in position, is in the correct position.	31 days
~~SR 3.5.2.3~~	~~Verify ECCS piping is full of water.~~	~~31 days~~
SR 3.5.2.~~4~~ 3 [A6]	Verify each ECCS pump's developed head at the test flow point is greater than or equal to the required developed head.	In accordance with the Inservice Testing Program
SR 3.5.2.~~5~~ 4 [4.5.1.1]	Verify each ECCS automatic valve in the flow path that is not locked, sealed, or otherwise secured in position, actuates to the correct position on an actual or simulated actuation signal.	~~18~~ months (INSERT 3.5.2-2)

(continued)

Supplement 5

Insert 3.5.2-2

----------NOTE----------
Alternate testing may be performed by setting the pump head at the baseline value and measuring the flow to meet the requirements of SR 3.5.2.3.

CTS

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.5.2.5 [4.5.1.2]	Verify each ECCS pump starts automatically on an actual or simulated actuation signal.	~~[18]~~ months
~~SR 3.5.2.7~~	~~Verify, for each ECCS throttle valve listed below, each position stop is in the correct position.~~ ~~Valve Number~~ ~~[ ] [ ] [ ]~~	~~[18] months~~
[M11] SR 3.5.2.6	Verify, by visual inspection, ~~each~~ the ECCS train containment sump suction inlet is not restricted by debris and the suction inlet trash racks and screens show no evidence of structural distress or abnormal corrosion.	~~[18]~~ months

18

[3.3.1.1.h] Insert 3.5.2-3 7

[M12] Insert 3.5.2-4 6

Insert 3.5.2-3

SR 3.5.2.7	Verify the following valves in the listed position:			31 days
	Number	Position	Function	
	FCV-605	Closed/Motive Air Isolated	RHR	
	HCV-758	Closed/Motive Air Isolated	RHR	

Insert 3.5.2-4

SR 3.5.2.8	Verify the following manual valve in the listed position:			92 days
	Number	Position	Function	
	RHR-764	Locked Open	LHSI	

CTS

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.5.3.1	----------NOTE---------- An RHR train may be considered OPERABLE during alignment and operation for decay heat removal, if capable of being manually realigned to the ECCS mode of operation ----------	
	The following SRs are applicable for all equipment required to be OPERABLE:	In accordance with applicable SRs
	12 ~~[SR 3.5.2.1]~~ ~~[SR 3.5.2.8]~~ SR 3.5.2.4 3 ~~[SR 3.5.2.7]~~ SR 3.5.2.8 6	

[A 12]

[M 11]

Supplement 5

1 LCO 3.4.12 requires one SI pump to be disabled.

2 SR 3.5.1.1 frequency is specified as "once prior to removing power from the valve operator." These valves are required to have their power removed. Removal of power disables remote indication of the valve's position. To preclude the need for routine entry into containment, SR 3.5.1.1 specifies an initial verification of valve's position to be performed prior to removal of power from the valve operator. The SR 3.5.1.1 requirement to verify the valves are open prior to removal of power to the valve's operator coupled with the SR 3.5.1.5 requirement to verify every 31 days that the valve's power is removed provides reasonable assurance that the valves remain open.

With control power removed, the accumulator discharge isolation valves do not provide a remote position indication and thus requiring an entry into containment to verify the valve's position. The remaining valves, except for 878A and 878B (SI pump discharge cross-connect valves) are designed to provide remote position indication when their control power is removed. The 878A and 878B valves, although without remote position indication, are located outside containment and are thus accessible for position verification by observation of the valves.

The phrase limiting the applicability of SR 3.5.1.5 to when pressure > 2000 psig is eliminated. The current licensing basis requires these valves have control power removed at > 1000 psig. Since this is consistent with the applicability for ITS 3.5.1, there is no need to specify the value in SR 3.5.1.5.

3 Consistent with the current licensing basis, an action was added to LCOs 3.5.1 and SR 3.5.2 to permit restoration of control power or air to one valve identified in SRs 3.5.1.5, 3.5.2.1, and 3.5.2.7. The provision to restore power or air to one valve for the specified time does not permit repositioning the valve. With power or air restored to an accumulator isolation valve (valve remains in open position), the accumulator remains OPERABLE. In this circumstance, Condition B includes an allowed outage time equivalent to Condition C.1, but adds the requirement to immediately verify control power and air removed from all other valves identified in the SRs. This Required Action retains the CLB which permits restoring power to an accumulator valve for testing or maintenance without rendering the associated accumulator inoperable.

Since only one of the specified valves is permitted to have air or power restored, the Required Action is structured consistently in LCOs 3.5.1 and 3.5.2.

4 The HBRSEP design is not conducive to performing PIV testing, requiring up to 12 hours to complete the testing. Note 1 to 3.5.2 Applicability is modified to reflect this change. PIV testing involves isolating each

cold leg injection flow path, one at a time, from a common Safety Injection (SI) header to the respective RCS loop. During the period that the flow path is isolated, flow is only available to two of the three RCS loops. If a LOCA were to occur, 100% of the required SI flow would not be assured if the break occurs on one of the two available loops. Based upon previous plant experience in performing pressure isolation valve testing on the cold leg injection portion of the ECCS, the required time to perform pressure isolation valve testing in plant configurations that result in portions of the ECCS being inoperable is 12 hours. This time includes the time necessary to manipulate manual valves, establish stable pressure conditions, and measure leakages for all three cold leg injection pathway pressure isolation valves and includes a small margin of approximately 2 to 3 hours.

5 SR 3.5.1.5 and SR 3.5.2.1 are modified to specify control power removed from the valve operators consistent with the current licensing basis.

6 SR 3.5.2.8 is added to require surveillance of manual valve RHR-764. This surveillance is similar to ITS SR 3.5.2.1 which is performed for motor operated valves.

7 SR 3.5.2.7 is added to require surveillance of air operated valves FCV-605 and HCV-758. This surveillance is similar to ITS SR 3.5.2.1 which is performed for motor operated valves.

8 SR 3.5.2.2 is added to the SR list in SR 3.5.3.1 since it supports OPERABILITY in MODE 4.

9 The specification for Seal Injection Flow is not applicable to HBRSEP since the charging pumps are not used for safety injection.

10 The specification for Boron Injection Tank (BIT) is not applicable to HBRSEP. The BIT does not contain concentrated boric acid at HBRSEP.

11 Consistent with the current licensing basis, the four hour time limit for an inoperable accumulator is retained in the ITS. The four hour period provides a reasonable, although still limited, interval to restore the accumulator to OPERABLE status prior to requiring entry into Condition C.

12 Consistent with the current licensing basis (CLB), ISTS SR 3.5.2.1 is not applicable in MODE 4. The CLB for the valves in SR 3.5.2.1 requires the valves be deenergized in specified positions when reactor pressure is greater than 1000 psig. During a normal plant heatup or cooldown, RCS temperature is well above the upper MODE 4 temperature when RCS pressure is 1000 psig.

13 Not used.

14 The ISTS SR 3.5.1.3 upper limit for accumulator overpressure is modified to reflect the plant specific value. The specified upper limit is the value used in the plant specific ECCS analysis.

15 The ISTS SR 3.5.1.4 upper limit for accumulator boron concentration is modified to reflect the plant specific value. The specified upper limit is the value used in the plant specific ECCS analysis.

16 Note 2 to the Applicability to ISTS 3.5.2 is modified to permit entry into Mode 3 as well as operation in MODE 3. At HBRSEP the Applicability for LTOP is the same as the ITS Mode 3 to 4 transition temperature. When in the applicability for LTOP, two ECCS trains cannot be OPERABLE. The addition of the Words "Entry and" permits entry into MODE 3 where the requirements of LTOP do not apply, thus permitting restoring the second ECCS train to OPERABLE status.

17 The NUREG bracketed requirement to verify the ECCS piping is full of water is not a CLB requirement for HBRSEP. The HBRSEP Unit No. 2 design does not provide the capability to perform this SR. The SI System does not include vents at system high points which are necessary to perform this SR.

Changes to the CTS ECCS requirements were approved without imposing this additional requirement by issuance of amendments 119 and 153 on 6/20/88 and 11/21/94 respectively.

18 The CTS does not contain a requirement comparable to the bracketed ISTS SR 3.5.2.7 requirement to verify ECCS throttle valve positions. HBRSEP Unit No. 2 design does not utilize ECCS throttle valves with position stops (other than fully open and fully closed) for the purpose of throttling SI flow to the RCS loop with the LOCA break location as provided in the reference ISTS plant design.

19 The ISTS SR 3.5.1.4 upper limit for RWST boron concentration is modified to reflect the plant specific value. The specified upper limit is the value used in the plant specific accident and transient analysis.

20 Consistent with the current licensing basis, Required Action B.1 to verify that valves listed in SR 3.5.1.5 have control power removed immediately and Required Action B.2 to remove control power from the single valve listed in either SR 3.5.2.1 or SR 3.5.2.8 within a Completion Time of 4 hours, are added.

The specified valves are required to be deenergized in the specified position to preclude a single failure (including spurious actuation) from adversely affecting the capability of the ECCS. Since the design of the HBRSEP does not include fully independent trains, some piping

and valves are shared by both ECCS trains. Since the specified valves affect both trains, the provisions afforded by the ITS Actions for a single train are not applicable. Therefore the provisions afforded by Condition B are added to retain the CLB which permits restoring power or air to one valve among the SRs listed for maintenance or testing.

Since only one of the specified valves is permitted to have air or power restored, the Required Actions in Condition B are structured consistently with Required Actions B.1 and B.2 of LCO 3.5.1.

21 ISTS SR 3.5.2.4 requires testing by measuring the pump developed head at only one point of the pump characteristic curve by setting the flow and measuring the pump developed head. At HBRSEP, Unit No. 2,, no capability exists in MODES 1, 2, 3, or 4 to set the pump flow at a flow rate substantial enough to permit measurement of the developed head as a variable. The ASME Boiler & Pressure Vessel (B&PV) Code allows alternately to set the head at a baseline value and measure flow to determine measured pump performance to within an acceptable tolerance. This is the test method employed at HBRSEP, Unit No. 2 during the applicable MODES for ECCS. Therefore, the ITS SR 3.5.2.3 includes a NOTE to allow the alternate testing method in accordance with the ASME B&PV Code.

BASES

ACTIONS

A.1 (continued)

reduced. The boron in the accumulators contributes to the assumption that the combined ECCS water in the partially recovered core during the early reflooding phase of a large break LOCA is sufficient to keep that portion of the core subcritical. One accumulator below the minimum boron concentration limit, however, will have no effect on available ECCS water and an insignificant effect on core subcriticality during reflood. Boiling of ECCS water in the core during reflood concentrates boron in the saturated liquid that remains in the core. In addition, current analysis techniques demonstrate that the accumulators do not discharge following a large main steam line break. ~~for the majority of plants. Even if they do discharge their impact is minor and not a design limiting event.~~ Thus, 72 hours is allowed to return the boron concentration to within limits.

INSERT B 3.5.1-4

B.1

If one accumulator is inoperable for a reason other than boron concentration, the accumulator must be returned to OPERABLE status within 1 hour. In this Condition, the required contents of three accumulators cannot be assumed to reach the core during a LOCA. Due to the severity of the consequences should a LOCA occur in these conditions, the 1 hour Completion Time to open the valve, remove power to the valve, or restore the proper water volume or nitrogen cover pressure ensures that prompt action will be taken to return the inoperable accumulator to OPERABLE status. The Completion Time minimizes the potential for exposure of the plant to a LOCA under these conditions.

C.1 and C.2 D.1 and D.2

If the accumulator cannot be returned to OPERABLE status within the associated Completion Time, the plant must be brought to a MODE in which the LCO does not apply. To achieve this status, the plant must be brought to MODE 3 within 6 hours and pressurizer pressure reduced to

(continued)

Supplement 5

Insert 3.5.1-4

B.1 and B.2

If control power is restored to one valve identified in SR 3.5.1.5, immediate verification must be performed that no other valves listed in SR 3,5,2.1, and SR 3.5.2.7 have the control power or air restored. Additionally, Required Action B.2 requires the control power to be removed to the valve within 4 hours. In this condition, the valves could be subject to a spurious single failure that could result in closure of the valve and isolation of an accumulator. During the interval in which control power is restored, the valve remains in its required position The 4 hour Completion Time is reasonable considering a low probability of a spurious single failure coincident with a LOCA and is consistent with the 4 hour allowed outage time for one accumulator.

BASES

ACTIONS

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

≤ 1000 psig within 12 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.

~~D.1~~ E.1

If more than one accumulator is inoperable, the plant is in a condition outside the accident analyses; therefore, LCO 3.0.3 must be entered immediately.

SURVEILLANCE REQUIREMENTS

SR 3.5.1.1

prior to removing power from the operator

isolation valve

Each accumulator valve should be verified to be fully open ~~every 12 hours~~. This verification ensures that the accumulators are available for injection and ensures timely discovery if a valve should be less than fully open. If an isolation valve is not fully open, the rate of injection to the RCS would be reduced. Although a motor operated valve position should not change with power removed, a closed valve could result in not meeting accident analyses assumptions. This Frequency is considered reasonable in view of other administrative controls that ensure a mispositioned isolation valve is unlikely.

SR 3.5.1.2 and SR 3.5.1.3

Every 12 hours, borated water volume and nitrogen cover pressure are verified for each accumulator. This Frequency is sufficient to ensure adequate injection during a LOCA. Because of the static design of the accumulator, a 12 hour Frequency usually allows the operator to identify changes before limits are reached. Operating experience has shown this Frequency to be appropriate for early detection and correction of off normal trends.

(continued)

Supplement 5

Insert 3.5.1-4

B.1 and B.2

If control power is restored to one valve identified in SR 3.5.1.5, immediate verification must be performed that no other valves listed in SR 3.5.2.1, and SR 3.5.2.7 have the control power or air restored. Additionally, Required Action B.2 requires the control power to be removed to the valve within 4 hours. In this condition, the valves could be subject to a spurious single failure that could result in closure of the valve and isolation of an accumulator. During the interval in which control power is restored, the valve remains in its required position The 4 hour Completion Time is reasonable considering a low probability of a spurious single failure coincident with a LOCA and is consistent with the 4 hour allowed outage time for one accumulator.

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.5.1.4

The boron concentration should be verified to be within required limits for each accumulator every 31 days since the static design of the accumulators limits the ways in which the concentration can be changed. The 31 day Frequency is adequate to identify changes that could occur from mechanisms such as stratification or inleakage. Sampling the affected accumulator within 6 hours after a 1% volume increase will identify whether inleakage has caused a reduction in boron concentration to below the required limit. It is not necessary to verify boron concentration if the added water inventory is from the refueling water storage tank (RWST), because the water contained in the RWST is within the accumulator boron concentration requirements. This is consistent with the recommendation of NUREG-1366 (Ref. 5).

SR 3.5.1.5

Verification every 31 days that power is removed from each accumulator isolation valve operator when the pressurizer pressure is ≥ 2000 psig ensures that an active failure could not result in the undetected closure of an accumulator motor operated isolation valve. If this were to occur, only two accumulators would be available for injection given a single failure coincident with a LOCA. Since power is removed under administrative control, the 31 day Frequency will provide adequate assurance that power is removed.

This SR allows power to be supplied to the motor operated isolation valves when pressurizer pressure is < 2000 psig, thus allowing operational flexibility by avoiding unnecessary delays to manipulate the breakers during plant startups or shutdowns. Even with power supplied to the valves, inadvertent closure is prevented by the RCS pressure interlock associated with the valves.

Should closure of a valve occur in spite of the interlock, the SI signal provided to the valves would open a closed valve in the event of a LOCA.

(continued)

Supplement 5

Insert B3.5.1-3

The 70 gallon volume increase and time limit of 6 hours is based on preventing a reduction in boron concentration in an accumulator below 1950 ppm with an initial boron concentration of 2000 ppm assuming in-leakage of 70 gallons pure water at a maximum in-leakage rate of 0.2 gpm.

Insert B3.5.1-4

Not used.

BASES (continued)

APPLICABILITY

In MODES 1, 2, and 3, the ECCS OPERABILITY requirements for the limiting Design Basis Accident, a large break LOCA, are based on full power operation. Although reduced power would not require the same level of performance, the accident analysis does not provide for reduced cooling requirements in the lower MODES. ~~The centrifugal charging pump performance is based on a small break LOCA, which establishes the pump performance curve and has less dependence on power.~~ The SI pump performance requirements are based on a small break LOCA. MODE 2 and MODE 3 requirements are bounded by the MODE 1 analysis.

~~This LCO is only applicable in MODE 3 and above. Below MODE 3, the SI signal setpoint is manually bypassed by operator control, and system functional requirements are relaxed as described in LCO 3.5.3, "ECCS - Shutdown."~~

As indicated in Note 1, the flow path may be isolated for 8 hours in MODE 3, under controlled conditions, to perform pressure isolation valve testing per SR 3.4.14.1. ~~The flow path is readily restorable from the control room.~~

As indicated in Note 2, operation in MODE 3 with ECCS trains declared inoperable pursuant to LCO 3.4.12, "Low Temperature Overpressure Protection (LTOP) System," is necessary ~~for plants~~ with an LTOP arming temperature at or near the MODE 3 boundary temperature of 350°F. LCO 3.4.12 requires that certain pumps be rendered inoperable at and below the LTOP arming temperature. ~~When~~ Since this temperature is at or near the MODE 3 boundary temperature, time is needed to restore the inoperable pumps to OPERABLE status.

In MODES 5 and 6, plant conditions are such that the probability of an event requiring ECCS injection is extremely low. Core cooling requirements in MODE 5 are addressed by LCO 3.4.7, "RCS Loops - MODE 5, Loops Filled," and LCO 3.4.8, "RCS Loops - MODE 5, Loops Not Filled." MODE 6 core cooling requirements are addressed by LCO 3.9.5, "Residual Heat Removal (RHR) and Coolant Circulation - High Water Level," and LCO 3.9.6, "Residual Heat Removal (RHR) and Coolant Circulation - Low Water Level."

(continued)

Supplement 5

BASES (continued)

ACTIONS

A.1

With one or more trains inoperable and at least 100% of the ECCS flow equivalent to a single OPERABLE ECCS train available, the inoperable components must be returned to OPERABLE status within 72 hours. The 72 hour Completion Time is based on an NRC reliability evaluation (Ref. 5) and is a reasonable time for repair of many ECCS components.

An ECCS train is inoperable if it is not capable of delivering design flow to the RCS. Individual components are inoperable if they are not capable of performing their design function or supporting systems are not available.

The LCO requires the OPERABILITY of a number of ~~independent~~ active subsystems. Due to the redundancy of trains and the diversity of subsystems, the inoperability of one component in a train does not render the ECCS incapable of performing its function. Neither does the inoperability of two different components, each in a different train, necessarily result in a loss of function for the ECCS. The intent of this Condition is to maintain a combination of equipment such that 100% of the ECCS flow equivalent to a single OPERABLE ECCS train remains available. This allows increased flexibility in plant operations under circumstances when components in opposite trains are inoperable.

An event accompanied by a loss of offsite power and the failure of an EDG can disable one ECCS train until power is restored. A reliability analysis (Ref. 5) has shown that the impact of having one full ECCS train inoperable is sufficiently small to justify continued operation for 72 hours.

Reference 6 describes situations in which one component, such as an RHR crossover valve, can disable both ECCS trains. With one or more component(s) inoperable such that 100% of the flow equivalent to a single OPERABLE ECCS train is not available, the facility is in a condition outside the accident analysis. Therefore, LCO 3.0.3 must be immediately entered.

(continued)

Insert B3.5.2-4

Not used.

BASES

ACTIONS (continued)

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

Insert B 3.5.2-5

If the inoperable trains cannot be returned to OPERABLE status within the associated Completion Time, the plant must be brought to a MODE in which the LCO does not apply. To achieve this status, the plant must be brought to MODE 3 within 6 hours and MODE 4 within 12 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.5.2.1

Verification of proper valve position ensures that the flow path from the ECCS pumps to the RCS is maintained. Misalignment of these valves could render both ECCS trains inoperable. Securing these valves in position by removal of power or by key locking the control in the correct position ensures that they cannot change position as a result of an active failure or be inadvertently misaligned. These valves are of the type, described in Reference 6, that can disable the function of both ECCS trains and invalidate the accident analyses. A 12 hour Frequency is considered reasonable in view of other administrative controls that will ensure a mispositioned valve is unlikely.

control

SR 3.5.2.2

Verifying the correct alignment for manual, power operated, and automatic valves in the ECCS flow paths provides assurance that the proper flow paths will exist for ECCS 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 receives an actuation signal is allowed to be in a nonaccident position provided the valve will automatically reposition within the proper stroke time. This Surveillance does not require any testing or valve manipulation. Rather, it involves verification that those valves capable of being mispositioned are in the correct position. The 31 day Frequency is appropriate because the valves are operated

(continued)

Insert B3.5.2-5

B.1 and B.2

If control power or air is restored to one valve identified in SR 3.5.2.1 and SR 3.5.2.7, immediate verification must be performed that no other valves listed in SR 3.5.1.5 have the control power restored. Additionally, Required Action B.2 requires the control power to be removed to the valve within 24 hours. In this condition, the valves could be subject to a spurious single failure that could result in closure of the valve and isolation of an accumulator. During the interval in which control power is restored, the valve remains in its required position The 24 hour Completion Time is reasonable considering a low probability of a spurious single failure coincident with a LOCA.

BASES

SURVEILLANCE REQUIREMENTS

SR 3.5.2.2 (continued)

under administrative control, and an improper valve position would only affect a single train. This Frequency has been shown to be acceptable through operating experience.

8

~~SR 3.5.2.3~~

~~With the exception of the operating centrifugal charging pump, the ECCS pumps are normally in a standby, nonoperating mode. As such, flow path piping has the potential to develop voids and pockets of entrained gases. Maintaining the piping from the ECCS pumps to the RCS full of water ensures that the system will perform properly, injecting its full capacity into the RCS upon demand. This will also prevent water hammer, pump cavitation, and pumping of noncondensible gas (e.g., air, nitrogen, or hydrogen) into the reactor vessel following an SI signal or during shutdown cooling. The 31 day Frequency takes into consideration the gradual nature of gas accumulation in the ECCS piping and the procedural controls governing system operation.~~

SR 3.5.2.~~4~~ 3

Periodic surveillance testing of ECCS pumps to detect gross degradation caused by impeller structural damage or other hydraulic component problems is required by Section XI of the ASME Code. This type of testing may be accomplished by measuring the pump developed head at only one point of the pump characteristic curve. This verifies both that the measured performance is within an acceptable tolerance of the original pump baseline performance and that the performance at the test flow is greater than or equal to the performance assumed in the plant safety analysis. SRs are specified in the Inservice Testing Program, which encompasses Section XI of the ASME Code. Section XI of the ASME Code provides the activities and Frequencies necessary to satisfy the requirements.

35

Insert B 3.5.2-7

SR 3.5.2.~~5~~ 4 and SR 3.5.2.~~6~~ 5

These Surveillances demonstrate that each automatic ECCS valve actuates to the required position on an actual or

(continued)

Supplement 5

Insert B 3.5.2-7

Alternately, a Note to SR 3.5.2.3 permits pump performance verification by setting the pump head at the baseline value and measuring the test flow.

Insert B3.5.2-6

SR 3.5.2.7

Verification of proper valve position ensures the proper flow path is established for the LHSI system following operation in RHR mode. The Frequency of 31 days is commensurate with the accessibility and radiation levels involved in performing the surveillance (Ref. 6).

SR 3.5.2.8

Verification of proper valve position ensures the proper flow path is established for the Low Head Safety Injection (LHSI) system following operation in RHR mode. The Frequency of 92 days is based on the locked status for the valve as well as the accessibility and radiation levels involved in performing the surveillance (Ref. 6).

BASES

LCO (continued)

path capable of taking suction from the RWST and transferring suction to the containment sump.

The hot leg injection paths of the SI System, including valves, are not subject to the requirements of this specification

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

APPLICABILITY

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

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

In MODES 5 and 6, plant conditions are such that the probability of an event requiring ECCS injection is extremely low. Core cooling requirements in MODE 5 are addressed by LCO 3.4.7, "RCS Loops – MODE 5, Loops Filled," and LCO 3.4.8, "RCS Loops – MODE 5, Loops Not Filled." MODE 6 core cooling requirements are addressed by LCO 3.9.8, "Residual Heat Removal (RHR) and Coolant Circulation – High Water Level," and LCO 3.9.9, "Residual Heat Removal (RHR) and Coolant Circulation – Low Water Level."

ACTIONS

A.1

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

(continued)

1 In the conversion of the HBRSEP current Technical Specifications (CTS) to the proposed plant specific Improved Technical Specifications (ITS), certain wording preferences or conventions are adopted which do not result in technical changes (either actual or interpretational). Editorial changes or clarifications which involve the insertion of plant specific terms, parameters, or descriptions are used to preserve consistency with the CTS and licensing basis.

2 The HBRSEP design does not include the interlock for the accumulator motor operated isolation valves.

3 The HBRSEP design provides for three RCS loops and three accumulators.

4 The HBRSEP analysis does not include the additional 2 seconds for SI signal generation.

5 The bases are modified to reflect HBRSEP LOCA analysis methodology.

6 The HBRSEP analysis for large break LOCA assumes a reactor trip with rod insertion.

7 The HBRSEP analysis for main steam line break indicates the accumulators do not discharge.

8 The Bases is modified to agree with the applicable specification.

9 The HBRSEP design does not provide remote valve position indication when power is removed from the accumulator isolation valves. This precludes verification of valve position without entry into containment. The valves are verified open prior to removing power to the motor operator and the power is verified removed every 31 days.

10 A clarification is provided to explain the bases for the required boron concentration surveillance after the specified 70 gallon volume increase.

11 The HBRSEP design does not utilize centrifugal charging pumps. The charging pumps are of the positive displacement type and are not part of the ECCS. Plant specific terminology does not refer to the SI pumps as intermediate head pumps.

12 The HBRSEP design is redundant with respect to a single active failure. Additionally, the design utilizes some common piping between the RWST and ECCS pump suction piping.

13 Consistent with the current licensing basis, the accumulators are not required to be operable with RCS pressures < 1000 psig. The HBRSEP ECCS analysis does not include a specific analysis for events occurring at ≤ 1000 psig.

14 The HBRSEP design provides for splitting injection between the hot and cold legs simultaneously.

15 HBRSEP was designed and licensed to the proposed Appendix A to 10 CFR 50, which was published in the Federal Register on July 11, 1967 (32FR10213). Appendix A to 10 CFR 50 effective in 1971 and subsequently amended, is somewhat different from the proposed 1967 criteria. UFSAR section 3.1 includes an evaluation of HBRSEP with respect to the proposed 1967 criteria. The ISTS statement concerning the GDC criteria is modified in the ITS to reference the current licensing basis description in the UFSAR.

16 One charging pump is sufficient to maintain RCS inventory with break sizes up to 0.295 inch diameter. For a break up to this size, the steam generators continue to be used for heat removal.

17 HBRSEP design does not provide fully independent ECCS trains. HBRSEP design provides protection from a single active failure.

18 The bases are modified to clarify plant specific information regarding bypassing of some SI initiation circuitry during plant heatup and cooldown.

19 Not used.

20 Bases for added SR 3.5.2.7 and SR 3.5.2.8 are provided. Appropriate bases are included for the Note to SR 3.5.2.7.

21 Since one SI pump is required to be disabled when RCS temperature is ≤ 350°F, a clarification regarding restoration of power is added to the bases.

22 The HBRSEP design utilizes a common suction line from the RWST to the Safety Injection System, RHR System and Containment Spray System. The design provides two motor operated valves in series to isolate the RWST.

23 The HBRSEP ECCS design is based upon assumption of a single active failure. A passive failure is not considered either coincident or non-coincident with Design Basis Events.

24 The HBRSEP design does not utilize the charging pumps to provide injection during a LOCA event. The ECCS pumps are normally aligned to take suction from the RWST through two normally open motor operated valves.

25 The bases are modified to reflect HBRSEP analysis results. The maximum boron concentration is utilized in determining the minimum time to initiate hot leg injection during the recirculation phase of a LOCA response. The maximum RWST temperature is used in the containment

analysis for a main steam line break. The minimum RWST temperature is used in the containment analysis for inadvertent spray, ECCS backpressure for Loss of Coolant accidents and reactivity analysis for main steam line breaks.

26 The bases are modified to clarify that the RWST minimum volume assures long term cooling capability.

27 The bases are modified to clarify the impact of reduced containment pressure on ECCS performance.

28 The specification for Seal Injection Flow is not applicable to HBRSEP since the charging pumps are not used for safety injection.

29 The specification for Boron Injection Tank (BIT) is not applicable to HBRSEP. The BIT does not contain concentrated boric acid at HBRSEP.

30 The bases to Note 1 to 3.5.2 Applicability is modified to reflect this change. PIV testing involves isolating each cold leg injection flow path, one at a time, from a common Safety Injection (SI) header to the respective RCS loop. During the period that the flow path is isolated, flow is only available to two of the three RCS loops. If a LOCA were to occur, 100% of the required SI flow would not be assured if the break occurs on one of the two available loops. Based upon previous plant experience in performing pressure isolation valve testing on the cold leg injection portion of the ECCS, the required time to perform pressure isolation valve testing in plant configurations that result in portions of the ECCS being inoperable is 12 hours. This time includes the time necessary to manipulate manual valves, establish stable pressure conditions, and measure leakages for all three cold leg injection pathway pressure isolation valves and includes a small margin of approximately 2 to 3 hours. Additionally, the statement, "the flow path is readily restorable from the control room," is deleted from the ISTS bases because the valves utilized to isolate the SI cold leg injection headers are manual valves and portions of the system are depressurized through a vent during pressure isolation valve testing.

31 Appropriate bases are included for the Required Action B.1 to LCOs 3.5.1 and SR 3.5.2.

32 Clarification regarding application of the single failure criteria to the ECCS cold leg injection check valves is provided.

33 The HBRSEP analysis indicates a return to power is possible after a steam line break. Refer to UFSAR section 15.1.5.

34 The current licensing basis regarding non-applicability of this specification to the Safety Injection System hot leg pathways, including valves, is retained. The valves in the hot leg safety injection

pathways are required to be closed with control power removed. In this configuration, they are not OPERABLE. Manual operator action is required to restore control power and operate the valves.

35 The bases to ISTS SR 3.5.2.4 states that testing may be accomplished by measuring the pump developed head at only one point of the pump characteristic curve. The method normally utilized in the reference ISTS plant is to set the flow and measure the pump developed head. At HBRSEP, Unit No. 2,, no capability exists in MODES 1, 2, 3, or 4 to set the pump flow at a flow rate substantial enough to permit measurement of the developed head as a variable. The ASME Boiler & Pressure Vessel (B&PV) Code allows alternately to set the head at a baseline value and measure flow to determine measured pump performance to within an acceptable tolerance. This is the test method employed at HBRSEP, Unit No. 2 during the applicable MODES for ECCS. Therefore, the Bases to ITS SR 3.5.2.3 reflect a Note that includes the alternate testing method as allowed by the ASME B&PV Code.

3.5 EMERGENCY CORE COOLING SYSTEMS (ECCS)

3.5.1 Accumulators

LCO 3.5.1 Three ECCS accumulators shall be OPERABLE.

APPLICABILITY: MODES 1 and 2,
MODE 3 with pressurizer pressure > 1000 psig.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One accumulator inoperable due to boron concentration not within limits.	A.1 Restore boron concentration to within limits.	72 hours
B. One valve identified in SR 3.5.1.5 with control power restored.	B.1 Verify control power or air is removed to all valves identified in SR 3.5.2.1 and SR 3.5.2.7.	Immediately
	AND	
	B.2 Remove control power to valve.	4 hours
C. One accumulator inoperable for reasons other than Condition A.	C.1 Restore accumulator to OPERABLE status.	4 hours

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
D. Required Action and associated Completion Time of Condition A or B not met.	D.1 Be in MODE 3. AND D.2 Reduce pressurizer pressure to ≤ 1000 psig.	6 hours 12 hours
E. Two or more accumulators inoperable.	E.1 Enter LCO 3.0.3.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.5.1.1	Verify each accumulator isolation valve is fully open.	Once prior to removing power from the valve operator
SR 3.5.1.2	Verify borated water volume in each accumulator is ≥ 825 ft^3 and ≤ 841 ft^3.	12 hours
SR 3.5.1.3	Verify nitrogen cover pressure in each accumulator is ≥ 600 psig and ≤ 660 psig.	12 hours

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.5.1.4	Verify boron concentration in each accumulator is ≥ 1950 ppm and ≤ 2400 ppm.	31 days AND -----NOTE------ Only required to be performed for affected accumulators ---------- Once within 6 hours after each solution volume increase of ≥ 70 gallons that is not the result of addition from the refueling water storage tank
SR 3.5.1.5	Verify control power is removed from each accumulator isolation valve operator.	31 days

3.5 EMERGENCY CORE COOLING SYSTEMS (ECCS)

3.5.2 ECCS – Operating

LCO 3.5.2 Two ECCS trains shall be OPERABLE.

APPLICABILITY: MODES 1, 2, and 3

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

1. In MODE 3, both safety injection (SI) pump flow paths may be isolated by closing the isolation valves for up to 12 hours to perform pressure isolation valve testing per SR 3.4.14.1.

2. Entry and operation in MODE 3 with one required SI pump declared inoperable pursuant to LCO 3.4.12, "Low Temperature Overpressure Protection (LTOP) System," is allowed for up to 4 hours or until the temperature of all RCS cold legs exceeds 375°F, whichever comes first.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more trains inoperable. AND At least 100% of the ECCS flow equivalent to a single OPERABLE ECCS train available.	A.1 Restore train(s) to OPERABLE status.	72 hours
B. One valve identified in SR 3.5.2.1 and SR 3.5.2.7 with control power or air restored.	B.1 Verify control power isremoved to all valves identified in SR 3.5.1.5. AND	Immediately (continued)

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
B. (continued)	B.2 Remove control power or air to valve.	24 hours
C. Required Action and associated Completion Time not met.	C.1 Be in MODE 3. AND C.2 Be in MODE 4.	6 hours 12 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.5.2.1	Verify the following valves are in the listed position with control power to the valve operator removed. Number / Position / Function SI-862 A&B / Open / Low Head Safety Injection (LHSI) SI-863 A&B / Closed / LHSI SI-864 A&B / Open / LHSI, High Head Safety Injection (HHSI) SI-866 A&B / Closed / HHSI SI-878 A&B / Open / HHSI	12 hours
SR 3.5.2.2	Verify each ECCS manual, power operated, and automatic valve in the flow path, that is not locked, sealed, or otherwise secured in position, is in the correct position.	31 days

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.5.2.3	----------NOTE---------- Alternate testing may be performed by setting the pump head at the baseline value and measuring the flow to meet the requirements of SR 3.5.2.3. ---------- Verify each ECCS pump's developed head at the test flow point is greater than or equal to the required developed head.	In accordance with the Inservice Testing Program
SR 3.5.2.4	Verify each ECCS automatic valve in the flow path that is not locked, sealed, or otherwise secured in position, actuates to the correct position on an actual or simulated actuation signal.	18 months
SR 3.5.2.5	Verify each ECCS pump starts automatically on an actual or simulated actuation signal.	18 months
SR 3.5.2.6	Verify, by visual inspection, the ECCS train containment sump suction inlet is not restricted by debris and the suction inlet trash racks and screens show no evidence of structural distress or abnormal corrosion.	18 months

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.5.2.7	Verify the following valves in the listed position: Number — Position — Function FCV-605 — Closed/Motive Air Isolated — RHR HCV-758 — Closed/Motive Air Isolated — RHR	31 days
SR 3.5.2.8	Verify the following manual valve is locked in the listed position Number — Position — Function RHR-764 — Locked Open — LHSI	92 days

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.5.3.1	----------NOTE---------- An RHR train may be considered OPERABLE during alignment and operation for decay heat removal, if capable of being manually realigned to the ECCS mode of operation. ---------- The following SRs are applicable for all equipment required to be OPERABLE: SR 3.5.2.3 SR 3.5.2.6	In accordance with applicable SRs

BASES

ACTIONS

A.1 (continued)

large main steam line break. Thus, 72 hours is allowed to return the boron concentration to within limits.

B.1 and B.2

If control power is restored to one valve identified in SR 3.5.1.5, immediate verification must be performed that no other valves listed in SR 3.5.2.1, and SR 3.5.2.7 have the control power or air restored. Additionally, Required Action B.2 requires the control power to be removed to the valve within 4 hours. In this condition, the valves could be subject to a spurious single failure that could result in closure of the valve and isolation of an accumulator. During the interval in which control power is restored, the valve remains in its required position The 4 hour Completion Time is reasonable considering a low probability of a spurious single failure coincident with a LOCA and is consistent with the 4 hour allowed outage time for one accumulator.

C.1

If one accumulator is inoperable for a reason other than boron concentration, the accumulator must be returned to OPERABLE status within 4 hours. In this Condition, the required contents of two accumulators cannot be assumed to reach the core during a LOCA. Due to the severity of the consequences should a LOCA occur in these conditions, the 4 hour Completion Time to open the valve, remove power to the valve, or restore the proper water volume or nitrogen cover pressure ensures that prompt action will be taken to return the inoperable accumulator to OPERABLE status. The Completion Time minimizes the potential for exposure of the plant to a LOCA under these conditions.

D.1 and D.2

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

(continued)

BASES

ACTIONS

D.1 and D.2 (continued)

6 hours and pressurizer pressure reduced to ≤ 1000 psig within 12 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.

E.1

If more than one accumulator is inoperable, the plant is in a condition outside the accident analyses; therefore, LCO 3.0.3 must be entered immediately.

SURVEILLANCE REQUIREMENTS

SR 3.5.1.1

Each accumulator isolation valve should be verified to be fully open prior to removing power from the operator. This verification ensures that the accumulators are available for injection. If an isolation valve is not fully open, the rate of injection to the RCS would be reduced. Although a motor operated valve position should not change with power removed, a closed valve could result in not meeting accident analyses assumptions.

This Frequency is considered reasonable in view of other administrative controls that ensure a mispositioned isolation valve is unlikely.

SR 3.5.1.2 and SR 3.5.1.3

Every 12 hours, borated water volume and nitrogen cover pressure are verified for each accumulator. This Frequency is sufficient to ensure adequate injection during a LOCA. Because of the static design of the accumulator, a 12 hour Frequency usually allows the operator to identify changes before limits are reached. Operating experience has shown this Frequency to be appropriate for early detection and correction of off normal trends.

(continued)

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.5.1.4

The boron concentration should be verified to be within required limits for each accumulator every 31 days since the static design of the accumulators limits the ways in which the concentration can be changed. The 31 day Frequency is adequate to identify changes that could occur from mechanisms such as stratification or inleakage. Sampling the affected accumulator within 6 hours after a ≥ 70 gallon volume increase will identify whether inleakage has caused a reduction in boron concentration. The 70 gallon volume increase and time limit of 6 hours is based on preventing a reduction in boron concentration in an accumulator below 1950 ppm with an initial boron concentration of 2000 ppm assuming in-leakage of 70 gallons pure water at a maximum in-leakage rate of 0.2 gpm. It is not necessary to verify boron concentration if the added water inventory is from the refueling water storage tank (RWST), because the water contained in the RWST is within the accumulator boron concentration requirements. This is consistent with the recommendation of NUREG-1366 (Ref. 4).

SR 3.5.1.5

Verification every 31 days that control power is removed from each accumulator isolation valve operator ensures that an active failure could not result in the undetected closure of an accumulator motor operated isolation valve. If this were to occur, only one accumulator would be available for injection given a single failure coincident with a LOCA. Since power is removed under administrative control, the 31 day Frequency will provide adequate assurance that power is removed.

This SR allows power to be supplied to the motor operated isolation valves when pressurizer pressure is < 1000 psig, thus allowing operational flexibility by avoiding unnecessary delays to manipulate the breakers during plant startups or shutdowns.

REFERENCES

1. UFSAR, Section 6.2.1.

2. 10 CFR 50.46.

(continued)

BASES

REFERENCES (continued)

3. UFSAR, Chapter 15.

4. NUREG-1366, February 1990.

BASES

APPLICABILITY (continued)

based on full power operation. Although reduced power would not require the same level of performance, the accident analysis does not provide for reduced cooling requirements in the lower MODES. The SI pump performance requirements are based on a small break LOCA. MODE 2 and MODE 3 requirements are bounded by the MODE 1 analysis.

Although the LCO is applicable in MODES 1, 2 and 3, the pressurizer low pressure and high steam differential pressure SI signals may be blocked when pressurizer pressure is < 2000 psig. The high steam flow coincident with low steam pressure or low average coolant temperature SI signal may be blocked when average coolant temperature is < 543°F. These blocks facilitate plant heatup and cooldown (Ref. 4).

As indicated in Note 1, the flow path may be isolated for 12 hours in MODE 3, under controlled conditions, to perform pressure isolation valve testing per SR 3.4.14.1.

As indicated in Note 2, operation in MODE 3 with one ECCS train declared inoperable pursuant to LCO 3.4.12, "Low Temperature Overpressure Protection (LTOP) System," is necessary with an LTOP arming temperature at or near the MODE 3 boundary temperature of 350°F. LCO 3.4.12 requires that certain pumps be rendered inoperable at and below the LTOP arming temperature. Since this temperature is at or near the MODE 3 boundary temperature, time is needed to restore the inoperable pumps to OPERABLE status.

In MODES 5 and 6, plant conditions are such that the probability of an event requiring ECCS injection is extremely low. Core cooling requirements in MODE 5 are addressed by LCO 3.4.7, "RCS Loops – MODE 5, Loops Filled," and LCO 3.4.8, "RCS Loops – MODE 5, Loops Not Filled." MODE 6 core cooling requirements are addressed by LCO 3.9.4, "Residual Heat Removal (RHR) and Coolant Circulation – High Water Level," and LCO 3.9.5, "Residual Heat Removal (RHR) and Coolant Circulation – Low Water Level."

ACTIONS

A.1

With one or more trains inoperable and at least 100% of the ECCS flow equivalent to a single OPERABLE ECCS train

(continued)

BASES

ACTIONS

A.1 (continued)

available, the inoperable components must be returned to OPERABLE status within 72 hours. The 72 hour Completion Time is based on an NRC reliability evaluation (Ref. 5) and is a reasonable time for repair of many ECCS components.

An ECCS train is inoperable if it is not capable of delivering design flow to the RCS. Individual components are inoperable if they are not capable of performing their design function or supporting systems are not available.

The LCO requires the OPERABILITY of a number of subsystems. Due to the redundancy of trains and the diversity of subsystems, the inoperability of one active component in a train does not render the ECCS incapable of performing its function. Neither does the inoperability of two different components, each in a different train, necessarily result in a loss of function for the ECCS. The intent of this Condition is to maintain a combination of equipment such that 100% of the ECCS flow equivalent to a single OPERABLE ECCS train remains available. This allows increased flexibility in plant operations under circumstances when components in opposite trains are inoperable.

An event accompanied by a loss of offsite power and the failure of an EDG can disable one ECCS train until power is restored. A reliability analysis (Ref. 5) has shown that the impact of having one full ECCS train inoperable is sufficiently small to justify continued operation for 72 hours.

Reference 6 describes situations in which one component, such as an RHR crossover valve, can disable both ECCS trains. With one or more component(s) inoperable such that 100% of the flow equivalent to a single OPERABLE ECCS train is not available, the facility is in a condition outside the accident analysis. Therefore, LCO 3.0.3 must be immediately entered.

B.1 and B.2

If control power or air is restored to one valve identified in SR 3.5.2.1 and SR 3.5.2.7, immediate verification must be performed that no other valves listed in SR 3.5.1.5 have the

(continued)

BASES

ACTIONS

B.1 and B.2 (continued)

control power restored. Additionally, Required Action B.2 requires the control power to be removed to the valve within 24 hours. In this condition, the valves could be subject to a spurious single failure that could result in closure of the valve and isolation of an accumulator. During the interval in which control power is restored, the valve remains in its required position The 24 hour Completion Time is reasonable considering a low probability of a spurious single failure coincident with a LOCA.

C.1 and C.2

If the inoperable trains cannot be returned to OPERABLE status within the associated Completion Time, the plant must be brought to a MODE in which the LCO does not apply. To achieve this status, the plant must be brought to MODE 3 within 6 hours and MODE 4 within 12 hours. The allowed

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

SURVEILLANCE REQUIREMENTS

SR 3.5.2.1

Verification of proper valve position ensures that the flow path from the ECCS pumps to the RCS is maintained. Misalignment of these valves could render both ECCS trains inoperable. Securing these valves in position by removal of control power or by key locking the control in the correct position ensures that they cannot change position as a result of an active failure or be inadvertently misaligned. These valves are of the type, described in Reference 6, that can disable the function of both ECCS trains and invalidate the accident analyses. A 12 hour Frequency is considered reasonable in view of other administrative controls that will ensure a mispositioned valve is unlikely.

(continued)

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.5.2.2

Verifying the correct alignment for manual, power operated, and automatic valves in the ECCS flow paths provides assurance that the proper flow paths will exist for ECCS 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 receives an actuation signal is allowed to be in a nonaccident position provided the valve will automatically reposition within the proper stroke time. This Surveillance does not require anytesting or valve manipulation. Rather, it involves verification that those valves capable of being mispositioned are in the correct position. The 31 day Frequency is appropriate because the valves are operated under administrative control, and an improper valve position would only affect a single train. This Frequency has been shown to be acceptable through operating experience.

SR 3.5.2.3

Periodic surveillance testing of ECCS pumps to detect gross degradation caused by impeller structural damage or other hydraulic component problems is required by Section XI of the ASME Code. This type of testing may be accomplished by measuring the pump developed head at only one point of the pump characteristic curve. This verifies both that the measured performance is within an acceptable tolerance of the original pump baseline performance and that the performance at the test flow is greater than or equal to the performance assumed in the plant safety analysis. Alternately, a Note to SR 3.5.2.3 permits pump performance verification by setting the pump head at the baseline value and measuring the test flow. SRs are specified in the Inservice Testing Program, which encompasses Section XI of the ASME Code. Section XI of the ASME Code provides the activities and Frequencies necessary to satisfy the requirements.

SR 3.5.2.4 and SR 3.5.2.5

These Surveillances demonstrate that each automatic ECCS valve actuates to the required position on an actual or

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.5.2.4 and SR 3.5.2.5 (continued)

simulated SI signal and that each ECCS pump starts on receipt of an actual or simulated SI signal. This Surveillance is not required for valves that are locked, sealed, or otherwise secured in the required position under administrative controls. The 18 month Frequency is based on the need to perform these Surveillances under the conditions that apply during a plant outage and the potential for unplanned plant transients if the Surveillances were performed with the reactor at power. The 18 month Frequency is also acceptable based on consideration of the design reliability (and confirming operating experience) of the equipment. The actuation logic is tested as part of ESF Actuation System testing, and equipment performance is monitored as part of the Inservice Testing Program.

SR 3.5.2.6

Periodic inspections of the containment sump suction inlet ensure that it is unrestricted and stays in proper operating condition. The 18 month Frequency is based on the need to perform this Surveillance under the conditions that apply during a plant outage, on the need to have access to the location, and because of the potential for an unplanned transient if the Surveillance were performed with the reactor at power. This Frequency has been found to be sufficient to detect abnormal degradation and is confirmed by operating experience.

SR 3.5.2.7

Verification of proper valve position ensures the proper flow path is established for the LHSI system following operation in RHR mode. The Frequency of 31 days is commensurate with the accessibility and radiation levels involved in performing the surveillance (Ref. 6).

SR 3.5.2.8

Verification of proper valve position ensures the proper flow path is established for the LHSI system following operation in RHR mode. The Frequency of 92 days is based on

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.5.2.8 (continued)

the locked status for the valve as well as the accessibility and radiation levels involved in performing the surveillance (Ref. 6).

REFERENCES

1. UFSAR Paragraph 3.1.2.37.
2. 10 CFR 50.46.
3. UFSAR, Chapter 15.
4. UFSAR, Chapter 6.
5. NRC Memorandum to V. Stello, Jr., from R.L. Baer, "Recommended Interim Revisions to LCOs for ECCS Components," December 1, 1975.
6. IE Information Notice No. 87-01.
7. CP&L Letter to NRC, from G. E. Vaughn, "Emergency Core Cooling System (ECCS) Failure Mode and Effects Analysis (FMEA) Summary Information," May 7, 1991.

BASES

LCO (continued)

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

During an event requiring ECCS actuation, a flow path is required to provide an abundant supply of water from the RWST to the RCS via the ECCS pumps and their respective supply headers to each of the three cold leg injection nozzles. In the long term, this flow path may be switched to take its supply from the containment sump and to deliver its flow to the RCS hot and cold legs. The hot leg injection paths of the SI System, including valves, are not subject to the requirements of this specification.

APPLICABILITY

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

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

In MODES 5 and 6, plant conditions are such that the probability of an event requiring ECCS injection is extremely low. Core cooling requirements in MODE 5 are addressed by LCO 3.4.7, "RCS Loops – MODE 5, Loops Filled," and LCO 3.4.8, "RCS Loops – MODE 5, Loops Not Filled." MODE 6 core cooling requirements are addressed by LCO 3.9.4, "Residual Heat Removal (RHR) and Coolant Circulation – High Water Level," and LCO 3.9.5, "Residual Heat Removal (RHR) and Coolant Circulation – Low Water Level."

(continued)

SUPPLEMENT 5
CONVERSION PACKAGE SECTION 3.6
PAGE INSERTION INSTRUCTIONS

Remove and insert the following pages into Enclosure 14 to Serial: RNP-RA/96-0141.

	Remove Page	Insert Page
a.	Part 1, "Markup of Current Technical Specifications (CTS)"	
	NA	
b.	Part 2, "Discussion of Changes (DOCs) for CTS Markup"	
	NA	
c.	Part 3, " No Significant Hazards Consideration (NSHC), And Basis for Categorical Exclusion from 10 CFR 51.22	
	NA	
d.	Part 4, "Markup of NUREG-4131, Revision 1, Standard Technical Specifications-Westinghouse Plants, (ISTS)"	
	3.6-12	3.6-12
e.	Part 5, "Justification of Differences (JFDs) to ISTS"	
	4	4
	-	5
f.	Part 6, "Markup of ISTS Bases"	
	B 3.6-7, B 3.6-29, B 3.6-40, B3.6-41	B 3.6-7, B 3.6-29, B 3.6-40, B 3.6-41
	B 3.6-72, B 3.6-73	B 3.6-72, B 3.6-73
g.	Part 7, "Justification for Differences (JFDs) to ISTS Bases"	
	1 & 4	1 & 4
	-	5
h.	Part 8, "Proposed HBRSEP, Unit No. 2 ITS"	
	3.6-11 & 3.6-12	3.6-11 & 3.6-12
i.	Part 9. "Proposed Bases to HBRSEP, Unit No. 2 ITS Bases"	
	B 3.6-2, B 3.6-13, B 3.6-22, B 3.6-23	B 3.6-2, B 3.6-13, B 3.6-22, B 3.6-23
	B 3.6-24	B 3.6-24
	-	B 3.6-24a
	B 3.6-40, B 3.6-41	B 3.6-40, B 3.6-41

SUPPLEMENT 5
CONVERSION PACKAGE SECTION 3.6
PAGE INSERTION INSTRUCTIONS

Remove and insert the following pages into Enclosure 14 to Serial: RNP-RA/96-0141:

	Remove Page	Insert Page
j.	Part 10. "ISTS Generic Changes" NA	

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SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
Insert 3.6.3-2		
[SR 3.6.3.1	Verify each [42] inch purge valve is sealed closed, except for one purge valve in a penetration flow path while in Condition E of this LCO.	31 days]
[3.6.4.1] SR 3.6.3.2	Verify each [8] inch purge valve (supply + exhaust) + each 6 inch pressure + vacuum relief valve is closed, except when the [8] inch containment purge valves are open for pressure control, ALARA or air quality considerations for personnel entry, or for Surveillances that require the valves to be open.	31 days
[M13] SR 3.6.3.3	NOTES 1. Valves and blind flanges in high radiation areas may be verified by use of administrative controls. Verify each containment isolation manual valve and blind flange that is located outside containment and required to be closed during accident conditions is closed, except for containment isolation valves that are open under administrative controls.	31 days

(continued)

and not locked, sealed or otherwise secured

2. Not required to be performed for Penetration Pressurization System valves with a diameter ≤ 3/8 inch.

Appendix J. To address the NRC concern regarding periodic surveillance of the resilient seals on the containment purge and vent valves, the resilient seals on the containment purge and vent valves will be replaced this refueling outage, and will be replaced every other refueling outage henceforth. An evaluation has been performed of the seal material which found that the seal material life expectancy exceeds the proposed replacement frequency. An evaluation has been performed in accordance with 10 CFR 50.59, and the evaluation has found that this modification does not pose an unreviewed safety question.

24 The HBRSEP design provides position limits on the inboard 42 inch purge valves only.

25 There is no basis to exclude the 42 inch purge supply and exhaust valves from being open or open under administrative control. Consistent with the current licensing basis, these valves may be opened for specified purposes provided they are not opened concurrently with the 6 inch pressure and vacuum relief valves.

26 Brackets are removed and plant specific values are incorporated.

27 Condition D is augmented to address the Current Licensing Basis prohibition against the simultaneous operation of containment purge and either pressure or vacuum relief penetrations.

28 SR 3.6.2.1 and Note 2 to SR 3.6.2.1 are modified to add a reference to Option A to reflect the CLB for containment airlock testing. The specific airlock leakage acceptance criteria are not adopted because no such requirement currently exists. The airlock's contribution to Type B and C containment leakage is limited such that the Type B and C containment leakage cannot exceed the applicable Type B and C containment leakage limits.

29 ITS SR 3.6.3.2 requires that each containment isolation manual valve and blind flange that is located outside containment and not locked, sealed or otherwise secured and required to be closed during accident conditions is closed, except for containment isolation valves open under administrative controls. This SR is modified by a Note (Note 2), consistent with the HBRSEP Unit No. 2 current licensing basis, to not require this SR to be performed for Penetration Pressurization System (PPS) valves with a diameter of $\leq$ 3/8 inch. However, it is the intent that the SR must still be met during the MODES of Applicability.

The valves affected by this change are the PPS valves located in the North and South Electrical Penetration Vaults. These valves provide isolation for the penetration sleeve from the PPS or provide convenient test connections for local leak rate testing apparatus. The isolation valves and test connections are maintained in the closed position during

normal operations and are only repositioned for penetration testing. All piping involved is 3/8 inch tubing connected to a "canned" penetration which is welded to the containment liner inside the containment. The affected valves are not subject to accident pressure during an accident unless the penetration sleeve fails. There are a total of approximately 104, 3/8 inch, valves that are proposed to be exempted from the performance requirements of ITS SR 3.6.3.2. The justification for this change is as follows:

a. The North and South Electrical Penetration Vaults at HBRSEP Unit No. 2 are not equipped for routine access to each penetration. As a result, monthly verification of PPS valves at each penetration would require building scaffolding and/or treading over electrical penetrations resulting in some risk of damaging cabling or the penetration seal itself.

b. The affected valves are only operated for penetration testing purposes. Their pre-test and post-test positions are verified and documented in plant procedures.

c. The valves are not subject to containment atmosphere unless the penetration itself fails on the containment side. Therefore, two failures are required for a degradation of the integrity of containment (i.e., the penetration must fail and the valve for that particular penetration must be mispositioned).

d. The system involved consists of 3/8 inch tubing and valves. The consequences of a mispositioning of one of the affected valves is minimal assuming this were the release path.

e. The North and South Electrical Penetration Vaults penetration areas are only rarely accessed, namely for penetration testing. Because of the low maintenance requirements for the equipment in the area (i.e., cabling and cable penetrations), it is very unlikely that an inadvertent mispositioning of a valve would occur due to activities in the area.

BASES

BACKGROUND (continued)

2. closed by manual valves, blind flanges, or de-activated automatic valves secured in their closed positions, except as provided in LCO 3.6.3, "Containment Isolation Valves";

b. Each air lock is OPERABLE, except as provided in LCO 3.6.2, "Containment Air Locks";

c. All equipment hatches are closed; and

d. The pressurized sealing mechanism associated with a penetration is OPERABLE, except as provided in LCO 3.6.[illegible]

APPLICABLE SAFETY ANALYSES

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

The DBAs that result in a challenge to containment OPERABILITY from high pressures and temperatures are a loss of coolant accident (LOCA), a steam line break, and a rod ejection accident (REA) (Ref. 2). In addition, release of significant fission product radioactivity within containment can occur from a LOCA or REA. In the DBA analyses, it is assumed that the containment is OPERABLE such that, for the DBAs involving release of fission product radioactivity, release to the environment is controlled by the rate of containment leakage. The containment was designed with an allowable leakage rate of 0.1% of containment air weight per day (Ref. 3). This leakage rate, used to evaluate offsite doses resulting from accidents, is defined in 10 CFR 50, Appendix J (Ref. 1), as L_a: the maximum allowable containment leakage rate at the calculated peak containment internal pressure (P_a) resulting from the limiting DBA. The allowable leakage rate represented by L_a forms the basis for the acceptance criteria imposed on all containment leakage rate testing. L_a is assumed to be 0.1% per day in the safety analysis at P_a = [illegible] psig (Ref. 3).

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

(continued)

1

B 3.6 CONTAINMENT SYSTEMS

B 3.6.3 Containment Isolation Valves (Atmospheric, Subatmospheric, Ice Condenser, and Dual)

BASES

BACKGROUND

The containment isolation valves form part of the containment pressure boundary and provide a means for fluid penetrations not serving accident consequence limiting systems to be provided with two isolation barriers that are closed on a containment isolation signal. These isolation devices are either passive or active (automatic). Manual valves, de-activated automatic valves secured in their closed position (including check valves with flow through the valve secured), blind flanges, and closed systems are considered passive devices. Check valves, or other automatic valves designed to close without operator action following an accident, are considered active devices. Two barriers in series are provided for each penetration so that no single credible failure or malfunction of an active component can result in a loss of isolation or leakage that exceeds limits assumed in the safety analyses. One of these barriers may be a closed system. These barriers (typically containment isolation valves) make up the Containment Isolation System.

Manual valves qualifying as containment isolation valves are secured closed.

Automatic isolation signals are produced during accident conditions. Containment Phase "A" isolation occurs upon receipt of a safety injection signal. The Phase "A" isolation signal isolates nonessential process lines in order to minimize leakage of fission product radioactivity. Containment Phase "B" isolation occurs upon receipt of a containment pressure High-High signal and isolates the remaining process lines, except systems required for accident mitigation. In addition to the isolation signals listed above, the purge and exhaust valves receive an isolation signal on a containment high radiation condition. As a result, the containment isolation valves (and blind flanges) help ensure that the containment atmosphere will be isolated from the environment in the event of a release of fission product radioactivity to the containment atmosphere as a result of a Design Basis Accident (DBA).

Supply

The OPERABILITY requirements for containment isolation valves help ensure that containment is isolated within the

(continued)

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.6.3.4

This SR requires verification that each containment isolation manual valve and blind flange located outside containment and required to be closed during accident conditions is closed. The SR helps to ensure that post accident leakage of radioactive fluids or gases outside of the containment boundary is within design limits. This SR does not require any testing or valve manipulation. Rather, it involves verification, through a system walkdown, that those containment isolation valves outside containment and capable of being mispositioned are in the correct position. Since verification of valve position for containment isolation valves outside containment is relatively easy, the 31 day Frequency is based on engineering judgment and was chosen to provide added assurance of the correct positions. The SR specifies that containment isolation valves that are open under administrative controls are not required to meet the SR during the time the valves are open.

and not locked, sealed or otherwise secured

TSTF 45, R.1

INSERT B3.6.3.15

The Note applies to valves and blind flanges located in high radiation areas and allows these devices to be verified closed by use of administrative means. Allowing verification by administrative means is considered acceptable, since access to these areas is typically restricted during MODES 1, 2, 3 and 4 for ALARA reasons. Therefore, the probability of misalignment of these containment isolation valves, once they have been verified to be in the proper position, is small.

Note 2 states that performance of the SR is not required for the Penetration Pressurization System (PPS) valves with a diameter ≤ 3/8 inch. It is the intent that this SR must still be met, but performance is not required for PPS valves with a diameter ≤ 3/8 inch. The Note 2 allowance is consistent with the original plant licensing basis and is considered acceptable based on the low probability of these valves being mispositioned and the minimal consequences associated with the mispositioning of one of these valves.

SR 3.6.3.4

This SR requires verification that each containment isolation manual valve and blind flange located inside containment and required to be closed during accident conditions is closed. The SR helps to ensure that post accident leakage of radioactive fluids or gases outside of the containment boundary is within design limits. For containment isolation valves inside containment, the Frequency of "prior to entering MODE 4 from MODE 5 if not performed within the previous 92 days" is appropriate since these containment isolation valves are operated under

(continued)

Supplement 5

1

BASES

SURVEILLANCE REQUIREMENTS

SR 3.6.3.4 (continued)

3

administrative controls and the probability of their misalignment is low. The SR specifies that containment isolation valves that are open under administrative controls are not required to meet the SR during the ~~time they are~~ open.

TSTF 45, R.1

Insert B.3.6.3-16

This Note allows valves and blind flanges located in high radiation areas to be verified closed by use of administrative means. Allowing verification by administrative means is considered acceptable, since access to these areas is typically restricted during MODES 1, 2, 3, and 4, for ALARA reasons. Therefore, the probability of misalignment of these containment isolation valves, once they have been verified to be in their proper position, is small.

SR 3.6.3.5

4

20

Verifying that the isolation time of each ~~power operated and~~ automatic containment isolation valve is within limits is required to demonstrate OPERABILITY. The isolation time test ensures the valve will isolate in a time period less than or equal to that assumed in the safety analyses. The isolation time and Frequency of this SR are in accordance with the Inservice Testing Program ~~or 92 days~~.

20

Power operated

(IST)

20

In addition, to the IST Program testing Frequency, the 42 inch purge supply and exhaust valves will be tested prior to use if not tested within the previous quarter. Otherwise the 42 inch purge supply and exhaust valves are not cycled quarterly only for testing purposes.

~~SR 3.6.3.6~~

~~In subatmospheric containments, the check valves that serve a containment isolation function are weight or spring loaded to provide positive closure in the direction of flow. This ensures that these check valves will remain closed when the inside containment atmosphere returns to subatmospheric conditions following a DBA. SR 3.6.3.6 requires verification of the operation of the check valves that are testable during unit operation. The Frequency of 92 days is consistent with the Inservice Testing Program requirement for valve testing on a 92 day Frequency.~~

(continued)

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.6.6A.2

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

SR 3.6.6A.3

Verifying that each required containment cooling train ESW cooling flow rate to each cooling unit is ≥ 1700 gpm provides assurance that the design flow rate assumed in the safety analyses will be achieved (Ref. 3). The Frequency was developed considering the known reliability of the Cooling Water System, the two train redundancy available, and the low probability of a significant degradation of flow occurring between surveillances.

SR 3.6.6A.4

Verifying each containment spray pump's developed head at the flow test point is greater than or equal to the required developed head ensures that spray pump performance has not degraded during the cycle. Flow and differential pressure are normal tests of centrifugal pump performance required by Section XI of the ASME Code (Ref. 8). Since the containment spray pumps cannot be tested with flow through the spray headers, they are tested on recirculation flow. This test confirms one point on the pump design curve and is indicative of overall performance. Such inservice tests confirm component OPERABILITY, trend performance, and detect incipient failures by abnormal performance. The Frequency of the SR is in accordance with the Inservice Testing Program.

(continued)

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.6.6.5 and SR 3.6.6.6

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

SR 3.6.6.5

High

SR 3.6.6.6 must be performed with the isolation valves in the spray supply lines to the containment and spray and fire tank blocked closed.

The surveillance of containment sump isolation valves is also required by SR 3.5.2.5. A single surveillance may be used to satisfy both requirements.

44

SR 3.6.6.7

This SR requires verification that each containment cooling train actuates upon receipt of an actual or simulated safety injection signal. The 18 month Frequency is based on engineering judgment and has been shown to be acceptable through operating experience. See SR 3.6.6.5 and SR 3.6.6.6, above, for further discussion of the basis for the 18 month Frequency.

SR 3.6.6.8

With the containment spray inlet valves closed and the spray header drained of any solution, low pressure air or smoke can be blown through test connections. This SR ensures that each spray nozzle is unobstructed and provides assurance that spray coverage of the containment during an accident is not degraded. Due to the passive design of the nozzle, a test at the first refueling and at 10 year intervals is considered adequate to detect obstruction of the nozzles.

(continued)

1 In the conversion of the HBRSEP current Technical Specifications (CTS) to the proposed plant specific Improved Technical Specifications (ITS), certain wording preferences or conventions are adopted which do not result in technical changes (either actual or interpretational). Editorial changes or clarifications which involve the insertion of plant specific terms, parameters, or descriptions are used to preserve consistency with the CTS and licensing basis.

2 The description regarding the containment tendons is modified to reflect HBRSEP plant specific design.

3 The HBRSEP containment analysis does not provide a unique analysis for a LOCA resulting from a Rod Ejection. The containment response for a Rod Ejection Accident is bounded by the containment response for a LOCA.

4 The information in the Bases to ITS 3.6.1, Background regarding SR 3.6.1.1 and 10 CFR 50 Appendix J is deleted since it is inappropriate for the Bases Background section.

5 The Bases to ITS 3.6.3, Background is modified to provide information regarding the inter-relationship of Isolation Valve Seal Water (IVSW) System and containment isolation valves.

6 The HBRSEP design uses grouted tendons. HBRSEP has not committed to Regulatory Guide 1.35.

7 The HBRSEP design does not include provisions for indicating individual air lock door positions in the control room. A control room annunciator indicates when any airlock door is open.

8 Consistent with the current licensing basis, a specific leakage rate acceptance criteria applicable to the air lock is not adopted.

9 The HBRSEP design provides a single containment airlock.

10 Individual leakage rates for the containment air lock and purge valves are not specified in LCO 3.6.2, LCO 3.6.3 or the Containment Leakage Rate Testing Program.

11 The bases are revised to clarify that operation of the purge system is not limited to reducing the concentration of noble gases.

12 Consistent with the current licensing basis, the 42 inch purge valves are permitted to be open for safety reasons to support plant operations and maintenance activity within containment.

13 The HBRSEP does not include a Mini-purge System. To maintain consistency with the current licensing basis, an appropriate Background

and spray line filling. The 60 seconds is considered to conservatively bound these events.

40 The unmodified sentence in ISTS is not grammatically correct. The eliminated information is an interjection in the sentence structure that does not appear to be significant with respect to the information provided. Its elimination improves the clarity of the sentence without significantly altering its meaning.

41 Redundant containment cooling is provided by the remaining containment spray train and the containment cooling trains.

42 The remaining containment cooling trains are identified for clarity. The reference to iodine removal capability is not germane to the containment cooling function and inoperability of a containment cooling train.

43 The HBRSEP design for the Containment Spray System does not use check valves inside containment.

44 The containment isolation sump valves are not included in this surveillance requirement since the HBRSEP design utilizes a manual realignment of the containment spray pump suction from the RWST to the containment sump.

45 The HBRSEP design for the eductors provides for a spray mixture ph of $\geq$ 8.8 and $\leq$ 10.0 during the injection phase.

46 HBRSEP analysis assumes adequate coverage of the containment volume by the Containment Spray System.

47 Consistent with the current licensing basis, the NaOH concentration has only a minimum value.

48 Consistent with the current licensing basis, ISTS specification 3.6.8, Hydrogen Recombiners is not adopted in ITS. Appropriate bases for ITS Specification 3.6.8, Isolation Valve Seal Water System is provided.

49 The HBRSEP design does not include a Containment Hydrogen Mixing System.

50 The HBRSEP design does not include an Iodine Cleanup System.

51 Consistent with the current licensing basis, ISTS specification 3.6.12, Vacuum Relief Valves is not adopted in ITS.

52 The description of the containment spray pump testing requirements (in the Bases for ITS SR 3.6.6.4) are revised to reflect the HBRSEP Unit No. 2 plant design and current practice. The method normally utilized in the reference ISTS plant is to set the flow and measure the pump

developed head. At HBRSEP, Unit No. 2,, no capability exists in MODES 1, 2, 3, or 4 to set the pump flow at a flow rate substantial enough to permit measurement of the developed head as a variable. The ASME Boiler & Pressure Vessel (B&PV) Code allows alternately to set the head at a baseline value and measure flow to determine measured pump performance to within an acceptable tolerance. This is the test method employed at HBRSEP, Unit No. 2 during the applicable MODES for ECCS. Therefore, the Bases to ITS SR 3.6.6.4 includes the plant's testing method allowed by the ASME B&PV Code.

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.6.3.2	----------NOTE---------- 1. Not required to be performed for Penetration Pressurization System valves with a diameter ≤ 3/8 inch. 2. Valves and blind flanges in high radiation areas may be verified by use of administrative controls. ---------- Verify each containment isolation manual valve and blind flange that is located outside containment and not locked, sealed or otherwise secured and required to be closed during accident conditions is closed, except for containment isolation valves that are open under administrative controls.	31 days
SR 3.6.3.3	----------NOTE---------- Valves and blind flanges in high radiation areas may be verified by use of administrative means. ---------- Verify each containment isolation manual valve and blind flange that is located inside containment and not locked, sealed or otherwise secured and required to be closed during accident conditions is closed, except for containment isolation valves that are open under administrative controls.	Prior to entering MODE 4 from MODE 5 if not performed within the previous 92 days

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.6.3.4	Verify the isolation time of each automatic power operated containment isolation valve is within limits.	In accordance with the Inservice Testing Program
SR 3.6.3.5	Verify each automatic containment isolation valve that is not locked, sealed or otherwise secured in position, actuates to the isolation position on an actual or simulated actuation signal.	18 months
SR 3.6.3.6	Verify each 42 inch inboard containment purge valve is blocked to restrict the valve from opening > 70°.	18 months

BASES

BACKGROUND (continued)

b. The air lock is OPERABLE, except as provided in LCO 3.6.2, "Containment Air Lock";

c. The equipment hatch is closed and sealed; and

d. The Isolation Valve Seal Water (IVSW) sytem is OPERABLE, except as provided in LCO 3.6.8.

APPLICABLE SAFETY ANALYSES

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

The DBAs that result in a challenge to containment OPERABILITY from high pressures and temperatures are a loss of coolant accident (LOCA) and a steam line break (Ref. 2). In addition, release of significant fission product radioactivity within containment can occur from a LOCA. In the LOCA analyses, it is assumed that the containment is OPERABLE such that, for the LOCA, the release to the environment is controlled by the rate of containment leakage. The containment was designed with an allowable leakage rate of 0.1% of containment air weight per day (Ref. 2). This leakage rate, used to evaluate offsite doses resulting from accidents, is defined in 10 CFR 50, Appendix J (Ref. 1), as L_a: the maximum allowable containment leakage rate at the calculated peak containment internal pressure (P_a) resulting from the LOCA. The allowable leakage rate represented by L_a forms the basis for the acceptance criteria imposed on all containment leakage rate testing. L_a is assumed to be 0.1% per day in the safety analysis at P_a = 40.0 psig (Ref. 2).

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

The containment satisfies Criterion 3 of the NRC Policy Statement.

(continued)

B 3.6 CONTAINMENT SYSTEMS

B 3.6.3 Containment Isolation Valves

BASES

BACKGROUND

The containment isolation valves form part of the containment pressure boundary and provide a means for fluid penetrations not serving accident consequence limiting systems to be provided with two isolation barriers that are closed on a containment isolation signal. These isolation devices are either passive or active (automatic). Manual valves, de-activated automatic valves secured in their closed position (including check valves with flow through the valve secured), blind flanges, and closed systems are considered passive devices. Check valves, or other automatic valves designed to close without operator action following an accident, are considered active devices. Manual valves qualifying as containment isolation valves are secured closed. Two barriers in series are provided for each penetration so that no single credible failure or malfunction of an active component can result in a loss of isolation or leakage that exceeds limits assumed in the safety analyses. One of these barriers may be a closed system. These barriers (typically containment isolation valves) make up the Containment Isolation System.

Automatic isolation signals are produced during accident conditions. Containment Phase "A" isolation occurs upon receipt of a safety injection signal. The Phase "A" isolation signal isolates nonessential process lines in order to minimize leakage of fission product radioactivity. Containment Phase "B" isolation occurs upon receipt of a containment pressure High-High signal and isolates the remaining process lines, except systems required for accident mitigation. In addition to the isolation signals listed above, the purge supply and exhaust valves receive an isolation signal on a containment high radiation condition. As a result, the containment isolation valves (and blind flanges) help ensure that the containment atmosphere will be isolated from the environment in the event of a release of fission product radioactivity to the containment atmosphere as a result of a Design Basis Accident (DBA).

(continued)

BASES (continued)

SURVEILLANCE REQUIREMENTS

SR 3.6.3.1 (continued)

control, ALARA or air quality considerations for personnel entry, or for Surveillances that require the valves to be open. The valves are capable of closing in the environment following a LOCA. Therefore, these valves are allowed to be open for limited periods of time. The 31 day Frequency is consistent with other containment isolation valve requirements discussed in SR 3.6.3.3. Since it is not operationally necessary, it is desirable to preclude the 42 inch valves and 6 inch valves from being open at the same time. A Note to this SR restricts the 6 inch and 42 inch valves from being open simultaneously.

SR 3.6.3.2

This SR requires verification that each containment isolation manual valve and blind flange located outside containment and not locked, sealed or otherwise secured and required to be closed during accident conditions is closed. The SR helps to ensure that post accident leakage of radioactive fluids or gases outside of the containment boundary is within design limits. This SR does not require any testing or valve manipulation. Rather, it involves verification, through a system walkdown, that those containment isolation valves outside containment and capable of being mispositioned are in the correct position. Since verification of valve position for containment isolation valves outside containment is relatively easy, the 31 day Frequency is based on engineering judgment and was chosen to provide added assurance of the correct positions. The SR specifies that containment isolation valves that are open under administrative controls are not required to meet the SR during the time the valves are open. This SR does not apply to valves that are locked, sealed or otherwise secured in the closed position, since these were verified to be in the correct position upon locking, sealing or securing.

Note 1 applies to valves and blind flanges located in high radiation areas and allows these devices to be verified closed by use of administrative means. Allowing verification by administrative means is considered acceptable, since access to these areas is typically restricted during MODES 1, 2, 3 and 4 for ALARA reasons. Therefore, the probability of misalignment of these

(continued)

BASES (continued)

SURVEILLANCE REQUIREMENTS

SR 3.6.3.2 (continued)

containment isolation valves, once they have been verified to be in the proper position, is small. Note 2 states that performance of the SR is not required for the Penetration Pressurization System (PPS) valves with a diameter ≤ 3/8 inch. It is the intent that this SR must still be met, but performance is not required for PPS valves with diameter ≤ 3/8 inch. The Note 2 allowance is consistent with the original plant licensing basis and is considered acceptable based on the low probability of these valves being mispositioned and the minimal consequences associated with mispositioning one of these valves.

SR 3.6.3.3

This SR requires verification that each containment isolation manual valve and blind flange located inside containment and not locked, sealed or otherwise secured and required to be closed during accident conditions is closed. The SR helps to ensure that post accident leakage of radioactive fluids or gases outside of the containment boundary is within design limits. For containment isolation valves inside containment, the Frequency of "prior to entering MODE 4 from MODE 5 if not performed within the previous 92 days" is appropriate since these containment isolation valves are operated under administrative controls and the probability of their misalignment is low. The SR specifies that containment isolation valves that are open under administrative controls are not required to meet the SR during the time they are open. This SR does not apply to valves that are locked, sealed or otherwise secured in the closed position, since these were verified to be in the correct position upon locking, sealing or securing.

This Note allows valves and blind flanges located in high radiation areas to be verified closed by use of administrative means. Allowing verification by administrative means is considered acceptable, since access to these areas is typically restricted during MODES 1, 2, 3, and 4, for ALARA reasons. Therefore, the probability of misalignment of these containment isolation valves, once they have been verified to be in their proper position, is small.

(continued)

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.6.3.4

Verifying that the isolation time of each automatic power operated containment isolation valve is within limits is required to demonstrate OPERABILITY. The isolation time test ensures the valve will isolate in a time period less than or equal to that assumed in the safety analyses. The isolation time and Frequency of this SR are in accordance with the Inservice Testing (IST) Program. In addition to the IST program testing frequency, the 42 inch purge supply and exhaust valves will be tested prior to use if not tested within the previous quarter. Otherwise, the 42 inch purge supply and exhaust valves are not cycled quarterly only for testing purposes.

SR 3.6.3.5

Automatic containment isolation valves close on a containment isolation signal to prevent leakage of radioactive material from containment following a DBA. This SR ensures that each automatic containment isolation valve will actuate to its isolation position on a containment isolation signal. This surveillance is not required for valves that are locked, sealed, or otherwise secured in the required position under administrative controls. The 18 month Frequency is based on the need to perform this Surveillance under the conditions that apply during a plant outage and the potential for an unplanned transient if the Surveillance were performed with the reactor at power. Operating experience has shown that these components usually pass this Surveillance when performed at the 18 month Frequency. Therefore, the Frequency was concluded to be acceptable from a reliability standpoint.

SR 3.6.3.6

Verifying that each 42 inch inboard containment purge valve is blocked to restrict opening to ≤ 70° is required to ensure that the valves can close under DBA conditions within the times assumed in the analyses of References 1 and 2. If a LOCA occurs, the purge valves must close to maintain containment leakage within the values assumed in the accident analysis. At other times when purge valves are required to be capable of closing (e.g., during movement of

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.6.3.6 (continued)

irradiated fuel assemblies), pressurization concerns are not present, thus the purge valves can be fully open. The 18 month Frequency is appropriate because the blocking devices are typically removed only during a refueling outage.

REFERENCES

1. UFSAR, Chapter 15.
2. UFSAR, Section 6.2.
3. Standard Review Plan 6.2.4.

BASES

SURVEILLANCE REQUIREMENTS

SR 3.6.6.3 (continued)

train redundancy available, and the low probability of a significant degradation of flow occurring between surveillances.

SR 3.6.6.4

Verifying each containment spray pump's developed head at the flow test point is greater than or equal to the required developed head ensures that spray pump performance has not degraded during the cycle. Flow and differential pressure are normal tests of centrifugal pump performance required by Section XI of the ASME Code (Ref. 5). Since the containment spray pumps cannot be tested with flow through the spray headers, they are tested on recirculation flow. This test confirms pump performance is consistent with the pump design curve and is indicative of overall performance, by setting the pump head at the baseline value and measuring the test flow. Such inservice tests confirm component OPERABILITY, trend performance, and detect incipient failures by indicating abnormal performance. The Frequency of the SR is in accordance with the Inservice Testing Program.

SR 3.6.6.5 and SR 3.6.6.6

These SRs require verification that each automatic containment spray valve actuates to its correct position and that each containment spray pump starts upon receipt of an actual or simulated actuation of a containment High-High pressure signal. SR 3.6.6.5 is not required for valves that are locked, sealed, or otherwise secured in the required position under administrative controls. SR 3.6.6.6 must be performed with the isolation velves in the spray supply lines at the containment and spray additive tank locked closed. The 18 month Frequency is based on the need to perform these Surveillances under the conditions that apply during a plant outage and the potential for an unplanned transient if the Surveillances were performed with the reactor at power. Operating experience has shown that these components usually pass the Surveillances when performed at the 18 month Frequency. Therefore, the Frequency was concluded to be acceptable from a reliability standpoint.

(continued)

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.6.6.7

This SR requires verification that each containment cooling train actuates upon receipt of an actual or simulated safety injection signal. The 18 month Frequency is based on engineering judgment and has been shown to be acceptable through operating experience. See SR 3.6.6.5 and SR 3.6.6.6, above, for further discussion of the basis for the 18 month Frequency.

SR 3.6.6.8

With the containment spray inlet valves closed and the spray header drained of any solution, low pressure air or smoke can be blown through test connections. This SR ensures that each spray nozzle is unobstructed and provides assurance that spray coverage of the containment during an accident is not degraded. Due to the passive design of the nozzle, a test at 10 year intervals is considered adequate to detect obstruction of the nozzles.

REFERENCES

1. UFSAR, Section 3.1.
2. 10 CFR 50, Appendix K.
3. UFSAR, Section 6.2.
4. UFSAR, Section 9.4.
5. ASME, Boiler and Pressure Vessel Code, Section XI.

SUPPLEMENT 5
CONVERSION PACKAGE SECTION 3.7
PAGE INSERTION INSTRUCTIONS

Remove and insert the following pages into Enclosure 15 to Serial: RNP-RA/96-0141.

	Remove Page	Insert Page
a.	Part 1, "Markup of Current Technical Specifications (CTS)"	
	3.4-1, 3.4-2, 4.8-1	3.4-1, 3.4-2, 4.8-1
b.	Part 2, "Discussion of Changes (DOCs) for CTS Markup"	
	7	7
	-	7a
	11	11
	-	11a
	23	23
	-	23a
c.	Part 3, " No Significant Hazards Consideration (NSHC), And Basis for Categorical Exclusion from 10 CFR 51.22	
	4 & 5	4 & 5
	-	5a
d.	Part 4, "Markup of NUREG-4131, Revision 1, Standard Technical Specifications-Westinghouse Plants, (ISTS)"	
	3.7-1	3.7-1
	Insert 3.7.1-1 (no page number)	3.7-1a
	3.7-3, 3.7-13	3.7-3, 3.7-13
e.	Part 5, "Justification of Differences (JFDs) to ISTS"	
	1 through 4	1 through 4
	-	5 & 6
f.	Part 6, "Markup of ISTS Bases"	
	B 3.7-1, B 3.7-2	B 3.7-1, B 3.7-2
	Insert B 3.7.1-1 (no page number)	B 3.7-2a
	B 3.7-3	B 3.7-3
	Insert B 3.7.1-2 (no page number)	B 3.7-3a
	B 3.7-4, B 3.7-5, B 3.7-6, B 3.7-29	B 3.7-4, B 3.7-5, B 3.7-6, B 3.7-29
	-	B 3.7-29a
g.	Part 7, "Justification for Differences (JFDs) to ISTS Bases"	
	1, 6, 7, 9	1, 6, 7, 9
	10	10

SUPPLEMENT 5
CONVERSION PACKAGE SECTION 3.7
PAGE INSERTION INSTRUCTIONS

Remove and insert the following pages into Enclosure 15 to Serial: RNP-RA/96-0141.

	Remove Page	Insert Page
h.	Part 8, "Proposed HBRSEP, Unit No. 2 ITS"	
	3.7-1, 3.7-2, 3.7-3	3.7-1, 3.7-2, 3.7-3
	-	3.7-3a & 3.7-3b
	3.7-11, 3.7-12	3.7-11, 3.7-12
i.	Part 9. "Proposed Bases to HBRSEP, Unit No. 2 ITS Bases"	
	B 3.7-1 through B 3.7-6	B 3.7-1 through B 3.7-6
	-	B 3.7-6a
	B 3.7-27 through B 3.7-30	B 3.7-27 through B 3.7-30
j.	Part 10. "ISTS Generic Changes"	
	NA	

ITS

Specification 3.7.1

A1

3.4 SECONDARY STEAM AND POWER CONVERSION SYSTEM

Applicability

Applies to the operating status of turbine cycle.

Objective

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

Specification

A2

MODES 1, 2, 3

[Applicability] 3.4.1 The reactor coolant shall not be heated above 350°F unless the following conditions are met:

[LCO 3.7.1]

a. A minimum turbine cycle steam relieving capability of twelve (12) main steam safety valves operable. (shall be) (as specified in Tables 3.7.1-1 and 3.7.1-2) L1

b. Three auxiliary feedwater pumps must be operable. See 3.7.4

c. A minimum of 35,000 gallons of water in the condensate storage tank and an unlimited water supply from the lake via either leg of the plant Service Water System. See 3.7.5

d. Essential features including system piping and valves directly associated with the above components are operable. A3

e. The main steam stop valves are operable and capable of closing in five seconds or less. See 3.7.2

Add Table 3.7.1-1 and ACTIONS A & B — L1

Add ACTIONS "Note" — L2

Amendment No. 74

Supplement 5

ITS

A1

3.4.2 The specific activity of the secondary coolant system shall be ≤ 0.10 μCi/gram dose equivalent I-131 under all modes of operation from cold shutdown through power operation. When the specific activity of the secondary coolant system is > 0.10 μCi/gram dose equivalent I-131, be in at least HOT SHUTDOWN within 6 hours and cold shutdown within the following 30 hours.

See 3.7.15

The specific activity of the secondary coolant system shall be determined to be within the limit by performance of the sampling and analysis program of Table 4.1-2.

M1

MODES 1, 2, 3

[ACTION C] 3.4.3 If, during ~~power operations~~, any of the specifications in 3.4.1, with the exception of 3.4.1.b and 3.4.1.d as it applies to 3.4.1.b above, cannot be met within ~~24~~ 4 hours, the operator shall initiate procedures to put the plant in the hot shutdown condition. ~~If any of these specifications cannot be met within an additional 48 hours, the operator shall cool the reactor below 350°F using normal procedures.~~

See 3.7.4

M2

MODE 3 - 6hrs
MODE 4 - 12hrs

3.4.4 With one auxiliary feedwater pump and/or essential features inoperable, restore that auxiliary feedwater pump and/or essential features to operable status within 72 hours, or:

a. Submit a Special Report to the Commission within 30 days outlining the cause of the inoperability and the action taken to return the pump and/or essential features to operable status, and;

b. Restore all three auxiliary feedwater pumps and their essential features to operable status within 7 days or be in at least hot shutdown within 6 hours.

3.4.5 With two auxiliary feedwater pumps inoperable, restore at least one inoperable auxiliary feedwater pump to operable status within 24 hours or be in at least hot shutdown within 6 hours.

or 1 or more SGs w/ ≥ 3 MSSVs inoperable

M3

ITS

4.8 AUXILIARY FEEDWATER SYSTEM

Applicability

Applies to periodic testing requirements of the turbine-driven and motor-driven auxiliary feedwater pumps.

Objective

To verify the operability of the auxiliary feedwater system and its ability to respond properly when required.

Specification

31 days on a STAGGERED TEST BASIS

[SR 3.7.4.2] 4.8.1 Each motor driven auxiliary feedwater pump will be started at monthly intervals, run for 15 minutes, and determined that it is operable developed head ≥ req'd head and Note 2

[SR 3.7.4.2] 4.8.2 The steam turbine driven auxiliary feedwater pump by using motor operated steam admission valves will be started at monthly intervals, run for 15 minutes, and determined that it is operable when the reactor coolant system is above the cold shutdown condition. When periods of reactor cold shutdown extend this interval beyond one month, the test shall be performed within 24 hours of achieving stable plant conditions at ≥1000 psig in the steam generator following plant heatup.

[SR 3.7.4.2 NOTE 1]

automatic

[SR 3.7.4.3] 4.8.3 The auxiliary feedwater pumps discharge valves will be tested by operator action at monthly intervals.

18

4.8.4 These tests shall be considered satisfactory if control board indication and subsequent visual observation of the equipment demonstrate that all components have operated properly.

actual or simulated actuation signal

that are not locked, sealed, or otherwise secured in position

Add SR 3.7.4.1
SR 3.7.4.4
SR 3.7.4.5
SR 3.7.4.6

L6 L20 M11 L20 F1 M12 L7 A12 A13 M12 M13

TECHNICAL CHANGES - MORE RESTRICTIVE

M1 CTS Specification 3.4.3 has Applicability for Required Actions during "power operations" in lieu of the CTS Specification 3.4.1 Applicability of "reactor coolant temperature greater than 350°F." ITS Specifications 3.7.1 and 3.7.2 have Applicability of MODES 1, 2, and 3, which covers a broader operational band. MSSVs are needed for SG overpressure protection in MODES 1, 2 and 3. In MODES 4 and 5, there are no credible transients requiring the MSSVs. The steam generators are not normally used for heat removal in MODES 5 and 6, and thus cannot be overpressurized; there is no requirement for the MSSVs to be OPERABLE in these MODES. This change is more restrictive, and has no adverse impact on safety.

M2 CTS Specification 3.4.3 requires that, if the Specification cannot be met within 24 hours, the operator initiate procedures to place the unit in the hot shutdown condition, and if the Specification cannot be met in an additional 48 hours, the reactor be cooled to below 350°F. ITS Specification 3.7.1 requires that, if Required Actions and associated Completion Times are not met, the unit be placed in MODE 3 in 6 hours, and in MODE 4 in 12 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required unit conditions from full power conditions in an orderly manner and without challenging unit systems. This change is more restrictive, and has no adverse impact on safety.

M3 CTS Specification 3.4.3 is revised to adopt the ISTS 3.7.1 Condition C, where in the event one or more SGs with ≥ 3 MSSVs inoperable, the unit must be placed in MODE 3 in 6 hours and in MODE 4 in 12 hours. If the MSSVs cannot be restored to OPERABLE status within the associated Completion Time, or if one or more steam generators with ≥ 3 MSSVs inoperable, the unit must be placed in a MODE in which the LCO does not apply. To achieve this status, the unit must be placed in at least MODE 3 within 6 hours, and in MODE 4 within 12 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required unit conditions from full power conditions in an orderly manner and without challenging unit systems. Since this change imposes new requirements, it is more restrictive and has no adverse impact on safety.

M4 CTS Specification 3.4.3 requires that the LCO be met for all MODES within 24 hours, or the plant be put in the hot shutdown condition, and that if the LCO cannot be met in an additional 48 hours, the reactor be cooled to below 350°F. ITS Specification 3.7.2 Action A requires that for one MSIV inoperable in MODE 1 the LCO must be satisfied within 24 hours or Action B requires the plant must be placed in MODE 2 in 6 hours. If the MSIV cannot be restored to OPERABLE status within 24 hours, the unit must be placed in a MODE in which the LCO does not

M11 CTS Specifications 4.8.1 and 4.8.2 require that the AFW pumps be run for 15 minutes to determine that the pumps are OPERABLE. ITS SR 3.7.4.2 requires that the AFW pumps be run to determine that the developed head is greater than or equal to the required developed head. Because it is undesirable to introduce significant amounts of cold AFW into the steam generators while they are operating, Note 2 modifies ITS SR 3.7.4.2 and indicates that tests may be performed by setting the pump head at the baseline value and measuring the test flow, in lieu of the testing specified in ITS SR 3.7.4.2. Verifying that each AFW pump's developed head at the flow test point is greater than or equal to the required developed head or performance of the alternate tests in accordance with Section XI of ASME Code ensures that AFW pump performance has not degraded during the cycle. This change imposes more restrictive requirements necessary to ensure the AFW pumps are maintained OPERABLE, and has no adverse impact on safety.

M12 CTS Specification 4.8.3 requires that AFW pump discharge valves be tested. ITS SR 3.7.4.3 requires that all automatic valves (that are not locked, sealed, or otherwise secured in position) be tested. This SR verifies that AFW can be delivered to the appropriate steam generator in the event of any accident or transient that generates an AFW actuation signal, by demonstrating that each automatic valve in the flow path actuates to its correct position on an actual or simulated actuation signal. This change imposes more restrictive requirements, and has no adverse impact on safety.

M13 The CTS is revised to adopt ITS SR 3.7.4.1, SR 3.7.4.4 and SR 3.7.4.5 to provide assurance that AFW valves are in the correct position, that AFW pumps start automatically when required, and that required flow paths are properly aligned. Also, ITS SR 3.7.4.6 is added to ensure OPERABILITY of the "swing" motor driven AFW flow path. SR 3.7.4.1 requires verifying the correct alignment for manual, power operated, and automatic valves in the AFW System water and steam supply flow paths. This provides assurance that the proper flow paths will exist for AFW operation. SR 3.7.4.4 verifies that the AFW pumps will start in the event of any accident or transient that generates an AFW actuation signal by demonstrating that each AFW pump starts automatically on an actual or simulated actuation signal in MODES 1, 2, and 3. In MODE 4, the autostart function is not required. SR 3.7.4.5 verifies proper AFW System alignment and flow path OPERABILITY from the CST to each SG following extended outages to determine that no misalignment of valves has occurred. Since this change imposes new requirements, it is more restrictive and has no adverse impact on safety.

M14 CTS Specification 3.4.1 requires that the reactor coolant not be heated above 350°F unless the CST is OPERABLE. ITS Specification 3.7.5 has Applicability in MODES 1, 2, 3, and 4; and when SGs are being used for

heat removal. In MODE 4 the AFW System may need to be used for heat removal via the steam generators. The CST is necessary for OPERABILTY

TECHNICAL CHANGES - LESS RESTRICTIVE (SPECIFIC)

L1 CTS Specification 3.4.1.a requires that 12 main steam safety valves (MSSVs) be OPERABLE. ITS Specification 3.7.1 requires MSSVs to be OPERABLE as specified in Tables 3.7.1-1 and 3.7.1-2. Table 3.7.1-1 permits fewer MSSVs to be OPERABLE at reduced power levels. This is a relaxation of requirements, which is less restrictive. This change is acceptable, however, because Section III of the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code permits operation with fewer than 12 MSSVs OPERABLE as long as THERMAL POWER is limited, using a conservative heat balance calculation, such that the relief capacity of the MSSVs remaining OPERABLE. This is accomplished by restricting THERMAL POWER so that the energy transfer to the most limiting steam generator is not greater than the available relief capacity in that steam generator.

In the case of only a single inoperable MSSV on one or more steam generators when the Moderator Temperature Coefficient (MTC) is not positive, a reactor power reduction alone is sufficient to limit primary side heat generation such that overpressurization of the secondary side is precluded for any RCS heatup event. In addition, for this case there is sufficient total steam flow capacity provided by the turbine and remaining OPERABLE MSSVs to preclude overpressurization in the event of an increased reactor power due to reactivity insertion, such as in the event of an uncontrolled RCCA bank withdrawal at power. Therefore, ITS 3.7.1 Required Action A.1 is provided to require an appropriate reduction in reactor power within 4 hours.

In the case of multiple inoperable MSSVs on one or more steam generators, with a reactor power reduction alone there may be insufficient total steam flow capacity provided by the turbine and remaining OPERABLE MSSVs to preclude overpressurization in the event of an increased reactor power due to reactivity insertion, such as in the event of an uncontrolled RCCA bank withdrawal at power. In addition, for a single inoperable MSSV on one or more steam generators when the MTC is positive, the reactor power may increase as a result of an RCS heatup event such that flow capacity of the remaining OPERABLE MSSVs is insufficient. Therefore, in addition to ITS 3.7.1 Required Action B.1, which requires an appropriate reduction in reactor power within 4 hours, ITS 3.7.1 Required Action B.2 is provided to require an appropriate reduction in the Power Range Neutron Flux-High reactor trip setpoint within 72 hours. These power level and trip setpoint reduction limitations are specified in Table 3.7.1-1.

L2 CTS Specification 3.4.1 requires the plant to be shutdown if the requirements for OPERABILITY of the MSSVs are not met within 24 hours. A Note to ITS Actions 3.7.1 is added that allows separate condition entry for each MSSV. Since the CTS has no provision to increase the allowed outage time when one MSSV becomes inoperable after another, this change is less restrictive. This change is acceptable because the ITS Required Actions, after a short allowed outage time of four (4) hours, will ensure that THERMAL POWER reductions maintain the steam generator stored energy below the available relief capacity. Separate condition entry for each inoperable MSSV is necessary to allow the orderly adjustment of THERMAL POWER in response to the Required Actions. This change is consistent with NUREG-1431.

L3 CTS Specification 3.4.1 has Applicability such that the "... reactor coolant shall not be heated above 350°F." ITS Specification 3.7.2 has Applicability of MODE 1, and MODES 2 and 3 except when all Main Steam Isolation Valves (MSIVs) are closed. This change eliminates OPERABILITY requirements for MSIVs when they are closed. This is a relaxation of requirements, which is less restrictive. This change is acceptable, however, because when the MSIVs are closed, they are already performing their safety function. This change is consistent with NUREG-1431.

L4 CTS Specification 4.7.1 requires that main steam stop valves be tested at a Frequency of each refueling interval or 15 ± 3 months, whichever occurs first. ITS Specification 3.7.2 requires that the valves be tested at a Frequency in accordance with the Inservice Testing (IST) Program. This is a relaxation of requirements, which is less restrictive. This change is acceptable, however, because the IST Program currently requires testing at an 18 month Frequency, based on the refueling cycle, and which is an acceptable Frequency for this

The proposed changes do not involve any physical alteration of plant systems, structures or components or changes in parameters governing normal plant operation. These changes do not introduce a new mode of plant operation. Since any future changes to these requirements will be evaluated in accordance with licensee controlled programs, the possibility of a new or different kind of accident from any accident previously evaluated will not be permitted without further NRC review. Therefore, these changes do not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

The proposed changes will not reduce a margin of safety because they do not impact any safety analysis assumptions. The requirements that are transposed from the Technical Specifications to other licensee controlled documents are the same as the existing Technical Specifications. Since any future changes to these requirements will be evaluated in accordance with the requirements of licensee controlled programs, no reduction in any margin of safety will be permitted without further NRC review. Therefore, these changes do not involve any reduction in a margin of safety.

LESS RESTRICTIVE SPECIFIC CHANGES
("L1" Labeled Comments/Discussions)

Carolina Power & Light Company 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 safety function of the Main Steam Safety Valves (MSSVs) is to limit pressure in the steam generator to 110% of design pressure when passing 100% of design steam flow, and thus perform an accident mitigation function. The MSSVs form part of the pressure boundary of the steam generators and main steam lines. The inadvertent opening or failure of an MSSV to retain system pressure is bounded by the Steam Line Break analysis, and the number of OPERABLE MSSVs (i.e., able to perform the pressure relief function) does not affect the probability of an analyzed accident. The ITS restricts THERMAL POWER to reduced power levels and requires reduction of Power Range Neutron Flux-High trip setpoints, as applicable, in response to one or more inoperable MSSVs in accordance with the ASME Code, thereby preserving the design criteria and safety function of the MSSVs. Therefore the proposed change does not involve an increase in the probability or consequences of an accident previously evaluated.

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

   The proposed change does not involve any physical alteration of plant systems, structures or components, changes in parameters governing normal plant operation, or methods of operation. Continued operation with inoperable MSSVs is not permitted without a reduction in THERMAL POWER and Power Range Neutron Flux-High trip setpoints, as applicable. The proposed change does not introduce a new mode of operation or changes in the method of normal plant operation. Therefore, the possibility of a new or different kind of accident from any accident previously evaluated is not created.

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

   Operation in MODE 1 with fewer than 4 MSSVs per steam generator OPERABLE is permitted with a reduction in THERMAL POWER and Power Range Neutron Flux-High trip setpoints (as applicable), in accordance with ASME Code requirements. The reduction in THERMAL POWER and Power Range Neutron Flux-High trip setpoints, as applicable, will ensure that SG stored energy is maintained below the available relief capacity of the OPERABLE MSSVs. The SG stored energy will exceed the available relief capacity during the time period between discovery of an inoperable MSSV and completion of the THERMAL POWER and trip setpoint adjustment, as applicable, downward. This time period, however, is reasonably short. Therefore, the proposed change does not involve a significant reduction in a margin of safety.

LESS RESTRICTIVE SPECIFIC CHANGES
("L2" Labeled Comments/Discussions)

Carolina Power & Light Company 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 change will allow separate condition entry for each inoperable MSSV. The safety function of the MSSV is to limit pressure in the steam generator to 110% of design pressure when passing 100% of design steam flow, and thus perform an accident mitigation function. The MSSVs form part of the pressure boundary of the steam generators and main steam lines. The Required Actions ensure that THERMAL POWER reductions maintain the relief capacity below the remaining available relief capacity. The probability of occurrence of an accident is independent of the OPERABILITY of one or more MSSVs at any one time. The consequences of an accident during the time that one or more MSSVs is

inoperable are no greater than the consequences of an accident occurring during the 24 hour allowed outage time currently permitted for one or more inoperable MSSVs. Therefore, the proposed change does not involve an

CTS

3.7 PLANT SYSTEMS

3.7.1 Main Steam Safety Valves (MSSVs)

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

[3.4.1] APPLICABILITY: MODES 1, 2, and 3.

ACTIONS

[L2] ----------NOTE----------
Separate Condition entry is allowed for each MSSV.

	CONDITION	REQUIRED ACTION		COMPLETION TIME
[L1]	~~A. One or more required MSSVs inoperable.~~	~~A.1~~	~~Reduce power to less than or equal to the applicable % RTP listed in Table 3.7.1-1.~~	~~4 hours~~
[3.4.3]	~~B.~~ C. Required Action and associated Completion Time not met. OR One or more steam generators with ~~less than [two] MSSVs OPERABLE.~~ ≥ 3 MSSVs inoperable	~~B.1~~ C. AND ~~B.2~~ C.	Be in MODE 3. Be in MODE 4.	6 hours 12 hours

INSERT 3.7.1-1 (31)

(31)

ITS INSERT 3.7.1-1 (Main Steam Safety Valves (MSSVs))

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more steam generators with one MSSV inoperable and the Moderator Temperature Coefficient (MTC) zero or negative at all power levels.	A.1 Reduce THERMAL POWER to < 51 % RTP.	4 hours
B. One or more steam generators with two or more MSSVs inoperable. OR One or more steam generators with one MSSV inoperable and MTC positive at any power level.	B.1 Reduce THERMAL POWER to less than or equal to the Maximum Allowable % RTP specified in Table 3.7.1-1 for the number of OPERABLE MSSVs. AND B.2 ---------NOTE-------- Only required in MODE 1. ---------- Reduce the Power Range Neutron Flux - High reactor trip setpoint to less than or equal to the Maximum Allowable % RTP specified in Table 3.7.1-1 for the number of OPERABLE MSSVs.	4 hours 72 hours

CTS

[L1]

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

~~MINIMUM~~ NUMBER OF MSSVs PER STEAM GENERATOR ~~REQUIRED~~ OPERABLE	MAXIMUM ALLOWABLE ~~APPLICABLE~~ POWER (% RTP)
~~5~~	~~≤ 100~~
~~4~~ 3	~~≤ 80~~ ≤ 46 %
~~3~~ 2	~~≤ 60~~ ≤ 24 %
~~2~~	~~≤ 40~~

CTS

SURVEILLANCE REQUIREMENTS

	SURVEILLANCE	FREQUENCY
[M13] SR 3.7.5.1	Verify each AFW manual, power operated, and automatic valve in each water flow path, and in both steam supply flow paths to the steam turbine driven pump, that is not locked, sealed, or otherwise secured in position, is in the correct position.	31 days
[4.8.1] [4.8.2] SR 3.7.5.2	----------NOTES---------- 1. Not required to be performed for the turbine driven AFW pump until 24 hours after ≥ 1000 psig in the steam generator. ---------- Verify the developed head of each AFW pump at the flow test point is greater than or equal to the required developed head.	31 days on a STAGGERED TEST BASIS
[4.8.3] SR 3.7.5.3	----------NOTE---------- Not applicable in MODE 4 when steam generator is relied upon for heat removal. ---------- Verify each AFW automatic valve that is not locked, sealed, or otherwise secured in position, actuates to the correct position on an actual or simulated actuation signal.	18 months

(continued)

2. Alternate testing may be performed by setting the pump head at the baseline value, and measuring test flow to meet the requirements of SR 3.7.4.2.

Supplement 5

1 In the conversion of the HBRSEP current Technical Specifications (CTS) to the proposed plant specific Improved Technical Specifications (ITS), certain wording preferences or conventions are adopted which do not result in technical changes (either actual or interpretational). Editorial changes which involve the insertion of plant specific terms or parameters are used to preserve consistency with the CTS and licensing basis.

2 ITS Tables 3.7.1-1 and 3.7.1-2 are completed to reflect each of the 3 steam generators is equipped with 4 main steam safety valves (MSSVs). The applicable power as a percent of RATED THERMAL POWER is adjusted in accordance with the heat balance algorithm included in Westinghouse letter NSAL-94-001, "Operation at Reduced Power Levels with Inoperable MSSVs," January 20, 1994.

3 ITS 3.7.3 is modified to reflect Main Feedwater (MFW)System configuration. The valve arrangement consists of a main feedwater isolation valve (MFIV) and a main feedwater regulating valve (MFRV) in series, with a single MFRV bypass valve that bypasses both the MFIV and the MFRV. The bypass valve is used to regulate feedwater flow when feedwater flow demand is low.

4 ITS 3.7.3 Action C.1 Completion Time is reduced to 8 hours. The MFW bypass valves are single valves without redundancy. An 8 hour Completion Time is more reasonable, and is consistent with the Completion Time for Action D.1, which also addresses the condition where redundancy is lost. The 8 hour Completion Time considers the loss of automatic isolation in a flow path as a loss of isolation capability on a single steam generator, without crediting the check valve in the flow path.

5 A new Surveillance Requirement, SR 3.7.3.2, is adopted in the ITS to address MFIV closure time. The MFIVs are motor operated valves, with a closure time of ≤ 80 seconds, which is assumed in the safety analysis.

6 ISTS 3.7.4 is not adopted in the ITS. The atmospheric dump valves (ADVs) are the steam generator power operated relief valves (PORVs). The only safety related function of the PORV is to retain pressure in the steam generator. The design basis does not include a safety related PORV function to open. Subsequent Specifications are renumbered accordingly.

7 ITS 3.7.4 is modified to reflect the AFW System configuration of three pumps that supply AFW to the three steam generators through four flow paths, three of which are part of the motor driven subsystem. One flow path applies to the steam supply, suction, discharge, and injection lines associated with the steam driven AFW pump. Because of the configuration of normal and emergency power supplies to the various active components of the system, the AFW System cannot be configured

into discrete "trains," as the ISTS would imply. The LCO and associated Actions are modified, as appropriate, to reflect the plant specific configuration and current licensing basis.

8 A new Surveillance Requirement, SR 3.7.4.6, is adopted in the ITS to periodically verify operability of the automatic bus transfer switch associated with the "swing" motor driven AFW flow path.

9 ITS 3.7.5 is modified to require that the backup Service Water supply to the AFW System be OPERABLE, to assure cold shutdown capability.

10 ITS 3.7.5 Required Action A.2 Completion Time is reduced to 24 hours, consistent with current licensing basis.

11 ITS 3.7.5 is modified by adopting new ACTION C, which provides remedial actions to address the Condition when the LCO requirement to have an OPERABLE backup Service Water supply to the AFW system is not met.

12 A new Surveillance Requirement, SR 3.7.5.2, is adopted to periodically verify that the LCO requirement to have an OPERABLE backup Service Water supply to the AFW System is met.

13 ITS 3.7.6 is modified to reflect plant design basis consisting of three trains of CCW pumps, two of which are powered from emergency power supplies. LCO 3.7.6 is modified to state that the required CCW pumps are those that are "... powered from emergency power supplies ..." Additionally, to distinguish the two essential trains from the other train, the word "required" is added to Condition A, SR 3.7.6.1 and SR 3.7.6.2.

14 ISTS SR 3.7.7.2 is not adopted in the ITS. Plant design does not include automatic valves to isolate essential from nonessential portions of the system upon initiation of an actuation signal.

15 ITS 3.7.7 is modified in the LCO to require separate OPERABILITY of the turbine building service water loop isolation valves, which are not configured into separate, independent trains. New Conditions B and C are adopted to address inoperability of the turbine building service water loop isolation valves.

16 ISTS 3.7.8, Required Action A.1, Note 2, is not adopted in the ITS. The inoperability of one SWS loop does not result in the inoperability of RHR loops in MODE 4 because both SWS headers operate in a cross-tied configuration under both normal and accident conditions.

17 A new Surveillance Requirement, SR 3.7.7.4, is adopted in the ITS to periodically verify operability of the automatic bus transfer switch associated with one turbine building SWS loop isolation valve.

18 ITS 3.7.9 and ITS 3.7.10 are modified to reflect a 48 hour allowed outage time for the inoperability of two CREFS trains, and two WCCU trains, respectively, and the associated shutdown actions if one train of the respective systems is not restored within the 48 allowed outage time. These changes are being made for consistency with the current licensing basis.

19 ITS SR 3.7.9.4 is modified to reflect verification of ability to pressurize the control room habitability envelope to a "positive" pressure relative to adjacent building areas, consistent with applicable safety analyses and current licensing basis.

20 The plant design basis for the Ultimate Heat Sink (UHS) is the Lake Robinson impoundment. The Actions and Surveillance Requirements of ITS 3.7.8 are modified accordingly to reflect the plant design basis and eliminate reference to cooling towers.

21 ISTS 3.7.12 is not adopted in the ITS. The ECCS Pump Room Exhaust Air Cleanup System provides no safety function, and therefore no Technical Specifications are required.

22 ITS 3.7.11 is modified to reflect that the Fuel Building Air Cleanup System (FBACS) is a manually actuated, single train system that is required to be operating during movement of irradiated fuel assemblies in the building. The FBACS has no safety function in the mitigation of the consequences of reactor accidents. The FBACS safety function is to mitigate the consequences of a fuel handling accident in the Fuel Building.

23 ITS SR 3.7.11.3 is modified to reflect the FBACS safety function as maintaining the atmospheric pressure in the Fuel Building "negative" with respect to the outside atmosphere to assure that any airborne radioactivity resulting from a fuel handling accident is passed through the FBACS prior to release to the atmosphere. The offsite total dose consequences were analyzed assuming a total release of activity from the fuel handling accident. Therefore, the consequences of a fuel handling accident are unrelated to the system flow rate as long as the building pressure remains negative with respect to the outside atmosphere. Consequently, there is no design requirement for the FBACS to maintain a specific negative pressure at any specific flow rate.

24 ISTS 3.7.14 is not adopted in the ITS. Plant design basis does not include a Penetration Room Exhaust Air Cleanup System, and therefore no Technical Specifications are required.

25 ITS 3.7.12 is modified to reflect that the fuel storage pool minimum level of 21 feet is consistent with the assumptions for iodine decontamination factors used in the fuel handling accident analysis, and bounds the sensible heat sink assumptions used in time-to-boil

calculations. There are certain pool situations where more than 21 feet, but less than 23 feet occur. This is due to physical dimensions of the racks and pool , and the setpoints used for control room level alarms. Only about 22 feet of water above the highest spent fuel rack may be available. The value of 21 feet was selected for the ITS limit to allow for brief variations about the level of 22 feet.

26 ITS 3.7.13 Applicability is modified to require boron concentration to meet requirements during new and spent fuel movement activities in the fuel storage pool, consistent with current licensing basis. This ensures that K_{eff} remains within the analyses when fuel movement activity is taking place and takes credit for the dual verification that occurs during movement of new or spent fuel in the fuel storage pool. The Required Action when the LCO is not met is to suspend movement of fuel assemblies in the fuel storage pool, which places the Specification in a mode in which it is no longer Applicable, and therefore Required Action A.2.1 is unnecessary. The provision in the ISTS to limit Applicability of the LCO to only those times when verification has not been performed following the last movement of fuel assemblies does not apply, nor does Required Action A.2.2.

27 ITS Specification 3.7.14 is modified to reflect the current new and spent fuel storage requirements. Since specific design requirements are applied to the storage of new fuel to prevent maximum new fuel packing that would result in new fuel storage outside the assumptions of the new fuel storage criticality analysis, ISTS 3.7.14 was modified to include new fuel storage in addition to spent fuel storage in order to provide a Required Action and a Surveillance Requirement to the storage of new fuel. Additionally, since the spent fuel storage criticality analysis is not dependent on fuel burnup, the ISTS format for the LCO and referenced figure is not adopted in ITS. The only limitations on spent fuel storage are fuel assembly configuration restrictions provided in Updated Final Safety Analysis Report (UFSAR) Table 9.1.2-2, that apply to locations in either the high or low density spent fuel storage racks. The details of fuel assembly configurations and locations are appropriately controlled as currently included in the UFSAR. Therefore, the resulting ITS LCO 3.7.14 is written to provide an LCO requirement to store new and spent fuel in approved locations, and provide the necessary required action and surveillance requirement.

28 ITS SR 3.7.4.5 is modified by a Note that allows entry into and operation in MODE 3 and MODE 2 prior to performing the SR for the steam driven AFW pump. This is necessary because sufficient decay heat is not available following an extended outage. The unit must be at a point of adding minimum core heat in order to provide sufficient steam to operate the steam driven AFW pump to verify water flow.

29 ISTS 3.7.13.1 requires (for plants with heaters) each FBACS train to be operated for ≥ 10 continuous hours with heaters operating. The wording

of ISTS 3.7.13.1 is revised in ITS 3.7.11.1 to require the FBACS to be operated for ≥ 10 continuous hours with the heaters operating automatically. This change is necessary to reflect the HBRSEP Unit No. 2 design of the Fuel Building Air Cleanup System (the system consists of a single train) and the fact that the heaters cycle on and off automatically to control humidity.

30 ISTS SR 3.7.5.2 requires verification that each AFW pump's developed head at the test flow point is greater than or equal to the required developed head. This requirement is interpreted as requiring full flow testing. ITS SR 3.7.4.2 requires the same testing to be performed but is modified by Note 2 which states "Testing may be performed by setting the pump head and measuring the test flow to meet the requirements of SR 3.7.4.2." The HBRSEP Unit No. 2 AFW design does not provide the capability to do full flow testing of AFW pumps during all applicable MODES of operation since it is undesirable to introduce significant amounts of cold AFW into the steam generators while they are operating. However, ASME Section XI provides alternate testing requirements for these circumstances. Therefore, for reduced flow or recirculation pump testing performed in applicable MODES of operation, alternate testing in accordance with ASME Section XI is allowed in lieu of the full flow testing required by ITS SR 3.7.4.2. This change is consistent with current plant practice.

31 Changes to ISTS 3.7.1 (ITS 3.7.1), Main Steam Safety Valves (MSSVs), are proposed to address recent issues related to improper Bases assumptions and overpressurization scenarios with inoperable MSSVs. The changes are necessary to address the following:

a. ISTS 3.7.1 and associated Bases for requiring a reduction in reactor power proportional to the relief capacity of the remaining OPERABLE MSSVs is incorrect. As described in Westinghouse letter NSAL-94-001, "Operation at Reduced Power Levels with Inoperable MSSVs," January 20, 1994, and NRC Information Notice 94-60, "Potential Overpressurization of Main Steam System," August 22, 1994, the required reduction in reactor power is not directly proportional to the reduction in MSSV relieving capacity due to the effects of certain reactor trips that occur at full power which may not occur at partial power conditions. ISTS 3.7.1 and the associated Bases are revised to employ the heat balance algorithm included in NSAL-94-001.

b. For operation at partial power levels with a positive Moderator Temperature Coefficient (MTC), changes are made to require a reduction in the Power Range Neutron Flux-High reactor trip setpoint in addition to a reduction in reactor power when the MTC is positive. This is necessary to limit the primary side heat generation that may occur during an RCS heatup event. With a positive MTC, a heatup of the coolant will result in a power

increase which requires additional steam relieving capacity.

c. Changes are made to require a reduction in the Power Range Neutron Flux-High trip setpoint in addition to a reduction in reactor power when there is more than one inoperable MSSV on any single steam generator. This addresses a recently identified Westinghouse issue. For reactivity insertion events such as an uncontrolled RCCA bank withdrawal from a partial power level, the reactor power will increase during the transient until a reactor trip occurs on Overtemperature ΔT or Power Range Neutron Flux-High. With more than one inoperable MSSV on any steam generator, the combined steam flow capacity of the OPERABLE MSSVs and the turbine may be insufficient in some cases to prevent overpressurization of the Main Steam System prior to reaching the reactor trip setpoint.

d. Changes are made to statements in the Bases that are misleading or inconsistent with safety analysis methods.

A generic change has been submitted for the above described changes.

B 3.7 PLANT SYSTEMS

B 3.7.1 Main Steam Safety Valves (MSSVs)

BASES

BACKGROUND

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

~~Five~~ Four MSSVs are located on each main steam header, outside containment, upstream of the main steam isolation valves, as described in the FSAR, Section ~~10.3.1~~ 10.3.2 (Ref. 1). The MSSVs ~~capacity criteria is 110% of rated steam flow at~~ 110% of the steam generator design pressure, ~~This meets~~ in accor. to the requirements of the ASME Code, Section III (Ref. 2). The MSSV design includes staggered setpoints, according to Table 3.7.1-2 in the accompanying LCO, so that only the needed valves will actuate. Staggered setpoints reduce the potential for valve chattering that is due to steam pressure insufficient to fully open all valves following a turbine reactor trip.

U

must have sufficient capacity to limit the secondary system pressure to ≤

or

2 65 3

APPLICABLE SAFETY ANALYSES

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

65

The events that challenge the relieving capacity of the MSSVs, and thus RCS pressure, are those characterized as decreased heat removal events, which are presented in the FSAR, Section ~~15.2~~ 15.2.8 (Ref. 3). Of these, the ~~turbine trip without steam dump~~ loss of external electrical load is the limiting AOO. This event also terminates normal feedwater flow to the steam generators.

U

4

The transient response for ~~turbine trip without a direct reactor trip~~ occurring from full power presents no hazard to the integrity of the RCS

Safety analysis demonstrates that

(continued)

65

BASES

APPLICABLE SAFETY ANALYSES (continued)

or the Main Steam System. If a minimum reactivity feedback is assumed, the reactor is tripped on high pressurizer pressure. In this case, the pressurizer safety valves open, and RCS pressure remains below 110% of the design value. The MSSVs also open to limit the secondary steam pressure.

INSERT B 3.7.1-1

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

The MSSVs satisfy Criterion 3 of the NRC Policy Statement.

LCO

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

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

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

Move to end of second ¶ in SR 3.7.1.1, Pg B 3.7-5

(continued)

ITS INSERT B 3.7.1-1 (Main Steam Safety Valves (MSSVs))

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

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

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

BASES

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

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

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

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

A.1

action must be taken

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

Operation with less than all ~~five~~ four MSSVs OPERABLE for each steam generator is permissible, if THERMAL POWER is proportionally limited to the relief capacity of the remaining MSSVs. This is accomplished by restricting THERMAL POWER so that the energy transfer to the most limiting steam generator is not greater than the available relief capacity in that steam generator. For example, if one MSSV is inoperable in one steam generator, the relief capacity of that steam generator is reduced by approximately 20%. To offset this reduction in relief capacity, energy transfer to that steam generator must be similarly reduced by at least 20%. This is accomplished by reducing THERMAL POWER by at least 20%, which conservatively limits the energy transfer to all steam generators to approximately 80% of total capacity, consistent with the relief capacity of the most limiting steam generator.

Insert B 3.7.1-2

(continued)

ITS INSERT B 3.7.1-2 (Main Steam Safety Valves (MSSVs))

A.1

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

The maximum THERMAL POWER corresponding to the heat removal capacity of the remaining OPERABLE MSSVs is determined via a conservative heat balance calculation as described in the attachment to Reference 4, with an appropriate allowance for calorimetric power uncertainty.

B.1 and B.2

In the case of multiple inoperable MSSVs on one or more steam generators, with a reactor power reduction alone there may be insufficient total steam flow capacity provided by the turbine and remaining OPERABLE MSSVs to preclude overpressurization in the event of an increased reactor power due to reactivity insertion, such as in the event of an uncontrolled RCCA bank withdrawal at power. Furthermore, for a single inoperable MSSV on one or more steam generators when the MTC is positive, the reactor power may increase as a result of an RCS heatup event such that flow capacity of the remaining OPERABLE MSSVs is insufficient. Therefore, in addition to Required Action B.1, which specifies an appropriate reduction in reactor power within 4 hours, Required Action B.2 specifies that the Power Range Neutron Flux-High reactor trip setpoint be reduced within 72 hours.

The maximum THERMAL POWER corresponding to the heat removal capacity of the remaining OPERABLE MSSVs is determined via a conservative heat balance calculation as described in the attachment to Reference 4, with an appropriate allowance for Nuclear Instrumentation System trip channel uncertainties.

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

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

BASES

ACTIONS

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A.1 (continued)

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

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

where:

A = the relief capacity of the MSSV; and

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

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

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

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

where:

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

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

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

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

(continued)

BASES

ACTIONS (continued) -

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

Required Actions are not completed

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

≥ 3 inoperable MSSVs

1

65

SURVEILLANCE REQUIREMENTS

SR 3.7.1.1

This SR verifies the OPERABILITY of the MSSVs by the verification of each MSSV lift setpoint in accordance with the Inservice Testing Program. The ASME Code, Section XI (Ref. ~~4~~ 5), requires that safety and relief valve tests be performed in accordance with ANSI/ASME OM-1-198~~8~~ 6 (Ref. ~~5~~ 6). According to Reference ~~5~~ 6, the following tests are required:

a. Visual examination;

b. Seat tightness determination;

c. Setpoint pressure determination (lift setting);

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

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

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

Insert sentence from LCO Bases pg. B 3.7-2

This SR is modified by a Note that allows entry into and operation in MODE 3 prior to performing the SR. The MSSVs may be either bench tested or tested in situ at hot

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(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.7.1.1 (continued)

conditions using an assist device to simulate lift pressure. If the MSSVs are not tested at hot conditions, the lift setting pressure shall be corrected to ambient conditions of the valve at operating temperature and pressure.

REFERENCES

1. FSAR, Section 10.3.1 10.3.2
2. ASME, Boiler and Pressure Vessel Code, Section III, Article NC-7000, Class 2 Components
3. FSAR, Section 15.2.8
4. ASME, Boiler and Pressure Vessel Code, Section XI.
5. ANSI/ASME OM-1-1987
6. NRC Information Notice 94-60, "Potential Overpressurization of Main Steam System," August 22, 1994.

BASES (continued)

SURVEILLANCE REQUIREMENTS

SR 3.7.6.1

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

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

SR 3.7.6.2

Verifying that each AFW pump's developed head at the flow test point is greater than or equal to the required developed head ensures that AFW pump performance has not degraded during the cycle. Flow and differential head are normal tests of centrigufal pump performance required by Section XI of the ASME Code (Ref. 2). Because it is undesirable to introduce cold AFW into the steam generators while they are operating, this testing is performed on recirculation flow. This test confirms one point on the pump design curve and is indicative of overall performance. Such inservice tests confirm component OPERABILITY, trend performance, and detect incipient failures by indicating abnormal performance. Performance of inservice testing discussed in the ASME Code, Section XI (Ref. 2) (only required at 3 month intervals) satisfies this requirement. The 31 day Frequency on a STAGGERED TEST BASIS results in testing each pump once every 3 months, as required by Reference 2.

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

(continued)

Supplement 5

Insert B 3.7.4-9

Because it is undesirable to introduce significant amounts of cold AFW into the steam generators while they are operating, Note 2 indicates alternate tests may be performed by setting the pump head at the baseline value and measuring flow to meet the requirements in SR 3.7.4.2. This alternate testing is permitted by Section XI of the ASME Code.

1 In the conversion of the HBRSEP current Technical Specifications (CTS) to the proposed plant specific Improved Technical Specifications (ITS), certain wording preferences or conventions are adopted which do not result in technical changes (either actual or interpretational). Editorial changes which involve the insertion of plant specific terms or parameters are used to preserve consistency with the CTS and licensing basis.

2 Bases 3.7.1 are modified to reflect each of the 3 steam generators is equipped with 4 main steam safety valves (MSSVs). Four MSSVs are assumed in the safety analysis. The applicable power as a percent of RATED THERMAL POWER is adjusted in accordance with the heat balance algorithm included in Westinghouse letter NSAL-94-001, "Operation at Reduced Power Levels with Inoperable MSSVs," January 20, 1994.

3 Bases text is modified for clarity, consistency, or to correct a typographical or grammatical error.

4 Bases 3.7.1 are modified to reflect that the limiting anticipated operational occurrence (AOO) is the loss of external electrical load event; and that no MSSV failures are assumed in the accident analysis.

5 Bases 3.7.1 are modified to reflect that the approved third ten year inservice inspection program references ANSI/ASME OM-1-1981, and that the MSSVs are not equipped with balancing devices.

6 Bases 3.7.1 reference to ASME Boiler and Pressure Vessel Code is modified to reflect the codes in effect when the components were designed.

7 Bases 3.7.2 are modified to reflect that the MSIVs are designed as air operated stop check valves, with an air operator to maintain the check valves open against normal steam flow. The design basis failure is to fail as is. A downstream check valve prevents reverse flow in the event of an AOO or accident. The MSIV bypass valves do not receive an auto close signal, and are normally closed.

8 Bases 3.7.2 are modified to reflect plant specific safety analyses.

9 Bases 3.7.3 are modified to reflect plant specific Main Feedwater System configuration. The valve arrangement consists of a main feedwater isolation valve (MFIV) and a main feedwater regulating valve (MFRV) in

41 A new Surveillance Requirement, SR 3.7.7.4, is adopted to periodically verify operability of the automatic bus transfer switch associated with one Turbine Building SWS loop isolation valve.

42 Plant design basis for Ultimate Heat Sink (UHS) is the Lake Robinson impoundment. The Bases are modified accordingly to reflect plant basis, and eliminate reference to cooling towers.

43 Bases 3.7.8 are modified to incorporate the underlying assumptions for the 30 day water supply from the UHS. Additional References are also provided.

44 Bases 3.7.9 are modified to reflect the design basis and configuration. Specifically, the emergency pressurization mode of operation of the CREFS is a mode in which the system automatically actuates to an emergency air recirculation and pressurization condition upon a Safety Injection (SI) signal or a control room area high radiation signal. Operation in this mode provides filtered outside air to the control room at a rate which maintains the control room envelope at a positive pressure in relation to adjacent areas and 0.125" water gauge relative to the outside atmosphere. The system is designed to meet control room habitability requirements during a design basis accident while sustaining a single active failure.

45 CREFS is designed for design basis accident radiological conditions, and for response to a fuel handling accident. CREFS is not designed for chemical or toxic gas conditions or for a release from the waste gas decay tank.

46 Bases 3.7.9 are modified to reflect a 48 hour allowed outage time for the inoperability of two CREFS trains, consistent with current licensing basis.

47 The CREFS design basis does not include heating units.

48 CREFS filter testing program is described in ITS Section 5.5.11, "Ventilation Filter Testing Program (VFTP)." The program includes a commitment to Regulatory Guide 1.52, Revision 2, March 1978, Positions C.5 and C.6 only.

49 Bases 3.7.9 are modified to provide additional support the Frequency for performance of SR 3.7.9.3, and to be consistent with other ISTS Surveillances.

50 Bases 3.7.10 are modified to reflect the design basis and configuration. Descriptive information for the control room water cooled condensing units (WCCUs) is provided. Specifically, the control room air conditioning heat removal equipment consists of two independent trains of refrigeration equipment, with the exception of the shared SWS supply to the WCCU subsystem. Each train is sized to remove the design heat load from the control room while maintaining the control room temperature between 70°F and 77°F, (i.e., below the design limit of 85°F). One WCCU is maintained in continuous operation while the other unit is maintained in a standby auto start status. The standby unit will autostart upon a trip of the operating WCCU.

51 Bases 3.7.10 Action E is modified to reflect a 48 hour allowed outage time for the inoperability of two WCCU trains and Action F is adopted to place the plant in a MODE where the LCO is not applicable if the 48 hour AOT can not me met. These changes are consistent with current licensing basis.

52 The Bases are revised to reflect changes made to the Specification.

53 ISTS 3.7.12 is not adopted in the ITS. The ECCS Pump Room Exhaust Air Cleanup System provides no safety function, and therefore no Technical Specifications are required.

54 Bases 3.7.11 are modified to reflect that the Fuel Building Air Cleanup System (FBACS) is a manually actuated, single train system that is required to be operating during movement of irradiated fuel assemblies in the building. The FBACS has no safety function in the mitigation of the consequences of reactor accidents. The FBACS safety function is to mitigate the consequences of a fuel handling accident in the Fuel Building.

55 Bases 3.7.11 for the FBACS Applicable Safety Analyses are modified to be consistent with plant design criteria and the accident analyses provided in UFSAR Section 15.7.4.

ISTS to limit Applicability of the LCO to only those times when verification has not been performed following the last movement of fuel assemblies does not apply, nor does Required Action A.2.2.

62 Bases 3.7.14 are modified to reflect current licensing basis for new and spent fuel assembly storage.

63 Bases 3.7.15 are modified to reflect the SLB analyzed dose calculation in UFSAR Section 15.1.5.4.

64 Bases for SR 3.7.4.5 are modified to provide clarity, and to remove repetitive wording.

65 Changes to ISTS 3.7.1 (ITS 3.7.1), Main Steam Safety Valves (MSSVs), and associated Bases are proposed to address recent issues related to improper Bases assumptions and overpressurization scenarios with inperable MSSVs. The changes are necessary to address the following:

a. ISTS 3.7.1 and associated Bases for requiring a reduction in reactor power proportional to the relief capacity of the remaining OPERABLE MSSVs is incorrect. As described in Westinghouse letter NSAL-94-001, "Operation at Reduced Power Levels with Inoperable MSSVs," January 20, 1994, and NRC Information Notice 94-60, "Potential Overpressurization of Main Steam System," August 22, 1994, the required reduction in reactor power is not directly proportional to the reduction in MSSV relieving capacity due to the effects of certain reactor trips that occur at full power which may not occur at partial power conditions. ISTS 3.7.1 and the associated Bases are revised to employ the heat balance algorithm included in NSAL-94-001.

b. For operation at partial power levels with a positive Moderator Temperature Coefficient (MTC), changes are made to require a reduction in the Power Range Neutron Flux-High reactor trip setpoint in addition to a reduction in reactor power when the MTC is positive. This is necessary to limit the primary side heat generation that may occur during an RCS heatup event. With a positive MTC, a heatup of the coolant will result in a power increase which requires additional steam relieving capacity.

c. Changes are made to require a reduction in the Power Range Neutron Flux-High trip setpoint in addition to a reduction in reactor power when there is more than one inoperable MSSV on any single steam generator. This addresses a recently identified Westinghouse issue. For reactivity insertion events such as an uncontrolled RCCA bank withdrawal from a partial power level, the reactor power will increase during the transient until a reactor trip occurs on Overtemperature ΔT or Power Range Neutron Flux-High. With more than one inoperable MSSV on any steam generator,

the combined steam flow capacity of the OPERABLE MSSVs and the turbine may be insufficient in some cases to prevent overpressurization of the Main Steam System prior to reaching the reactor trip setpoint.

d. Changes are made to statements in the Bases that are misleading or inconsistent with safety analysis methods.

A generic change has been submitted for the above described changes.

3.7 PLANT SYSTEMS

3.7.1 Main Steam Safety Valves (MSSVs)

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

APPLICABILITY: MODES 1, 2, and 3.

ACTIONS

NOTE
Separate Condition entry is allowed for each MSSV.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more steam generators with one MSSV inoperable and the Moderator Temperature Coefficient (MTC) zero or negative at all power levels.	A.1 Reduce THERMAL POWER to < 51 % RTP.	4 hours

(continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
B. One or more steam generators with two or more MSSVs inoperable. OR One or more steam generators with one MSSV inoperable and MTC positive at any power level.	B.1 Reduce THERMAL POWER to less than or equal to the Maximum Allowable % RTP specified in Table 3.7.1-1 for the number of OPERABLE MSSVs.	4 hours
	AND	
	B.2 ---------NOTE-------- Only required in MODE 1 ---------- Reduce the Power Range Neutron Flux - High reactor trip setpoint to less than or equal to the Maximum Allowable % RTP specified in Table 3.7.1-1 for the number of OPERABLE MSSVs.	72 hours
C. Required Action and associated Completion Time not met. OR One or more steam generators with less than two MSSVs OPERABLE.	C.1 Be in MODE 3.	6 hours
	AND	
	C.2 Be in MODE 4.	12 hours

SURVEILLANCE REQUIREMENTS

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

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

NUMBER OF OPERABLE MSSVs PER STEAM GENERATOR	MAXIMUM ALLOWABLE POWER (% RTP)
3	≤ 46
2	≤ 24

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

VALVE NUMBER			LIFT SETTING (psig ± 3%)
STEAM GENERATOR			
A	B	C	
SV1-1A	SV1-1B	SV1-1C	1085
SV1-2A	SV1-2B	SV1-2C	1110
SV1-3A	SV1-3B	SV1-3C	1125
SV1-4A	SV1-4B	SV1-4C	1140

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.7.4.1	Verify each AFW manual, power operated, and automatic valve in each water flow path, and in the steam supply flow path to the steam driven AFW pump, that is not locked, sealed, or otherwise secured in position, is in the correct position.	31 days
SR 3.7.4.2	----------NOTES---------- 1. Not required to be performed for the steam driven AFW pump until 24 hours after ≥ 1000 psig in the steam generator. 2. Alternate testing may be performed by setting the pump head at the baseline value and measuring the flow to meet the requirements of SR 3.7.4.2. ---------- Verify the developed head of each AFW pump at the flow test point is greater than or equal to the required developed head.	31 days on a STAGGERED TEST BASIS
SR 3.7.4.3	----------NOTE---------- Not applicable in MODE 4 when steam generator is being used for heat removal. ---------- Verify each AFW automatic valve that is not locked, sealed, or otherwise secured in position, actuates to the correct position on an actual or simulated actuation signal.	18 months

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.7.4.4	----------NOTES---------- 1. Not required to be performed for the steam driven AFW pump until 24 hours after ≥ 1000 psig in the steam generator. 2. Not applicable in MODE 4 when steam generator is being used for heat removal. ---------- Verify each AFW pump starts automatically on an actual or simulated actuation signal.	18 months
SR 3.7.4.5	----------NOTE---------- Not required to be performed for the steam driven AFW pump until prior to entering MODE 1. ---------- Verify proper alignment of the required AFW flow paths by verifying flow from the condensate storage tank to each steam generator.	Prior to entering MODE 2, whenever unit has been in MODE 5 or 6 for > 30 days
SR 3.7.4.6	Verify the AFW automatic bus transfer switch associated with discharge valve V2-16A operates automatically on an actual or simulated actuation signal.	18 months

B 3.7 PLANT SYSTEMS

B 3.7.1 Main Steam Safety Valves (MSSVs)

BASES

BACKGROUND

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

Four MSSVs are located on each main steam header, outside containment, upstream of the main steam isolation valves, as described in the UFSAR, Section 10.3.2 (Ref. 1). The MSSVs must have sufficient capacity to limit the secondary system pressure to ≤ 110% of the steam generator design pressure in order to meet the requirements of the ASME Code, Section III (Ref. 2). The MSSV design includes staggered setpoints, according to Table 3.7.1-2 in the accompanying LCO, so that only the needed valves will actuate. Staggered setpoints reduce the potential for valve chattering that is due to steam pressure insufficient to fully open all valves following a turbine or reactor trip.

APPLICABLE SAFETY ANALYSES

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

The events that challenge the relieving capacity of the MSSVs, and thus RCS pressure, are those characterized as decreased heat removal events, which are presented in the UFSAR, Section 15.2 (Ref. 3). Of these, the loss of external electrical load is the limiting AOO. This event also terminates normal feedwater flow to the steam generators.

The safety analysis demonstrates that the transient response for loss of external electrical load occurring from full power presents no hazard to the integrity of the RCS or the Main Steam System.

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

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

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

The UFSAR safety analyses discussed above assume that all of the MSSVs for each steam generator are OPERABLE. If there are inoperable MSSV(s), it is necessary to limit the primary system power during steady state operation and AOOs to a value that does not result in exceeding the combined steam flow capacity of the turbine (if available) and the remaining OPERABLE MSSVs. The required limitation on primary system power necessary to prevent secondary system overpressurization is conservatively arrived at by a simple heat balance calculation. In some circumstances it is necessary to limit the primary side heat generation that can be achieved during an AOO by reducing the setpoint of the Power Range Neutron Flux-High reactor trip function. For example, if more than one MSSV on a single steam generator is inoperable, an uncontrolled RCCA bank withdrawal at power

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

event occurring from a partial power level may result in an increase in reactor power that exceeds the combined steam flow capacity of the turbine and the remaining OPERABLE MSSVs. Thus, for multiple inoperable MSSVs on the same steam generator it is necessary to prevent this power increase by lowering the Power Range Neutron Flux-High setpoint to an appropriate value. When the Moderator Temperature Coefficient (MTC) is positive, the reactor power may increase above the initial value during an RCS heatup event (e.g., turbine trip). Thus, for any number of inoperable MSSVs it is necessary to reduce the trip setpoint if a positive MTC may exist at partial power conditions.

The MSSVs satisfy Criterion 3 of the NRC Policy Statement.

LCO

The accident analysis assumes four MSSVs per steam generator are OPERABLE to provide overpressure protection for design basis transients occurring at 102% RTP. The LCO, therefore, also requires that four MSSVs per steam generator be OPERABLE.

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

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

APPLICABILITY

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

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

(continued)

BASES (continued)

ACTIONS

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

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

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

A.1

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

The maximum THERMAL POWER corresponding to the heat removal capacity of the remaining OPERABLE MSSVs is determined via a conservative heat balance calculation as described in the attachment to Reference 4, with an appropriate allowance for calorimetric power uncertainty.

B.1 and B.2

In the case of multiple inoperable MSSVs on one or more steam generators, with a reactor power reduction alone there may be insufficient total steam flow capacity provided by the turbine and remaining OPERABLE MSSVs to preclude overpressurization in the event of an increased reactor power due to reactivity insertion, such as in the event of an uncontrolled RCCA bank withdrawal at power. Furthermore,

(continued)

BASES

ACTIONS

B.1 and B.2 (continued)

for a single inoperable MSSV on one or more steam generators when the MTC is positive, the reactor power may increase as a result of an RCS heatup event such that flow capacity of the remaining OPERABLE MSSVs is insufficient. Therefore, in addition to Required Action B.1, which specifies an appropriate reduction in reactor power within 4 hours, Required Action B.2 specifies that the Power Range Neutron Flux-High reactor trip setpoint be reduced within 72 hours.

The maximum THERMAL POWER corresponding to the heat removal capacity of the remaining OPERABLE MSSVs is determined via a conservative heat balance calculation as described in the attachment to Reference 4, with an appropriate allowance for Nuclear Instrumentation System trip channel uncertainties.

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

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

C.1 and C.2

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

(continued)

BASES (continued)

SURVEILLANCE REQUIREMENTS

SR 3.7.1.1

This SR verifies the OPERABILITY of the MSSVs by the verification of each MSSV lift setpoint in accordance with the Inservice Testing Program. The ASME Code, Section XI (Ref. 5), requires that safety and relief valve tests be performed in accordance with ANSI/ASME OM-1-1981 (Ref. 6). According to Reference 6, the following tests are required:

a. Visual examination;

b. Seat tightness determination;

c. Setpoint pressure determination (lift setting);

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

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

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

REFERENCES

1. UFSAR, Section 10.3.2.

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

3. UFSAR, Section 15.2.

(continued)

BASES

REFERENCES (continued)

4. NRC Information Notice 94-60, "Potential Overpressure of Main Steam System," August 22, 1994.

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

6. ANSI/ASME OM-1-1981.

BASES

ACTIONS (continued)

F.1

In MODE 4, either the reactor coolant pumps or the RHR loops can be used to provide forced circulation. This is addressed in LCO 3.4.6, "RCS Loops – MODE 4." With one required AFW train inoperable, action must be taken to immediately restore the inoperable train to OPERABLE status. The immediate Completion Time is consistent with LCO 3.4.6.

SURVEILLANCE REQUIREMENTS

SR 3.7.4.1

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

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

SR 3.7.4.2

Verifying that each AFW pump's developed head at the flow test point is greater than or equal to the required developed head ensures that AFW pump performance has not degraded during the cycle. Flow and differential head are normal tests of centrifugal pump performance required by Section XI of the ASME Code (Ref. 4) to monitor centrifugal pump performance. This test confirms one point on the pump design curve and is indicative of overall performance. Such inservice tests confirm component OPERABILITY, trend performance, and detect incipient failures by indicating abnormal performance. Performance of inservice testing

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.7.4.2 (continued)

discussed in Reference 4 (only required at 3 month intervals) satisfies this requirement. The 31 day Frequency on a STAGGERED TEST BASIS results in testing each pump once every 3 months, as required by Reference 4.

This SR is modified by two Notes. Note 1 indicates that the SR should be deferred until suitable test conditions are established. This deferral is required because there is insufficient steam pressure to perform the test. Because it is undesirable to introduce significant amounts of cold AFW into the steam generators while they are operating, Note 2 indicates alternate pump tests may be performed by setting the pump head at the baseline value and measuring flow to meet the requirements in SR 3.7.4.2. This alternate testing is permitted by Section XI of the ASME Code.

SR 3.7.4.3

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

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

SR 3.7.4.4

This SR verifies that the AFW pumps will start in the event of any accident or transient that generates an AFW actuation

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.7.4.4 (continued)

automatically on an actual or simulated actuation signal in MODES 1, 2, and 3. In MODE 4, the autostart function is not required. The 18 month Frequency is based on the need to perform this Surveillance under the conditions that apply during a unit outage and the potential for an unplanned transient if the Surveillance were performed with the reactor at power.

This SR is modified by two Notes. Note 1 indicates that the SR be deferred until suitable test conditions are established. This deferral is required because there is insufficient steam pressure to perform the test. Note 2 states that the SR is not required in MODE 4. In MODE 4, the heat removal requirements would be less providing more time for operator action to manually start the required AFW pump.

SR 3.7.4.5

This SR verifies proper AFW System alignment and flow path OPERABILITY from the CST to each SG following extended outages to determine that no misalignment of valves has occurred. The SR is performed prior to entering MODE 2 after more than 30 days in MODE 5 or 6. OPERABILITY of AFW flow paths must be verified before sufficient core heat is generated that would require the operation of the AFW System during a subsequent shutdown. The Frequency is reasonable, based on engineering judgement and other administrative controls that ensure that flow paths remain OPERABLE.

This SR is modified by a Note that allows entry into and operation in MODE 3 and MODE 2 prior to performing the SR for the steam driven AFW pump. This is necessary because sufficient decay heat is not available following an extended outage. The unit must be at a point of adding minimum core heat in order to provide sufficient steam to operate the steam driven AFW pump to verify water flow.

SR 3.7.4.6

This SR verifies that the automatic bus transfer switch associated with the "swing" motor driven AFW flow path discharge valve V2-16A will function properly to automatically transfer the power source from the aligned

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.7.4.6 (continued)

emergency power source to the other emergency power source upon loss of power to the aligned emergency power source. The Surveillance consists of two tests to assure that the switch will perform in either direction. One test is performed with the automatic bus transfer switch aligned to one emergency power source initially, and the test is repeated with the switch initially aligned to the other emergency power source. Periodic testing of the switch is necessary to demonstrate OPERABILITY. Operating experience has shown that this component usually passes the Surveillance when performed at the 18 month Frequency. Therefore, the Frequency is acceptable from a reliability standpoint.

REFERENCES

1. UFSAR, Section 10.4.8.
2. UFSAR, Section 15.2.8.
3. UFSAR, Section 15.2.7.
2. ASME, Boiler and Pressure Vessel Code, Section XI.

SUPPLEMENT 5
CONVERSION PACKAGE SECTION 3.8
PAGE INSERTION INSTRUCTIONS

Remove and insert the following pages into Enclosure 16 to Serial: RNP-RA/96-0141.

	Remove Page	Insert Page
a.	Part 1, "Markup of Current Technical Specifications (CTS)"	
	4.1-10a, 4.6-3	4.1-10a, 4.6-3
b.	Part 2, "Discussion of Changes (DOCs) for CTS Markup"	
	6	6
	-	6a
	14 & 21	14 & 21
c.	Part 3, " No Significant Hazards Consideration (NSHC), And Basis for Categorical Exclusion from 10 CFR 51.22	
	12	12
d.	Part 4, "Markup of NUREG-4131, Revision 1, Standard Technical Specifications-Westinghouse Plants, (ISTS)"	
	3.8-9, 3.8-24, 3.8-31, 3.8-32, 3.8-36	3.8-9, 3.8-24, 3.8-31, 3.8-32, 3.8-36
	-	3.8-36a
	3.8-37	3.8-37
e.	Part 5, "Justification of Differences (JFDs) to ISTS"	
	5 & 8	5 & 8
	-	9
f.	Part 6, "Markup of ISTS Bases"	
	B 3.8-5a, B3.8-21, B 3.8-34, B 3.8-51	B 3.8-5a, B 3.8-21, B 3.8-34, B 3.8-51
	B 3.8-54	B 3.8-54
	Insert B 3.8.4-3 (no page number)	B 3.8-58a
	B 3.8-66, B 3.8-67, B 3.8-76	B 3.8-66, B 3.8-67, B 3.8-76
	-	B 3.8-76a
	B 3.8-77, B 3.8-77a	B 3.8-77, B 3.8-77a
	Insert B 3.8.9-1 (no page number)	B 3.8-79a
g.	Part 7, "Justification for Differences (JFDs) to ISTS Bases"	
	5	5
h.	Part 8, "Proposed HBRSEP, Unit No. 2 ITS"	
	3.8-6, 3.8-19, 3.8-25, 3.8-26, 3.8-30, 3.8-31	3.8-6, 3.8-19, 3.8-25, 3.8-26, 3.8-30, 3.8-31

SUPPLEMENT 5
CONVERSION PACKAGE SECTION 3.8
PAGE INSERTION INSTRUCTIONS

Remove and insert the following pages into Enclosure 16 to Serial: RNP-RA/96-0141.

	Remove Page	Insert Page
i.	Part 9. "Proposed Bases to HBRSEP, Unit No. 2 ITS Bases"	
	B 3.8-5	B 3.8-5
	B 3.8-15 through B 3.8-25	B 3.8-15 through B 3.8-25
	B 3.8-39, B 3.8-42, B3.8-44, B 3.8-52	B 3 8-39, B 3.8-42, B 3.8-44, B 3.8-52
	B 3.8-53, B 3.8-61, B 3.8-62, B 3.8-63	B 3.8-53, B 3.8-61, B 3.8-62, B 3.8-63
	B 3.8-64, B 3.8-66	B 3.8-64, B 3.8-66
j.	Part 10. "ISTS Generic Changes"	
	NA	

ITS

(A1)

In accordance with the Diesel Fuel oil Testing program (A24)

TABLE 4.1.2 (Continued)

FREQUENCIES FOR SAMPLING TESTS

(LA6)

[SR3.8.3.2]

	Check	Frequency	Maximum Time Between Tests
~~11. U-2 Diesel Generator Fuel Oil Storage Tank~~	API or Specific Gravity, Water and Sediment, Viscosity	Monthly	45 days
~~12. U-1 I-C Turbine Fuel Oil Storage Tanks or Tank Truck~~	API or Specific Gravity, Water and Sediment, Viscosity	Prior to transfer to U-2	N/A

See ITS 5.5

Verify fuel oil properties of stored fuel oil are tested in accordance with, and maintained within the limits of the Diesel fuel oil Testing Program (A24)

Supplement 5

Specification 3.0.3

Verify battery capacity is ≥ 80% for the A battery and ≥ 90% for the B battery when (M11) (A1)

[SR 3.8.4.6] The batteries shall be subjected to a performance test once every (A21) ~~five years~~ (CTS insert 4.6.3.5A) discharge

[SR 3.8.4.5 and Note 2] (when) The batteries shall be subjected to a service test at least once per 18 months during a shutdown (A15) verify that the battery capacity is adequate to supply and maintain in OPERABLE status the actual or simulated emergency loads for the design duty cycle. Surveillance 4.6.3.5 may be performed at five year intervals in lieu of this test (required)

[SR 3.8.4.5 Note 1]

4.6.4 Pressurizer Heaters' Emergency Power Supply

The emergency power supply for the pressurizer heaters shall be demonstrated operable each refueling shutdown by transferring power from normal to the emergency power supply and energizing the heaters (See 3.4.9)

4.6.5 Battery Chargers (ISI 25.7) (Verify) (terminal) (M12)

[SR 3.8.4.1] Demonstrate the in-service battery charger is operable by monitoring the output voltage daily five days per week (and) (7 days) during normal equalizing charges (L4)

Basis

The tests specified are designed to demonstrate that the diesel generators will provide power for operation of equipment. They also assure that the emergency generator system controls and the control systems for the safety features equipment will function automatically in the event of a loss of all normal 480 V AC station service power." (A6)

The test to ensure proper operation of engineered safety features upon loss of AC power is initiated by tripping the breakers supplying normal power to the 480 volt buses and initiating a safety injection signal. This test demonstrates the proper tripping of motor feeder breakers, main supply and tie breakers on the affected bus, operation of the diesel generators, and sequential starting of essential equipment.

Add SR 3.8.4.2
SR 3.8.4.3 (M13)
SR 3.8.4.4

Add SR 3.8.4.6 Note (M16)

Amendment No. 80, 132, 174
Supplement 5

A22 CTS does not provide any allowable time for battery parameters indicating an inoperable Battery. In this situation, CTS requires declaring the associated battery inoperable. ITS RA B requires immediately declaring the associated battery inoperable. Therefore, this change is considered administrative and is consistent with ISTS.

A23 With an inoperable DC or AC instrument bus power electrical distribution subsystem, CTS does not provide any specific actions. In this condition the actions are specified by 3.0. CTS 3.0 requires the unit be placed in hot shutdown within 8 hours and cold shutdown within an additional 30 hours. ITS 3.8.9 RA B.1 and C.1 permits up to two hours to restore the DC and AC instrument bus power electrical distribution subsystem to OPERABLE status. With Required Actions and associated Completion Times not met, ITS 3.8.9 RA F.1 and F.2 requires the unit be placed in MODE 3 within 6 hours and MODE 5 within 36 hours. Neither RA E.1 or RAs F.1 and F.2 specify when the shutdown must begin. Provided the unit is in MODE 3 within 8 hours and MODE 5 within 38 hours, the ITS requirements are met. With an inoperable DC or AC instrument bus power electrical distribution subsystem, both CTS and ITS require achieving hot shutdown within eight hours and cold shutdown within 38 hours. Therefore, these changes are administrative changes and are consistent with ISTS.

A24 The technical content of CTS Table 4.1.2, items 11 and 12, relating to diesel fuel oil testing requirements is moved to Chapter 5.0 of the ITS in accordance with the format of the Westinghouse Standard Technical Specifications, NUREG-1431. Any technical changes are addressed with the content of ITS 5.5.13, Diesel Fuel Oil Testing Program. A Surveillance Requirement is added (ITS SR 3.8.3.2) to clarify that the tests of the Diesel Fuel Oil Testing Program must also be completed and passed for determining OPERABILITY of the DGs. Since this is a presentation preference that maintains current requirements, this change is considered administrative.

TECHNICAL CHANGES - MORE RESTRICTIVE

M1 CTS 3.7.1 and 3.7.2 requires AC power sources to be OPERABLE with the reactor critical and during power operation. These CTS conditions encompass ITS MODES 1 and 2. ITS 3.8.1 requires OPERABILITY of AC power sources in MODES 1, 2, 3 and 4. The AC sources are required to be OPERABLE in MODES 1, 2, 3, and 4 to ensure that:

a. Acceptable fuel design limits and reactor coolant pressure boundary limits are not exceeded as a result of AOOs or abnormal transients; and

b. Adequate core cooling is provided and containment OPERABILITY and other vital functions are maintained in the event of a postulated DBA.

CTS 3.7.1.d requires DG fuel oil requirements to be met when the reactor is critical. This CTS applicability is comparable to ITS MODES 1 and 2. ITS 3.8.3 requires applicability of the DG fuel oil requirements in MODES 1, 2, 3 and 4. The AC sources (LCO 3.8.1 and LCO 3.8.2) are required to ensure the availability of the required power to shut down the reactor and maintain it in a safe shutdown condition after an AOO or a postulated DBA. Since stored diesel fuel oil supports LCO 3.8.1 and

constitute a battery discharge provided the battery terminal voltage and float current return to pre-transient values. This inspection is also consistent with IEEE-450 (Ref. 3), which recommends special inspections following a severe discharge or overcharge, to ensure that no significant degradation of the battery occurs as a consequence of such discharge or overcharge. SR 3.8.6.3 requires verification that the average temperature of representative cells is ≥ 67°F is consistent with a recommendation of IEEE-450 (Ref. 3), that states that the temperature of electrolytes in representative cells should be determined on a quarterly basis. This SR ensures that the operating temperatures remain within an acceptable operating range. These limits are based on manufacturer recommendations and unit specific calculations regarding available battery ampacity and battery temperature. Unit specific calculations are based on battery ampacity available at the specified battery temperature. The additional applicability for battery cell parameters in MODES 5 and 6 is an additional restriction upon unit operation and is consistent with ISTS. Battery cell parameters are required when the DC power source is required to be OPERABLE.

M16 CTS requirements comparable to ITS Specification 3.8.7 do not exist. The initial conditions of Design Basis Accident (DBA) and transient analyses in the UFSAR assume Engineered Safety Feature systems are OPERABLE. The AC Instrument Bus Sources are designed to provide the required capacity, capability, redundancy, and reliability to ensure the availability of necessary power to portions of the RPS and ESFAS instrumentation and controls so that the fuel, Reactor Coolant System, and containment design limits are not exceeded. The OPERABILITY of the AC Instrument Bus Sources is consistent with the initial assumptions of the accident analyses and is based on meeting the design basis of the unit. This includes maintaining required AC instrument buses OPERABLE during accident conditions in the event of:

a. An assumed loss of all offsite AC electrical power or all onsite AC electrical power; or

b. An assumed loss of offsite power and a worst case single active failure.

Therefore this is a more restrictive requirement upon unit operation and is consistent with ISTS.

M17 CTS requirements comparable to ITS Specification 3.8.8 do not exist. The initial conditions of Design Basis Accident (DBA) and transient analyses in the UFSAR assume Engineered Safety Feature systems are OPERABLE. The AC Instrument Bus Sources are designed to provide the required capacity, capability, redundancy, and reliability to ensure the

LA5 CTS 4.6.1.4 contains details regarding DG load limits during DG operation. The details regarding the load limits are not retained in the ITS, but are relocated to the ITS Bases. Changes to the Bases are controlled in accordance with ITS 5.5.14, Technical Specification Bases Control Program.

CTS 4.6.1.5 contains details regarding DG testing method. The details regarding the test method are not retained in the ITS, but are relocated to appropriate plant controlled documents.

The details associated with the involved Specifications are not required to be in the ITS to provide adequate protection of the public health and safety, since the ITS still retains the requirement for OPERABILITY of the AC power sources. This approach provides an effective level of regulatory control and provides for a more appropriate change control process. The level of safety of facility operation is unaffected by the change because there is no change in the overall operational requirements. Furthermore, NRC and utility resources associated with processing license amendments to these requirements will be reduced. Therefore, relocation of these details is acceptable.

LA6 CTS Table 4.1.2, items 11 and 12 contain details associated with testing the DG fuel oil (i.e., which diesel fuel oil storage tanks the testing requirements apply to). The details regarding the DG fuel oil testing are not retained in the ITS, but are relocated to the ITS Bases.

The details associated with the involved Specifications are not required to be in the ITS to provide adequate protection of the public health and safety, since the ITS still retains the requirements for OPERABILITY of the AC power sources. This approach provides an effective level of regulatory control and provides for a more appropriate change control process. The level of safety of facility operation is unaffected by the change because there is no change in the overall operational requirements. Furthermore, NRC and utility resources associated with processing license amendments to these requirements will be reduced. Therefore, relocation of these details is acceptable.

LA7 CTS 4.6.3.2 and 4.6.3.4 specify details regarding battery testing methodology and data recording. These details regarding the battery testing are not retained in the ITS, but are relocated to appropriate plant controlled documents.

The details associated with the involved Specifications are not required to be in the ITS to provide adequate protection of the public health and safety, since the ITS still retains the requirement for OPERABILITY of the batteries. This approach provides an effective level of regulatory control and provides for a more appropriate change control process. The

initiators of any accident previously evaluated. Additionally, appropriate breaker coordination is still maintained utilizing the magnetic trip element for one of the two breakers associated with each ABT. The surveillance requirements confirm proper operation of the required breakers. As a result the probability of an accident is not affected by the elimination of the test requirement associated with the thermal trip elements or the second breaker of each pair associated with an ABT. The consequences of an accident are independent of the breaker trip element coordination testing scope. Accordingly, the consequences of an accident are not increased. Therefore, this change does not involve an increase in the probability or consequences of an accident previously evaluated.

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

   The proposed change does not involve any changes in parameters governing normal plant operation. Appropriate breaker coordination is still verified utilization the magnetic trip device for only one of the two breakers associated with an ABT. The surveillance requirements confirm proper operation of the breakers. The change provides reasonable assurance the breakers will perform their required function. Therefore, the possibility of a new or different kind of accident from any accident previously evaluated is not created.

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

   The margin of safety has not been reduced since appropriate breaker coordination is still verified. The new surveillance requirement provides appropriate testing of the breakers. Therefore, the margin of safety is not reduced.

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE	FREQUENCY
SR 3.8.1.~~9~~ 8	
----------NOTES---------- 1. This Surveillance shall not be performed in MODE 1 or 2. ~~However, credit may be taken for unplanned events that satisfy this SR.~~ TSTF-8, R2 [M 6] 2. If performed with the DG synchronized with offsite power, it shall be performed at a power factor ≤ [0.9]. ----------	
Verify each DG rejects a load greater than or equal to its associated single largest post-accident load, ~~and~~	~~[~~18 months~~]~~
~~a. Following load rejection, the frequency is ≤ [63] Hz;~~ ~~b. Within [3] seconds following load rejection, the voltage is ≥ [3740] V and ≤ [4580] V; and~~ ~~c. Within [3] seconds following load rejection, the frequency is ≥ [58.8] Hz and ≤ [61.2] Hz.~~ does not trip on overspeed. 45	
~~SR 3.8.1.10~~ ~~----------NOTE---------- This Surveillance shall not be performed in MODE 1 or 2. However, credit may be taken for unplanned events that satisfy this SR. ----------~~	
~~Verify each DG operating at a power factor ≤ [0.9] does not trip and voltage is maintained ≤ [5000] V during and following a load rejection of ≥ [4500] kW and ≤ [5000] kW.~~	~~[~~18 months~~]~~ 13

(continued)

CTS

3.8 ELECTRICAL POWER SYSTEMS

3.8.4 DC Sources – Operating

[3.7.1c] LCO 3.8.4 The Train A and Train B DC electrical power subsystems shall be OPERABLE.

[3.7.1] APPLICABILITY: MODES 1, 2, 3, and 4.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
[3.7.2.f] A. One DC electrical power subsystem inoperable.	A.1 Restore DC electrical power subsystem to OPERABLE status.	2 hours
[3.7.2.f] B. Required Action and Associated Completion Time not met.	B.1 Be in MODE 3. AND [3.7.2.f] B.2 Be in MODE 5.	6 hours 36 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
[4.6.5] SR 3.8.4.1 Verify battery terminal voltage is ≥ ~~129~~ 125.7 V on float charge.	7 days

(continued)

CTS

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
[A22] B. Required Action and associated Completion Time of Condition A not met. OR One or more batteries with average electrolyte temperature of the representative cells < ~~60°F~~ 67°F. OR One or more batteries with one or more battery cell parameters not within Category C values.	B.1 Declare associated battery inoperable.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
4.6.3.1 SR 3.8.6.1	Verify battery cell parameters meet Table 3.8.6-1 Category A limits.	7 days

(continued)

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CTS

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
[M15] SR 3.8.6.2	Verify battery cell parameters meet Table 3.8.6-1 Category B limits.	92 days AND Once within 24 hours after a battery discharge < ~~[~~110~~]~~ V AND Once within 24 hours after a battery overcharge > ~~[~~150~~]~~ V
[M15] SR 3.8.6.3	Verify average electrolyte temperature of representative cells is ≥ ~~[60]°F~~ 67°F.	92 days

CTS

3.8 ELECTRICAL POWER SYSTEMS

AC Instrument Bus Sources (44)

3.8.8 ~~Inverters~~ – Shutdown

Insert 3.8.8-1 (44) (2)

[M17] LCO 3.8.8 ~~Inverters shall be OPERABLE to support~~ the onsite ~~Class 1E~~ AC ~~vital~~ bus electrical power distribution subsystem(s) required by LCO 3.8.10, "Distribution Systems – Shutdown." (24)

instrument Insert 3.8.8-2 (44)

[M17] APPLICABILITY: MODES 5 and 6,
During movement of irradiated fuel assemblies.

NOTE
LCO 3.0.3 is Not Applicable (31)

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
[M17] A. One or more ~~required~~ ~~inverters~~ inoperable. (AC Instrument Bus Sources) (44)	A.1 Declare affected required feature(s) inoperable.	Immediately
	OR	
[M17]	A.2.1 Suspend CORE ALTERATIONS.	Immediately
	AND	
[M17]	A.2.2 Suspend movement of irradiated fuel assemblies.	Immediately
	AND	
[M17]	A.2.3 Initiate action to suspend operations involving positive reactivity additions.	Immediately
	AND	
		(continued)

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Insert 3.8.8-1

The following shall be OPERABLE:

a. One inverter or constant voltage transformer (CVT) capable of supplying one train of

Insert 3.8.8-2

; and

b. One source of AC instrument bus power, other than that required by LCO 3.8.8.a, capable of supplying the remaining onsite AC instrument bus

CTS

44

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
[M17] A. (continued)	A.2.4 Initiate action to restore required ~~inverters~~ AC Instrument Bus Sources to OPERABLE status.	Immediately

44

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
[M17] SR 3.8.8.1 Verify correct ~~inverter~~ voltage, ~~&~~ frequency, ~~&~~ and alignments to required AC ~~vital~~ instrument buses.	7 days

[M17]

NOTE

Actual voltage and frequency measurement is required for AC instrument buses supplied from CVTs.

44

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eight 120 V instrument buses. The remaining four 120 V instrument buses are supplied from constant voltage transformers (CVT). The ISTS Inverters specification has been appropriately modified to incorporate the CVTs as comparable AC Instrument Bus supplies. An appropriate Condition with an associated Required Action is provided to address the condition with an inoperable CVT. An appropriate Surveillance Requirement, SR 3.8.7.2, is provided for the CVTs. The constant voltage transformers are not provided with installed instrumentation which provide output voltage. The SRs for CVTs include verification of voltage availability without requiring actual measurement of voltage.

24 The term instrument bus is used in lieu of vital bus.

25 For clarity and consistency with other comparable Conditions as well as the associated bases, the term subsystem was added to the Condition associated with the AC instrument bus inoperable.

26 Consistent with the CLB Conditions and associated Required Actions are added for circuit protection features associated with specified loads. Appropriate Surveillance requirements associated with these circuit protection features are added. The HBRSEP design includes specified components which are powered from both AC power trains by utilization of an Automatic Bus Transfer (ABT). Circuit protection for these loads is necessary to ensure protection from common cause failure of both AC power trains.

27 The HBRSEP design does not provide a generator differential current trip for the DG output breaker. There is no CTS requirement comparable to the suggested requirement. Since the generator differential current protective feature does not exist, it is not appropriate to be included in the SR.

28 ISTS SR 3.8.1.16 requires verification that the DGs can be manually synchronized and then automatically transfer the load from the DG to the offsite source and then return to the "ready to load" state. The STS bases indicates the synchronization is manual but the load transfer is described as an automatic load transfer. The HBRSEP design does not provide such an automatic transfer from the onsite AC source to the offsite AC circuit. This transfer must be accomplished manually. Since this capability is not consistent with plant design or the CLB, the SR is not adopted.

29 For consistency with Condition B and SR 3.8.6.3, LCO 3.8.6 is modified to explicitly include requirements for electrolyte temperature.

30 ITS ST 3.8.3.6 is a preventative type of SR. Sediment in the tank, or failure to perform this SR, does not necessarily result in an inoperable

into consideration the likelihood of a change in component or system status.

41 The HBRSEP design does not include the DG test mode override feature.

42 There is no CLB for the duration of the battery charger surveillance. Additionally, the NUREG does not establish a basis for the bracketed 8 hour duration. A four hour test duration is adopted. Four hours is considered sufficient to permit electronic components within the battery charger to stabilize at operating temperature. This value is also the consensus value recommended by the IEEE Standards Coordination Committee (SCC) 29 for Station batteries during a discussion within the Nuclear Task Force at the recent spring meeting.

43 There is no current licensing basis for the NUREG bracketed value for battery charger current specified for ITS SR 3.8.4.6. The HBRSEP design provides a charger rated at 300 Amps. This value for battery charging current is sufficient to meet design and safety analysis assumptions regarding battery charger current.

44 The HBRSEP design utilizes inverters to provide AC power for four of the eight 120 V instrument buses. The remaining four 120 V instrument buses are supplied from constant voltage transformers (CVTs). Therefore, NUREG-1431 LCO 3.8.8, Inverters - Shutdown, is appropriately modified to incorporate the constant voltage transformers (CVTs) as comparable AC instrument bus supplies. LCO 3.8.8, Inverters - Shutdown, is also revised to reflect more specific requirements for each required AC instrument bus electrical power distribution subsystem similar to LCO 3.8.2, AC Sources - Shutdown. As currently written, NUREG-1431 LCO 3.8.8 implies that a DC battery backed inverter is required to be OPERABLE for both trains of the AC instrument bus electrical power distribution subsystem (when two trains are required by LCO 3.8.10). The requirements for the second subsystem (train) are proposed to be relaxed to require a source of power be capable of supplying power to the associated AC instrument bus electrical power distribution subsystem. The Bases of LCO 3.8.8 are revised to state that when the redundant train of the AC instrument bus electrical power distribution subsystem is required by LCO 3.8.10, the power source for this AC instrument bus may consist of : 1) the inverter powered by its associated battery; 2) a CVT; or 3) an offsite circuit providing power through a motor control center. This change is necessary to avoid unnecessarily requiring equipment to be declared inoperable when other sources of power are available to the redundant required AC instrument buses. This change potentially impacts the power supply requirements for AC instrument buses which support the OPERABILITY of the following Technical Specification equipment required during shutdown:

Source Range Neutron Flux monitors for Nuclear Instrumentation (ITS 3.9.2);

Pressurizer PORVs for Low Temperature Overpressure Protection (ITS 3.4.12);

Containment radiation monitors for Containment Ventilation Isolation Instrumentation (ITS 3.3.6);

Control room radiation monitor for Control Room Emergency Filtration System (CREFS) Instrumentation (ITS 3.3.7); and

Automatic Actuation Logic and Actuation Relays for CREFS Instrumentation (ITS 3.3.7)

The proposed change to the requirements for AC instrument bus sources will continue to assure that sufficient power is available to support the response to events postulated during shutdown conditions in the event of a loss of offsite power or a single failure. It should also be noted that this change is consistent with the initial philosophy of the ITS NUREGs.

45 ISTS SR 3.8.1.9 is adopted in ITS as SR 3.8.1.8, with the acceptance criteria changed to state that the DG does not trip on overspeed. A test similar to this SR has only been performed once in the past and acceptance criteria were not established for that test other than the DG would not trip on overspeed. The test showed that the DG could reject a large load (i.e, a Containment Spray pump and a Containment Cooling Unit) without experiencing an overspeed trip. Since the DG does not trip, the DG remains OPERABLE and the emergency bus continues to perform its required function.

Insert B3.8-5a

The Completion Time for inoperability of the offsite source is 12 hours. The rationale for the 12 hours is that Regulatory Guide 1.93 (Ref. 9) allows a Completion Time of 24 hours for two required offsite circuits inoperable when two offsite sources are incorporated into the design, based upon the assumption that two complete safety trains are OPERABLE. When no offsite sources are OPERABLE, this assumption is not the case, and a shorter Completion Time of 12 hours is appropriate.

BASES

SURVEILLANCE REQUIREMENTS

SR 3.8.1.~~9~~ 8 (continued)

overspeed, which, if excessive, might result in a trip of the engine. This Surveillance demonstrates the DG load response characteristics and capability to reject the largest single load without exceeding ~~predetermined voltage and frequency and while maintaining a specified margin to the overspeed trip.~~ For this unit, the single load for each DG ~~and its nameplate rating is as follows:~~ is a Safety Injection pump rated at 380 Brake Horsepower. This Surveillance may be accomplished by: (10)

a. Tripping the DG output breaker with the DG carrying greater than or equal to its associated single largest post-accident load while paralleled to offsite power, or while solely supplying the bus; or

b. Tripping its associated single largest post-accident load with the DG solely supplying the bus.

~~As required by IEEE-308 (Ref. 12), the load rejection test is acceptable if the increase in diesel speed does not exceed 75% of the difference between synchronous speed and the overspeed trip setpoint, or 15% above synchronous speed, whichever is lower.~~

~~The time, voltage, and frequency tolerances specified in this SR are derived from Regulatory Guide 1.9 (Ref. 3) recommendations for response during load sequence intervals. The 3 seconds specified is equal to 60% of a typical 5 second load sequence interval associated with sequencing of the largest load. The voltage and frequency specified are consistent with the design range of the equipment powered by the DG. SR 3.8.1.9.a corresponds to the maximum frequency excursion, while SR 3.8.1.9.b and SR 3.8.1.9.c are steady state voltage and frequency values to which the system must recover following load rejection.~~ The ~~18 month~~ Frequency is consistent with the recommendation of Regulatory Guide 1.108 (Ref. ~~9~~ 8). (10)

This SR is modified by two Notes. The reason for Note 1 is that during operation with the reactor critical, performance of this SR could cause perturbations to the electrical distribution systems that could challenge continued steady state operation and, as a result, unit safety systems. ~~Credit may be taken for unplanned events that satisfy this SR.~~ In order to ensure that the DG is tested under load

TSTF-8, R2

(continued)

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BASES

REFERENCES (continued)

8. Regulatory Guide 1.108, Rev. 1, August 1977.
10. Regulatory Guide 1.137, Rev. 1, ~~dated~~ October 1979
11. ~~ASME, Boiler and Pressure Vessel Code, Section XI~~
12. ~~IEEE Standard 308-1978.~~ Regulatory Guide 1.9, Rev. 3 July 1993.
9. Regulatory Guide 1.93, Rev. 0, December 1974

Supplement 5

BASES

BACKGROUND (continued)

The DC power distribution system is described in more detail in Bases for LCO 3.8.9, "Distribution System – Operating," and LCO 3.8.10, "Distribution Systems – Shutdown."

Each battery has adequate storage capacity to carry the required load continuously for at least 2 hours and to perform three complete cycles of intermittent loads discussed in the FSAR, Chapter [8] (Ref. 4).

Each 125 VDC battery is separately housed in a ventilated room apart from its charger and distribution centers. Each subsystem is located in an area separated physically and electrically from the other subsystem to ensure that a single failure in one subsystem does not cause a failure in a redundant subsystem. There is no sharing between redundant Class 1E subsystems, such as batteries, battery chargers, or distribution panels.

The batteries for Train A and Train B DC electrical power subsystems are sized to produce required capacity at 80% of nameplate rating, corresponding to warranted capacity at end of life cycles and the 100% design demand. Battery size is based on 125% of required capacity and, after selection of an available commercial battery, results in a battery capacity in excess of 150% of required capacity. The voltage limit is 2.13 V per cell, which corresponds to a total minimum voltage output of 128 V per battery discussed in the FSAR, Chapter [8] (Ref. 4). The criteria for sizing large lead storage batteries are defined in IEEE-485 (Ref. 5).

Each Train A and Train B DC electrical power subsystem has ample power output capacity for the steady state operation of connected loads required during normal operation, while at the same time maintaining its battery bank fully charged. Each battery charger also has sufficient capacity to restore the battery from the design minimum charge to its fully charged state within 24 hours while supplying normal steady state loads discussed in the FSAR, Chapter [8] (Ref. 4).

APPLICABLE SAFETY ANALYSES

The initial conditions of Design Basis Accident (DBA) and transient analyses in the FSAR, Chapter [6] (Ref. 6), and in the FSAR, Chapter [15] (Ref. 7), assume that Engineered Safety Feature (ESF) systems are OPERABLE. The DC

(continued)

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BASES

ACTIONS

B.1 and B.2 (continued)

within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required unit conditions from full power conditions in an orderly manner and without challenging plant systems. ~~The Completion Time to bring the unit to MODE 5 is consistent with the time required in Regulatory Guide 1.93 (Ref. 8).~~

SURVEILLANCE REQUIREMENTS

SR 3.8.4.1

Verifying battery terminal voltage while on float charge for the batteries helps to ensure the effectiveness of the charging system and the ability of the batteries to perform their intended function. Float charge is the condition in which the charger is supplying the continuous charge required to overcome the internal losses of a battery (or battery cell) and maintain the battery (or a battery cell) in a fully charged state. The voltage requirements are based on the nominal design voltage of the battery and are consistent with the initial voltages assumed in the battery sizing calculations and permit a single battery cell to be jumpered out. The 7 day Frequency is consistent with manufacturer recommendations and IEEE-450 (Ref. ~~8~~ 5).

~~SR 3.8.4.2~~

~~Visual inspection to detect corrosion of the battery cells and connections, or measurement of the resistance of each intercell, interrack, intertier, and terminal connection, provides an indication of physical damage or abnormal deterioration that could potentially degrade battery performance.~~

~~The limits established for this SR must be no more than 20% above the resistance as measured during installation or not above the ceiling value established by the manufacturer.~~

~~The Surveillance Frequency for these inspections, which can detect conditions that can cause power losses due to resistance heating, is 92 days. This Frequency is considered acceptable based on operating experience related to detecting corrosion trends.~~

(continued)

Insert B3.8.4-3

An acceptance criterion of 80% of rated capacity is applicable to the "A" battery only. An acceptance criterion of 91% is applicable to the "B" battery since the battery's capacity is not as great.

Insert B3.8.4-4

with an extra allowance for a 18 month test frequency for batteries which have shown degradation or have reached 85% for battery "A" and 95% for battery "B" of expected life.

BASES

ACTIONS

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

Continued operation is only permitted for 31 days before battery cell parameters must be restored to within Category A and B limits. With the consideration that, while battery capacity is degraded, sufficient capacity exists to perform the intended function and to allow time to fully restore the battery cell parameters to normal limits, this time is acceptable prior to declaring the battery inoperable.

B.1

With one or more batteries with one or more battery cell parameters outside the Category C limit for any connected cell, sufficient capacity to supply the maximum expected load requirement is not assured and the corresponding DC electrical power subsystem must be declared inoperable. Additionally, other potentially extreme conditions, such as not completing the Required Actions of Condition A within the required Completion Time or average electrolyte temperature of representative cells falling below 60°F, are also cause for immediately declaring the associated DC electrical power subsystem inoperable.

67°F

SURVEILLANCE REQUIREMENTS

SR 3.8.6.1

This SR verifies that Category A battery cell parameters are consistent with IEEE-450 (Ref. 3), which recommends regular battery inspections (at least one per month) including voltage, specific gravity, and electrolyte temperature of pilot cells.

Measured to the nearest 0.01 Volts

In addition, if water is added to any pilot cell, the amount must be recorded.

SR 3.8.6.2

The quarterly inspection of specific gravity and voltage is consistent with IEEE-450 (Ref. 3). In addition, within 24 hours of a battery discharge < ~~X~~110~~X~~ V or a battery overcharge > ~~X~~150~~X~~ V, the battery must be demonstrated to meet Category B limits. Transients ~~such as motor starting transients~~ which may momentarily cause battery voltage to drop to ≤ ~~X~~110~~X~~ V, do not constitute a battery discharge

Data obtained must be compared to the data from the previous SR to detect signs of abuse or deterioration.

(continued)

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If water is added to any battery cell the amount must be recorded. Data obtained must be compared to the data from the previous SR to detect signs of abuse or deterioration.

BASES

SURVEILLANCE REQUIREMENTS

SR 3.8.6.2 (continued)

provided the battery terminal voltage and float current return to pre-transient values. This inspection is also consistent with IEEE-450 (Ref. 3), which recommends special inspections following a severe discharge or overcharge, to ensure that no significant degradation of the battery occurs as a consequence of such discharge or overcharge.

Data obtained must be compared to the data from the previous SR to detect signs of abuse or deterioration!

SR 3.8.6.3

This Surveillance verification that the average temperature of representative cells is ≥60°F 67°F is consistent with a recommendation of IEEE-450 (Ref. 3), that states that the temperature of electrolytes in representative cells should be determined on a quarterly basis.

Lower than normal temperatures act to inhibit or reduce battery capacity. This SR ensures that the operating temperatures remain within an acceptable operating range. This limit is based on manufacturer recommendations.

Table 3.8.6-1

This table delineates the limits on electrolyte level, float voltage, and specific gravity for three different categories. The meaning of each category is discussed below.

Category A defines the normal parameter limit for each designated pilot cell in each battery. The cells selected as pilot cells are those whose temperature, voltage, and electrolyte specific gravity approximate the state of charge of the entire battery.

The Category A limits specified for electrolyte level are based on manufacturer recommendations and are consistent with the guidance in IEEE-450 (Ref. 3), with the extra ¼ inch allowance above the high water level indication for operating margin to account for temperatures and charge effects. In addition to this allowance, footnote a to Table 3.8.6-1 permits the electrolyte level to be above the specified maximum level during equalizing charge, provided it is not overflowing. These limits ensure that the plates

(continued)

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BASES (continued)

AC Instrument Bus Sources

LCO

The ~~inverters~~ ensure the availability of electrical power for the instrumentation for systems required to shut down the reactor and maintain it in a safe condition after an anticipated operational occurrence or a postulated DBA. ~~The battery powered inverters provide uninterruptible supply of AC electrical power to the AC vital buses even if the 4.16 kV safety buses are de-energized. OPERABILITY of the inverters requires that the AC vital bus be powered by the inverter.~~ This ensures the availability of sufficient ~~inverter~~ power sources to operate the unit in a safe manner and to mitigate the consequences of postulated events during shutdown (e.g., fuel handling accidents).

Insert B 3.8.8-1A

AC Instrument Bus Sources

APPLICABILITY

AC Instrument Bus Sources

The ~~inverters~~ required to be OPERABLE in MODES 5 and 6 and during movement of irradiated fuel assemblies provide assurance that:

a. Systems to provide adequate coolant inventory makeup are available for the irradiated fuel in the core;

b. Systems needed to mitigate a fuel handling accident are available;

c. Systems necessary to mitigate the effects of events that can lead to core damage during shutdown are available; and

d. Instrumentation and control capability is available for monitoring and maintaining the unit in a cold shutdown condition or refueling condition.

AC Instrument Bus Sources

~~Inverter~~ requirements for MODES 1, 2, 3, and 4 are covered in LCO 3.8.7.

ACTIONS

A.1, A.2.1, A.2.2, A.2.3, and A.2.4

Insert B3.8.8-1

With one or more required AC instrument bus inoperable when

If two trains are required by LCO 3.8.10, "Distribution Systems - Shutdown," the remaining OPERABLE ~~inverters~~ may be capable of supporting sufficient required features to allow continuation of CORE ALTERATIONS, fuel movement, and operations with a potential for positive reactivity additions. By the allowance of the option to declare

AC Instrument Bus Sources

(continued)

Supplement 5

Insert B3.8.8-1A

At least one AC instrument bus train energized by one battery powered inverter or a constant voltage transformer (CVT) ensures that the preferred source of AC instrument bus electrical power is available to at least one AC instrument bus. OPERABILITY of the inverters and CVTs requires that the AC instrument bus be powered from the associated inverter or CVT, as applicable. When the redundant train of the AC instrument bus electrical power distribution subsystem is required by LCO 3.8.10, the power source for this AC instrument bus may consist of : 1) the inverter powered by its associated battery; 2) the CVT; or 3) an offsite circuit providing power through a motor control center.

Insert B3.8.8-1

LCO 3.0.3 is not applicable while in MODE 5 and 6. However, since irradiated fuel assembly movement can occur in MODE 1, 2, 3, or 4, the ACTIONS have been modified by a Note stating that LCO 3.0.3 is not applicable. If moving fuel assemblies while in MODE 1, 2, 3, or 4, the fuel movement is independent of reactor operations. Therefore, in either case, inability to suspend movement of irradiated fuel assemblies would not be sufficient reason to require a reactor shutdown.

(1)

BASES

AC Instrument Bus Sources (10)

ACTIONS

A.1, A.2.1, A.2.2, A.2.3, and A.2.4 (continued)

required features inoperable with the associated ~~inverter(s)~~ inoperable, appropriate restrictions will be implemented in accordance with the affected required features LCOs' Required Actions. In many instances, this option may involve undesired administrative efforts. Therefore, the allowance for sufficiently conservative actions is made (i.e., to suspend CORE ALTERATIONS, movement of irradiated fuel assemblies, and operations involving positive reactivity additions). The Required Action to suspend positive reactivity additions does not preclude actions to maintain or increase reactor vessel inventory, provided the required SDM is maintained.

Suspension of these activities shall not preclude completion of actions to establish a safe conservative condition. These actions minimize the probability of the occurrence of postulated events. It is further required to immediately initiate action to restore the required ~~inverters~~ and to continue this action until restoration is accomplished in order to provide the necessary ~~inverter~~ power to the ~~unit~~ safety systems. (10) AC Instrument Bus Sources

The Completion Time of immediately is consistent with the required times for actions requiring prompt attention. The restoration of the required ~~inverters~~ should be completed as quickly as possible in order to minimize the time the unit safety systems may be without power or powered from a ~~constant voltage source transformer~~. Non-preferred source

SURVEILLANCE REQUIREMENTS

SR 3.8.8.1 instrument (27)

(10) required

This Surveillance verifies that the inverters are functioning properly with all required circuit breakers closed and AC ~~vital~~ buses energized from the inverter. The verification of proper voltage and frequency output ensures that the required power is readily available for the instrumentation connected to the AC ~~vital~~ buses. The 7 day Frequency takes into account the redundant capability of the ~~inverters and~~ other indications available in the control room that alert the operator to inverter malfunctions. (for the inverters) (10) instrument associated AC Instrument Bus Sources, (10)

Insert B3.8.8-2

(continued)

and that required circuit breakers are closed and required instrument buses are energized from the CVTs or other sources, as allowed by LCO 3.8.8.b.

Supplement 5

Insert B3.8.8-2

and administrative requirements governing alignment of electrical equipment.

This SR is modified by a Note which states that voltage and frequency measurement is not required for the AC instrument buses supplied from CVTs. For these buses, observing status lights, instrument displays, etc. is sufficient to confirm that the required power is readily available to the AC instrument buses supplied from the CVTs.

Insert B3.8.9-1

The 480 V ESF bus E2 is normally powered from the 115 kV switchyard through the startup transformer, 4.160 kV bus 3 and station service transformer 2G. The 480 V ESF bus E1 is normally powered from the main generator through the unit main (auxiliary) transformer, 4.160 kV buses 1 and 2 and station service transformer 2F. A main generator lockout causes 4.160 kV buses 1 and 2 to be automatically transferred to the startup transformer which results in 480 V ESF bus E1 being supplied from the startup transformer.

Should a failure of the startup transformer occur, a spare startup transformer located onsite can be jumpered into service. During the time that the startup transformer is out of service, the unit auxiliary transformer can supply power to the onsite distribution system by back-feeding the main transformer from the 230 kV switchyard. Prior to back-feeding the main transformer from the 230 kV switchyard, the generator must be disconnected from the main transformer by removing the connecting straps. The main transformer backfeeding will only be done during cold shutdown unless nuclear safety consideration require it to be done during hot shutdown when no other offsite power sources are available.

Insert B3.8.9-2

The Auxiliary Feedwater (AFW) Header Discharge Valve to S/G "A", V2-16A and the Service Water System (SWS) Turbine Building Supply Valve (emergency supply), V6-I6C are powered from both Train A and Train B of the AC electrical bus distribution system by utilization of Automatic Bus Transfer (ABT) devices and molded case circuit breakers connected to each AC distribution train. Magnetic trip elements for these circuit breakers (two breakers per valve) provide circuit protection to prevent common mode failure (i.e., transfer of a fault from one electrical bus to the redundant bus) of both trains of the AC distribution systems.

Insert B3.8.9-3

The 120 VAC instrument buses are arranged in two load groups per train. One load group is made up of two instrument buses normally powered from an inverter. The remaining load group is made up of two instrument buses powered from a constant voltage transformer powered from the associated AC emergency bus. The alternate power supply for the inverter supplied instrument buses and the constant voltage transformer supplied instrument buses is an AC source powered from the station AC power distribution system, and its use is governed by LCO 3.8.7, "AC Instrument Bus Sources - Operating."

systems. Therefore to ensure accuracy as well as consistency with other ITS sections, the term "single active failure" is used.

53 Provided clarification that in MODES 5 and 6 the unit auxiliary transformer backfed through the unit main transformer can be used as part of the qualified offsite circuit. This is CLB for HBRSEP Unit No. 2. The use of back charged unit auxiliary transformer when unit is shutdown is described in UFSAR Sections 8.2 and 8.3. This capability was reviewed and approved by NRC by issuance of Amendment No. 88 dated 1/2/85.

54 The references are modified based upon either plant specific utilization in the associated Bases or specific applicability to the facility.

55 The minimum battery voltage output of 2.13 volts per cell and total output of 128 volts is not discussed in the UFSAR.

56 The bases to SR 3.8.4.1 are revised to reflect the voltage associated with a single battery cell jumpered out. This change is consistent with the current licensing basis which does not specify the battery float voltage requirement.

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.8.1.6	Verify the fuel oil transfer system operates to automatically transfer fuel oil from storage tank to the day tank.	31 days
SR 3.8.1.7	----------NOTES---------- All DG starts may be preceded by an engine prelube period. ---------- Verify each DG starts from standby condition and achieves in ≤ 10 seconds, voltage ≥ 467 V and frequency ≥ 58.8 Hz, and after steady state conditions are reached, maintains voltage ≥ 467 V and ≤ 493 V and frequency ≥ 58.8 Hz and ≤ 61.2 Hz.	184 days
SR 3.8.1.8	----------NOTES---------- 1. This Surveillance shall not be performed in MODE 1 or 2. 2. If performed with the DG synchronized with offsite power, it shall be performed at a power factor ≤ 0.9. ---------- Verify each DG rejects a load greater than or equal to its associated single largest post-accident load and does not trip on overspeed.	18 months

(continued)

3.8 ELECTRICAL POWER SYSTEMS

3.8.4 DC Sources – Operating

LCO 3.8.4 The Train A and Train B DC electrical power subsystems shall be OPERABLE.

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

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One DC electrical power subsystem inoperable.	A.1 Restore DC electrical power subsystem to OPERABLE status.	2 hours
B. Required Action and Associated Completion Time not met.	B.1 Be in MODE 3. AND B.2 Be in MODE 5.	6 hours 36 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
SR 3.8.4.1 Verify battery terminal voltage is ≥ 125.7 V on float charge.	7 days

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
B. Required Action and associated Completion Time of Condition A not met. OR One or more batteries with average electrolyte temperature of the representative cells < 67°F. OR One or more batteries with one or more battery cell parameters not within Category C values.	B.1 Declare associated battery inoperable.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.8.6.1	Verify battery cell parameters meet Table 3.8.6-1 Category A limits.	7 days

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.8.6.2	Verify battery cell parameters meet Table 3.8.6-1 Category B limits.	92 days AND Once within 24 hours after a battery discharge < 110 V AND Once within 24 hours after a battery overcharge > 150 V
SR 3.8.6.3	Verify average electrolyte temperature of representative cells is ≥ 67°F.	92 days

3.8 ELECTRICAL POWER SYSTEMS

3.8.8 AC Instrument Bus Sources – Shutdown

LCO 3.8.8 The following shall be OPERABLE:

a. One inverter or one constant voltage transformer (CVT) capable of supplying one train of the onsite AC instrument bus electrical power distribution subsystem(s) required by LCO 3.8.10, "Distribution Systems – Shutdown"; and

b. One source of AC instrument bus power, other than that required by LCO 3.8.8.a, capable of supplying the remaining onsite AC instrument bus electrical power distribution subsystem(s) when required by LCO 3.8.10.

APPLICABILITY: MODES 5 and 6, and
During movement of irradiated fuel assemblies.

ACTIONS

NOTE
LCO 3.0.3 is not applicable.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more required AC instrument bus sources inoperable.	A.1 Declare affected required feature(s) inoperable.	Immediately
	OR	
	A.2.1 Suspend CORE ALTERATIONS.	Immediately
	AND	
	A.2.2 Suspend movement of irradiated fuel assemblies.	Immediately
	AND	
		(continued)

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. (continued)	A.2.3 Initiate action to suspend operations involving positive reactivity additions.	Immediately
	AND	
	A.2.4 Initiate action to restore required AC instrument bus sources to OPERABLE status.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.8.8.1	----------NOTE---------- Actual voltage and frequency measurement is not required for AC instrument buses supplied from CVTs. ----------	
	Verify correct voltage, frequency, and alignments to required AC instrument buses.	7 days

BASES

LCO (continued)

Proper sequencing of loads, including tripping of nonessential loads, is a required function for DG OPERABILITY.

The AC sources in one train are separate and independent (to the extent possible) of the AC sources in the other train. For the DGs, separation and independence are complete.

APPLICABILITY

The AC sources are required to be OPERABLE in MODES 1, 2, 3, and 4 to ensure that:

a. Acceptable fuel design limits and reactor coolant pressure boundary limits are not exceeded as a result of AOOs or abnormal transients; and

b. Adequate core cooling is provided and containment OPERABILITY and other vital functions are maintained in the event of a postulated DBA.

The AC power requirements for MODES 5 and 6 are covered in LCO 3.8.2, "AC Sources – Shutdown and During Movement of Irradiated Fuel Assemblies."

ACTIONS

A.1

Required Action A.1, which only applies if the train cannot be powered from an offsite source, is intended to provide assurance that an event coincident with a single failure of the associated DG will not result in a complete loss of safety function of critical redundant required features. These features are powered from the redundant AC electrical power train. This includes motor driven auxiliary feedwater pumps. Single train systems, such as turbine driven auxiliary feedwater pumps, may not be included.

The Completion Time for inoperability of the offsite source is 12 hours. The rationale for the 12 hours is that Regulatory Guide 1.93 (Ref. 9) allows a Completion Time of 24 hours for two required offsite circuits inoperable when two offsite sources are incorporated into the design, based

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.8.1.5 (continued)

breakdown of the fuel oil by bacteria. Frequent checking for and removal of accumulated water minimizes fouling and provides data regarding the watertight integrity of the fuel oil system. The Surveillance Frequencies are established by Regulatory Guide 1.137 (Ref. 6). This SR is for preventative maintenance. The presence of water does not necessarily represent failure of this SR, provided the accumulated water is removed during the performance of this Surveillance.

SR 3.8.1.6

This Surveillance demonstrates that each required fuel oil transfer pump operates and transfers fuel oil from the storage tank to its associated day tank. This is required to support continuous operation of standby power sources. This Surveillance provides assurance that the fuel oil transfer pump is OPERABLE, the fuel oil piping system is intact, the fuel delivery piping is not obstructed, and the controls and control systems for automatic fuel transfer systems are OPERABLE.

The frequency of 31 days is based on the design of fuel transfer system. The pumps operate automatically in order to maintain an adequate volume of fuel oil in the day tanks during or following DG testing.

SR 3.8.1.7

See SR 3.8.1.2.

SR 3.8.1.8

Each DG is provided with an engine overspeed trip to prevent damage to the engine. Recovery from the transient caused by the loss of a large load could cause diesel engine overspeed, which, if excessive, might result in a trip of the engine. This Surveillance demonstrates the DG load response characteristics and capability to reject the largest single load without exceeding the overspeed trip.

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.8.1.8 (continued)

For this unit, the single load for each DG is a safety injection pump rated at 380 Brake Horsepower. This Surveillance may be accomplished by:

a. Tripping the DG output breaker with the DG carrying greater than or equal to its associated single largest post-accident load while paralleled to offsite power, or while solely supplying the bus; or

b. Tripping its associated single largest post-accident load with the DG solely supplying the bus.

The 18 month Frequency is consistent with the recommendation of Regulatory Guide 1.108 (Ref. 8).

This SR is modified by two Notes. The reason for Note 1 is that during operation with the reactor critical, performance of this SR could cause perturbations to the electrical distribution systems that could challenge continued steady state operation and, as a result, unit safety systems. In order to ensure that the DG is tested under load conditions that are as close to design basis conditions as possible, Note 2 requires that, if synchronized to offsite power, testing must be performed using a power factor ≤ 0.9. This power factor is chosen to be representative of the actual design basis inductive loading that the DG would experience.

SR 3.8.1.9

This Surveillance demonstrates the as designed operation of the standby power sources during loss of the offsite source. This test verifies all actions encountered from the loss of offsite power, including shedding of the nonessential loads and energization of the emergency buses and respective loads from the DG. It further demonstrates the capability of the DG to automatically achieve the required voltage and frequency within the specified time.

The DG autostart time of 10 seconds is derived from requirements of the accident analysis to respond to a design basis large break LOCA. The Surveillance should be continued for a minimum of 5 minutes in order to demonstrate

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.8.1.9 (continued)

that all starting transients have decayed and stability is achieved.

The requirement to verify the connection and power supply of permanent and auto connected loads is intended to satisfactorily show the relationship of these loads to the DG loading logic. In certain circumstances, many of these loads cannot actually be connected or loaded without undue hardship or potential for undesired operation. For instance, emergency Core Cooling Systems (ECCS) injection valves are not required to be stroked open, or high pressure injection systems are not capable of being operated at full flow, or residual heat removal (RHR) systems performing a decay heat removal function are not desired to be realigned to the ECCS mode of operation. In lieu of actual demonstration of connection and loading of loads, testing that adequately shows the capability of the DG systems to perform these functions is acceptable. This testing may include any series of sequential, overlapping, or total steps so that the entire connection and loading sequence is verified.

This SR is modified by three Notes. The reason for Note 1 is to minimize wear and tear on the DGs during testing. For the purpose of this testing, the DGs must be started from standby conditions, that is, with the engine coolant and oil continuously circulated and temperature maintained consistent with manufacturer recommendations. The reason for Note 2 is that performing the Surveillance would remove a required offsite circuit from service, perturb the electrical distribution system, and challenge safety systems. Note 3 to this SR permits removal of the bypass for protective trips after the DG has properly assumed its loads on the bus. This reduces exposure of the DG to undue risk of damage that might render it inoperable.

SR 3.8.1.10

This Surveillance demonstrates that the DG automatically starts and achieves the required voltage and frequency within the specified time (10 seconds) from the design basis actuation signal (LOCA signal) and operates for ≥ 5 minutes. Stable operation at the nominal voltage and frequency values

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.8.1.10 (continued)

is also essential to establishing DG OPERABILITY, but a time constraint is not imposed. This is because a typical DG will experience a period of voltage and frequency oscillations prior to reaching steady state operation if these oscillations are not damped out by load application. This period may extend beyond the 10 second acceptance criteria and could be a cause for failing the SR. In lieu of a time constraint in the SR, HBRSEP Unit No. 2 will monitor and trend the actual time to reach steady state operation as a means of assuring there is no voltage regulator or governor degradation which could cause a DG to become inoperable. The 5 minute period provides sufficient time to demonstrate stability. SR 3.8.1.10.d and SR 3.8.1.10.e ensure that permanently connected loads and emergency loads are energized from the offsite electrical power system on an ESF signal without loss of offsite power.

The requirement to verify the connection of permanent and autoconnected loads is intended to satisfactorily show the relationship of these loads to the DG loading logic. In certain circumstances, many of these loads cannot actually be connected or loaded without undue hardship or potential for undesired operation. For instance, ECCS injection valves are not required to be stroked open, or high pressure injection systems are not capable of being operated at full flow, or RHR systems performing a decay heat removal function are not desired to be realigned to the ECCS mode of operation. In lieu of actual demonstration of connection and loading of loads, testing that adequately shows the capability of the DG system to perform these functions is acceptable. This testing may include any series of sequential, overlapping, or total steps so that the entire connection and loading sequence is verified.

The Frequency of 18 months takes into consideration unit conditions required to perform the Surveillance and is intended to be consistent with the expected fuel cycle lengths. Operating experience has shown that these components usually pass the SR when performed at the 18 month Frequency. Therefore, the Frequency was concluded to be acceptable from a reliability standpoint.

This SR is modified by three Notes. The reason for Note 1 is to minimize wear and tear on the DGs during testing. For

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.8.1.10 (continued)

the purpose of this testing, the DGs must be started from standby conditions, that is, with the engine coolant and oil continuously circulated and temperature maintained consistent with manufacturer recommendations. The reason for Note 2 is that during operation with the reactor critical, performance of this Surveillance could cause perturbations to the electrical distribution systems that could challenge continued steady state operation and, as a result, unit safety systems. Note 3 to this SR permits removal of the bypass for protective trips after the DG has properly assumed its loads on the bus. This reduces exposure of the DG to undue risk of damage that might render it inoperable.

SR 3.8.1.11

This Surveillance demonstrates that DG noncritical protective functions (e.g., high coolant water temperature) are bypassed and critical protective functions (engine overspeed) trip the DG to avert substantial damage to the DG unit. A manual switch is provided which bypasses the non-critical trips. The noncritical trips are normally bypassed during DBAs and provide an alarm on an abnormal engine condition. This alarm provides the operator with sufficient time to react appropriately. The DG availability to mitigate the DBA is more critical than protecting the engine against minor problems that are not immediately detrimental to emergency operation of the DG. This SR is satisfied by simulating a trip signal to each of the non-critical trip devices and observing the DG does not receive a trip signal.

The 18 month Frequency is based on engineering judgment, taking into consideration unit conditions required to perform the Surveillance, and is intended to be consistent with expected fuel cycle lengths. Operating experience has shown that these components usually pass the SR when performed at the 18 month Frequency. Therefore, the Frequency was concluded to be acceptable from a reliability standpoint.

(continued)

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.8.1.12

This SR requires demonstration once per 18 months that the DGs can start and run continuously at full load capability for an interval of not less than 24 hours, ≥ 1.75 hours of which is at a load equivalent to 110% of the continuous duty rating and the remainder of the time at a load equivalent to the continuous duty rating of the DG. The DG start shall be a manually initiated start followed by manual syncronization with other power sources. Additionally, the DG starts for this Surveillance can be performed either from standby or hot conditions. The provisions for prelubricating and warmup, discussed in SR 3.8.1.2, and for gradual loading, discussed in SR 3.8.1.3, are applicable to this SR.

In order to ensure that the DG is tested under load conditions that are as close to design conditions as possible, testing must be performed using a power factor of ≤ 0.9. This power factor is chosen to be representative of the actual design basis inductive loading that the DG would experience. The load band is provided to avoid routine overloading of the DG. Routine overloading may result in more frequent teardown inspections in accordance with vendor recommendations in order to maintain DG OPERABILITY. The 18 month Frequency takes into consideration unit conditions required to perform the Surveillance, and is intended to be consistent with expected fuel cycle lengths.

This Surveillance is modified by three Notes. Note 1 states that momentary transients due to changing bus loads do not invalidate this test. Similarly, momentary power factor transients above the power factor limit will not invalidate the test. The reason for Note 2 is that during operation with the reactor critical, performance of this Surveillance could cause perturbations to the electrical distribution systems that could challenge continued steady state operation and, as a result, unit safety systems. Note 3 to this SR permits removal of the bypass for protective trips after the DG has properly assumed its loads on the bus. This reduces exposure of the DG to undue risk of damage that might render it inoperable.

(continued)

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.8.1.13

This Surveillance demonstrates that the diesel engine can restart from a hot condition, such as subsequent to shutdown from normal Surveillances, and achieve the required voltage and frequency within 10 seconds. The 10 second time is derived from the requirements of the accident analysis to respond to a design basis large break LOCA. Stable operation at the nominal voltage and frequency values is also essential to establishing DG OPERABILITY, but a time constraint is not imposed. This is because a typical DG will experience a period of voltage and frequency oscillations prior to reaching steady state operation if these oscillations are not damped out by load application. This period may extend beyond the 10 second acceptance criteria and could be a cause for failing the SR. In lieu of a time constraint in the SR, HBRSEP Unit No. 2 will monitor and trend the actual time to reach steady state operation as a means of assuring there is no voltage regulator or governor degradation which could cause a DG to become inoperable. The 18 month Frequency is based on engineering judgement and is intended to be consistent with expected fuel cycle lengths.

This SR is modified by two Notes. Note 1 ensures that the test is performed with the diesel sufficiently hot. The load band is provided to avoid routine overloading of the DG. Routine overloads may result in more frequent teardown inspections in accordance with vendor recommendations in order to maintain DG OPERABILITY. The requirement that the diesel has operated for at least 2 hours at full load conditions prior to performance of this Surveillance is based on manufacturer recommendations for achieving hot conditions. Momentary transients due to changing bus loads do not invalidate this test. Note 2 allows all DG starts to be preceded by an engine prelube period to minimize wear and tear on the diesel during testing.

SR 3.8.1.14

Under accident and loss of offsite power conditions, loads are sequentially connected to the bus by the automatic load sequencer. The sequencing logic controls the permissive and starting signals to motor breakers to prevent overloading of the DGs due to high motor starting currents. The 0.4

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.8.1.14 (continued)

seconds load sequence time interval tolerance ensures that sufficient time exists for the DG to restore frequency and voltage prior to applying the next load and that safety analysis assumptions regarding ESF equipment time delays are not violated. Reference 2 provides a summary of the automatic loading of ESF buses.

The Frequency of 18 months takes into consideration unit conditions required to perform the Surveillance, and is intended to be consistent with expected fuel cycle lengths.

This SR is modified by a Note. The reason for the Note is that performing the Surveillance would remove a required offsite circuit from service, perturb the electrical distribution system, and challenge safety systems.

SR 3.8.1.15

In the event of a DBA coincident with a loss of offsite power, the DGs are required to supply the necessary power to ESF systems so that the fuel, RCS, and containment design limits are not exceeded.

This Surveillance demonstrates the DG operation, as discussed in the Bases for SR 3.8.1.9, during a loss of offsite power actuation test signal in conjunction with an ESF actuation signal. In lieu of actual demonstration of connection and loading of loads, testing that adequately shows the capability of the DG system to perform these functions is acceptable. This testing may include any series of sequential, overlapping, or total steps so that the entire connection and loading sequence is verified.

The Frequency of 18 months takes into consideration unit conditions required to perform the Surveillance and is intended to be consistent with an expected fuel cycle length of 18 months.

This SR is modified by three Notes. The reason for Note 1 is to minimize wear and tear on the DGs during testing. For the purpose of this testing, the DGs must be started from standby conditions, that is, with the engine coolant and oil continuously circulated and temperature maintained

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.8.1.15 (continued)

consistent with manufacturer recommendations for DGs. The reason for Note 2 is that the performance of the Surveillance would remove a required offsite circuit from service, perturb the electrical distribution system, and challenge safety systems. Note 3 to this SR permits removal of the bypass for protective trips after the DG has properly assumed its loads on the bus. This reduces exposure of the DG to undue risk of damage that might render it inoperable.

SR 3.8.1.16

Transfer of the 4.160 kV bus 2 power supply from the auxiliary transformer to the start up transformer demonstrates the OPERABILITY of the offsite circuit network to power the shutdown loads. In lieu of actually initiating a circuit transfer, testing that adequately shows the capability of the transfer is acceptable. This transfer testing may include any sequence of sequential, overlapping, or total steps so that the entire transfer sequence is verified. The 18 month Frequency is based on engineering judgement taking into consideration the plant conditions required to perform the Surveillance, and is intended to be consistent with expected fuel cycle length.

This SR is modified by two Notes. The reason for Note 1 is that, during operation with the reactor critical, performance of this SR could cause perturbations to the electrical distribution systems that could challenge continued steady state operation and, as a result, unit safety systems. As stated in Note 2, automatic transfer capability to the SUT is not required to be met when the associated 4.160 kV bus and Emergency Bus are powered from the SUT. This is acceptable since the automatic transfer capability function has been satisfied in this condition.

SR 3.8.1.17

This Surveillance demonstrates that the DG starting independence has not been compromised. Also, this Surveillance demonstrates that each engine can achieve proper speed within the specified time when the DGs are started simultaneously. Stable operation at the nominal

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.8.1.17 (continued)

voltage and frequency values is also essential to establishing DG OPERABILITY, but a time constraint is not imposed. This is because a typical DG will experience a period of voltage and frequency oscillations prior to reaching steady state operation if these oscillations are not damped out by load application. This period may extend beyond the 10 second acceptance criteria and could be a cause for failing the SR. In lieu of a time constraint in the SR, HBRSEP Unit No. 2 will monitor and trend the actual time to reach steady state operation as a means of assuring there is no voltage regulator or governor degradation which could cause a DG to become inoperable.

The 10 year Frequency is based on engineering judgement.

This SR is modified by a Note. The reason for the Note is to minimize wear on the DG during testing. For the purpose of this testing, the DGs must be started from standby conditions, that is, with the engine coolant and oil continuously circulated and temperature maintained consistent with manufacturer recommendations.

REFERENCES

1. UFSAR, Section 3.1.
2. UFSAR, Chapter 8.
3. UFSAR, Chapter 6.
4. UFSAR, Chapter 15.
5. Generic Letter 84-15, "Proposed Staff Actions to Improve and Maintain Diesel Generator Reliability," July 2, 1984.
6. Regulatory Guide 1.137, Rev. 1, October 1979.
7. Regulatory Guide 1.9, Rev. 3, July 1993.
8. Regulatory Guide 1.108, Rev. 1, August 1977.

(continued)

BASES

REFERENCES (continued)

9. Regulatory Guide 1.93, Rev. 0, December 1974.

BASES

BACKGROUND (continued)

The DC power distribution system is described in more detail in Bases for LCO 3.8.9, "Distribution System – Operating," and LCO 3.8.10, "Distribution Systems – Shutdown and During Movement of Irradiated Fuel Assemblies."

Each battery has adequate storage capacity to carry the required load continuously for at least 1 hour (Ref. 1).

There is no sharing between redundant subsystems, such as batteries, battery chargers, or distribution panels.

The battery for Train A DC electrical power subsystem is sized to produce required capacity at 80% of nameplate rating, corresponding to warranted capacity at end of life cycles and the 100% design demand. Battery size is based on 125% of required capacity and, after selection of an available commercial battery, results in a battery capacity in excess of 150% of required capacity. The battery for Train B DC electrical power subsystem is sized to produce required capacity at 91% of nameplate rating, corresponding to warranted capacity at end of life cycles and the 100% design demand. Battery size is based on 110% of required capacity and, after selection of an available commercial battery, results in a battery capacity in excess of 120% of required capacity. The voltage limit is 2.13 V per cell, which corresponds to a total minimum voltage output of 128 V per battery.

Each Train A and Train B DC electrical power subsystem has ample power output capacity for the steady state operation of connected loads required during normal operation, while at the same time maintaining its battery bank fully charged. Each battery charger also has sufficient capacity to restore the battery from a partial discharge condition to its fully charged state within 24 hours while supplying normal steady state loads discussed in the UFSAR, Chapter 8 (Ref. 2).

APPLICABLE SAFETY ANALYSES

The initial conditions of Design Basis Accident (DBA) and transient analyses in the UFSAR, Chapter 6 (Ref. 3), and in the UFSAR, Chapter 15 (Ref. 4), assume that Engineered Safety Feature (ESF) systems are OPERABLE. The DC electrical power system provides normal and emergency DC

(continued)

BASES

ACTIONS

B.1 and B.2 (continued)

required unit conditions from full power conditions in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.8.4.1

Verifying battery terminal voltage while on float charge for the batteries helps to ensure the effectiveness of the charging system and the ability of the batteries to perform their intended function. Float charge is the condition in which the charger is supplying the continuous charge required to overcome the internal losses of a battery (or battery cell) and maintain the battery (or a battery cell) in a fully charged state. The voltage requirements are based on the nominal design voltage of the battery and are consistent with the initial voltages assumed in the battery sizing calculations and permit a single battery cell to be jumpered out. The 7 day Frequency is consistent with manufacturer recommendations and IEEE-450 (Ref. 5).

SR 3.8.4.2

Visual inspection of the battery cells, cell plates, and battery racks provides an indication of physical damage or abnormal deterioration that could potentially degrade battery performance.

The 18 month frequency is based on engineering judgement and operational experience and is sufficient to detect battery and rack degradation on a long term basis.

SR 3.8.4.3

Visual inspection of intercell, intertier, and terminal connections provide an indication of physical damage or abnormal deterioration that could indicate degraded battery condition. The anticorrosion material is used to help ensure good electrical connections and to reduce terminal deterioration. The visual inspection for corrosion is not intended to require removal of and inspection under each

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.8.4.5 (continued)

The reason for Note 2 is that performing the Surveillance would perturb the electrical distribution system and challenge safety systems.

SR 3.8.4.6

A battery performance discharge test is a test of constant current capacity of a battery, normally done in the as found condition, after having been in service, to detect any change in the capacity determined by the acceptance test. The test is intended to determine overall battery degradation due to age and usage.

The performance discharge test may be used to satisfy SR 3.8.4.6 while satisfying the requirements of SR 3.8.4.5 at the same time.

The acceptance criteria for this Surveillance are consistent with IEEE-450 (Ref. 5). These references recommend that the battery be replaced if its capacity is below 80% of the manufacturer's rating. A capacity of 80% shows that the battery rate of deterioration is increasing, even if there is ample capacity to meet the load requirements. An acceptance criterion of 80% of rated capacity is applicable to the "A" battery only. An acceptance criterion of 91% is applicable to the "B" battery since the battery's capacity is not as great.

The Surveillance Frequency for this test is normally 60 months. If the battery shows degradation, or if the battery has reached 85% for Battery "A" or 95% for Battery "B" of its expected life, the Surveillance Frequency is reduced to 18 months. Degradation is indicated, according to IEEE-450 (Ref. 5), when the battery capacity drops by more than 10% relative to its capacity on the previous performance test or when it is ≥ 10% below the manufacturer's rating. These Frequencies are generally consistent with the recommendations in IEEE-450 (Ref. 5) with an extra allowance for a 18 month test frequency for batteries which have shown degradation or have reached 85% for battery "A" and 95% for battery "B" of expected life.

(continued)

BASES

ACTIONS

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

considered reasonable. The verification is repeated at 7 day intervals until the parameters are restored to Category A or B limits. This periodic verification is consistent with the normal Frequency of pilot cell Surveillances.

Continued operation is only permitted for 31 days before battery cell parameters must be restored to within Category A and B limits. With the consideration that, while battery capacity is degraded, sufficient capacity exists to perform the intended function and to allow time to fully restore the battery cell parameters to normal limits, this time is acceptable prior to declaring the battery inoperable.

B.1

With one or more batteries with one or more battery cell parameters outside the Category C limit for any connected cell, sufficient capacity to supply the maximum expected load requirement is not assured and the corresponding DC electrical power subsystem must be declared inoperable. Additionally, other potentially extreme conditions, such as not completing the Required Actions of Condition A within the required Completion Time or average electrolyte temperature of representative cells falling below 67°F are also cause for immediately declaring the associated DC electrical power subsystem inoperable.

SURVEILLANCE REQUIREMENTS

SR 3.8.6.1

This SR verifies that Category A battery cell parameters are consistent with IEEE-450 (Ref. 3), which recommends regular battery inspections (at least one per month) including voltage (measured to the nearest 0.01 Volts), specific gravity, and electrolyte temperature of pilot cells. In addition, if water is added to any pilot cell, the amount must be recorded. Data attained must be compared to the data from the previous SR to detect signs of abuse or deterioration.

(continued)

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.8.6.2

The quarterly inspection of specific gravity and voltage is consistent with IEEE-450 (Ref. 3). In addition, within 24 hours of a battery discharge < 110 V or a battery overcharge > 150 V, the battery must be demonstrated to meet Category B limits. Transients, which may momentarily cause battery voltage to drop to ≤ 110 V, do not constitute a battery discharge provided the battery terminal voltage and float current return to pre-transient values. This inspection is also consistent with IEEE-450 (Ref. 3), which recommends special inspections following a severe discharge or overcharge, to ensure that no significant degradation of the battery occurs as a consequence of such discharge or overcharge. If water is added to any battery cell, the amount must be recorded. Data obtained must be compared to the data from the previous SR to detect signs of abuse or deterioration.

SR 3.8.6.3

This Surveillance verification that the average temperature of representative cells is ≥ 67°F is consistent with a recommendation of IEEE-450 (Ref. 3), that states that the temperature of electrolytes in representative cells should be determined on a quarterly basis. Data obtained must be compared to the data from the previous SR to detect signs of abuse or deterioration.

Lower than normal temperatures act to inhibit or reduce battery capacity. This SR ensures that the operating temperatures remain within an acceptable operating range. This limit is based on manufacturer recommendations.

Table 3.8.6-1

This table delineates the limits on electrolyte level, float voltage, and specific gravity for three different categories. The meaning of each category is discussed below.

Category A defines the normal parameter limit for each designated pilot cell in each battery. The cells selected

(continued)

B 3.8 ELECTRICAL POWER SYSTEMS

B 3.8.8 AC Instrument Bus Sources – Shutdown

BASES

BACKGROUND

A description of the AC Instrument Bus Sources is provided in the Bases for LCO 3.8.7, "AC Instrument Bus Sources – Operating."

APPLICABLE SAFETY ANALYSES

The initial conditions of Design Basis Accident (DBA) and transient analyses in the UFSAR, Chapter 6 (Ref. 1) and Chapter 15 (Ref. 2), assume Engineered Safety Feature systems are OPERABLE. The AC Instrument Bus Sources are designed to provide the required capacity, capability, redundancy, and reliability to ensure the availability of necessary power to the Reactor Protective System and Engineered Safety Features Actuation System instrumentation and controls so that the fuel, Reactor Coolant System, and containment design limits are not exceeded.

The OPERABILITY of the AC Instrument Bus Sources is consistent with the initial assumptions of the accident analyses and the requirements for the supported systems' OPERABILITY.

The OPERABILITY of the minimum AC Instrument Bus Sources to each AC instrument bus during MODES 5 and 6 ensures that:

a. The unit can be maintained in the shutdown or refueling condition for extended periods;

b. Sufficient instrumentation and control capability is available for monitoring and maintaining the unit status; and

c. Adequate power is available to mitigate events postulated during shutdown, such as a fuel handling accident.

The AC Instrument Bus Sources were previously identified as part of the distribution system and, as such, satisfy Criterion 3 of the NRC Policy Statement.

(continued)

BASES (continued)

LCO

The AC Instrument Bus Sources ensure the availability of electrical power for the instrumentation for systems required to shut down the reactor and maintain it in a safe condition after an anticipated operational occurrence or a postulated DBA. At least one AC instrument bus train energized by one battery powered inverter or a constant voltage transformer (CVT) ensure that the preferred source of AC instrument bus electrical power is available to at least one AC instrument bus. OPERABILITY of the inverters and CVTs requires that the AC instrument bus be powered by the associated inverter or CVT, as applicable. When the redundant train of the AC instrument bus electrical power distribution subsystem is required by LCO 3.8.10, the power source for this AC instrument bus may consist of:

1) the inverter powered by its associated battery;

2) the CVT; or

3) an offsite circuit providing power through a motor control center.

This ensures the availability of sufficient AC Instrument Bus Sources to operate the unit in a safe manner and to mitigate the consequences of postulated events during shutdown (e.g., fuel handling accidents).

APPLICABILITY

The AC Instrument Bus Sources required to be OPERABLE in MODES 5 and 6 and during movement of irradiated fuel assemblies provide assurance that:

a. Systems to provide adequate coolant inventory makeup are available for the irradiated fuel in the core;

b. Systems needed to mitigate a fuel handling accident are available;

c. Systems necessary to mitigate the effects of events that can lead to core damage during shutdown are available; and

d. Instrumentation and control capability is available for monitoring and maintaining the unit in a cold shutdown condition or refueling condition.

(continued)

BASES

APPLICABILITY (continued)

AC Instrument Bus Sources requirements for MODES 1, 2, 3, and 4 are covered in LCO 3.8.7.

ACTIONS

LCO 3.0.3 is not applicable while in MODE 5 or 6. However, since irradiated fuel assembly movement can occur in MODE 1, 2, 3 or 4, the ACTIONS have been modified by a Note stating that LCO 3.0.3 is not applicable. If moving fuel assemblies while in MODE 1, 2, 3 or 4, the fuel movement is independent of reactor operations. Therefore, in either case, inability to suspend movement of irradiated fuel assemblies would not be sufficient reason to require a reactor shutdown.

A.1, A.2.1, A.2.2, A.2.3, and A.2.4

With one or more required AC instrument bus sources inoperable when two trains are required by LCO 3.8.10, "Distribution Systems – Shutdown," the remaining OPERABLE AC Instrument Bus Sources may be capable of supporting sufficient required features to allow continuation of CORE ALTERATIONS, fuel movement, and operations with a potential for positive reactivity additions. By the allowance of the option to declare required features inoperable with the associated AC Instrument Bus Source inoperable, appropriate restrictions will be implemented in accordance with the affected required features LCOs' Required Actions. In many instances, this option may involve undesired administrative efforts. Therefore, the allowance for sufficiently conservative actions is made (i.e., to suspend CORE ALTERATIONS, movement of irradiated fuel assemblies, and operations involving positive reactivity additions). The Required Action to suspend positive reactivity additions does not preclude actions to maintain or increase reactor vessel inventory, provided the required SDM is maintained.

Suspension of these activities shall not preclude completion of actions to establish a safe conservative condition. These actions minimize the probability of the occurrence of postulated events. It is further required to immediately initiate action to restore the required AC Instrument Bus Sources and to continue this action until restoration is accomplished in order to provide the necessary AC Instrument Bus Source of power to the unit safety systems.

(continued)

BASES

ACTIONS (continued)

A.1, A.2.1, A.2.2, A.2.3, and A.2.4

The Completion Time of immediately is consistent with the required times for actions requiring prompt attention. The restoration of the required AC Instrument Bus Sources should be completed as quickly as possible in order to minimize the time the unit safety systems may be without power or powered from a non-preferred source.

SURVEILLANCE REQUIREMENTS

SR 3.8.8.1

This Surveillance verifies that the inverters are functioning properly with all required circuit breakers closed and required AC instrument buses energized from the inverter and that required circuit breakers are closed and required instrument buses are energized from the CVTs or other sources, as allowed by LCO 3.8.8.b. The verification of proper voltage and frequency output for the inverters ensures that the required power is readily available for the instrumentation connected to the associated AC instrument buses. The 7 day Frequency takes into account the redundant capability of the AC Instrument Bus Sources, other indications available in the control room that alert the operator to inverter malfunctions, and administrative requirements governing alignment of electrical equipment.

This SR is modified by a Note which states that voltage and frequency measurement is not required for the AC instrument buses supplied from CVTs. For these buses, observing status lights, instrument displays, etc. is sufficient to confirm that the required power is readily available to the AC instrument buses supplied from CVTs

REFERENCES

1. UFSAR, Chapter 6.

2. UFSAR, Chapter 15.

BASES

BACKGROUND (continued)

The secondary AC electrical power distribution system for each train includes the safety related motor control centers, and distribution panels shown in Table B 3.8.9-1. The Auxiliary Feedwater (AFW) Header Discharge Valve to S/G "A", V2-16A and the Service Water System (SWS) Turbine Building Supply Valve (emergency supply), V6-16C are powered from both Train A and Train B of the AC electrical bus distribution system by utilization of Automatic Bus Transfer (ABT) devices and molded case circuit breakers connected to each AC distribution train. Magnetic trip elements for these circuit breakers (two breakers per valve) provide circuit protection to prevent common mode failure (i.e., transfer of a fault from one electrical bus to the redundant bus) of both trains of the AC distribution systems.

The 120 VAC instrument buses are arranged in two load groups per train. One load group is made up of two instrument buses normally powered from an inverter. The remaining load group is made up of two instrument buses powered from a constant voltage transformer powered from the associated AC emergency bus. The alternate power supply for the inverter supplied instrument buses and the constant voltage transformer supplied instrument buses is an AC source powered from the station AC power distribution system, and its use is governed by LCO 3.8.7, "AC Instrument Bus Sources - Operating."

There are two redundant 125 VDC electrical power distribution subsystems (one for each train).

The list of all required distribution buses is presented in Table B 3.8.9-1.

APPLICABLE SAFETY ANALYSES

The initial conditions of Design Basis Accident (DBA) and transient analyses in the UFSAR, Chapter 6 (Ref. 1), and in the FSAR, Chapter 15 (Ref. 2), assume ESF systems are OPERABLE. The AC, DC, and AC instrument bus electrical power distribution systems are designed to provide sufficient capacity, capability, redundancy, and reliability to ensure the availability of necessary power to ESF systems so that the fuel, Reactor Coolant System, and containment design limits are not exceeded. These limits are discussed in more detail in the Bases for Section 3.2, Power

(continued)

SUPPLEMENT 5
CONVERSION PACKAGE SECTION 3.9
PAGE INSERTION INSTRUCTIONS

Remove and insert the following pages into Enclosure 17 to Serial: RNP-RA/96-0141.

	Remove Page	Insert Page
a.	Part 1, "Markup of Current Technical Specifications (CTS)"	
	3.8-1 (ITS 3.9.2) 3.8-1 (ITS 3.9.3)	3.8-1 (ITS 3.9.2) 3.8-1 (ITS 3.9.3)
	3.8-3, 3.8-2, 3.8-3	3.8-3, 3.8-2, 3.8-3
b.	Part 2, "Discussion of Changes (DOCs) for CTS Markup"	
	1 through 12	1 through 12
	-	13 through 15
c.	Part 3, " No Significant Hazards Consideration (NSHC), And Basis for Categorical Exclusion from 10 CFR 51.22	
	9 through 11	9 through 11
	-	12 through 14
d.	Part 4, "Markup of NUREG-4131, Revision 1, Standard Technical Specifications-Westinghouse Plants, (ISTS)"	
	3.9-13	3.9-13
	-	3.9-13i & 3.9-13ii
e.	Part 5, "Justification of Differences (JFDs) to ISTS"	
	1	1 &2
f.	Part 6, "Markup of ISTS Bases"	
	B 3.9-27	B 3.9-27
	-	B 3.9-27i, B 3.9-27ii, B 3.9-27iii, B 3.9-27 iv
g.	Part 7, "Justification for Differences (JFDs) to ISTS Bases"	
	1 & 2	1 & 2
h.	Part 8, "Proposed HBRSEP, Unit No. 2 ITS"	
	-	3.9-11 & 3.9-12
i.	Part 9. "Proposed Bases to HBRSEP, Unit No. 2 ITS Bases"	
	-	B 3.9-24 through B 3.9-27
j.	Part 10. "ISTS Generic Changes"	
	NA	

Specification 392

ITS

3.8 REFUELING

Applicability

Applies to operating limitations during refueling operations.

Objective

To minimize the possibility of an accident occurring during refueling operations that could affect public health and safety.

Specification

3.8.1 During refueling operations the following conditions shall be satisfied:

See 3.9.3

a. The equipment door and at least one door in the personnel air lock shall be properly closed. For those systems which provide a direct path from containment atmosphere to the outside atmosphere, all automatic containment isolation valves shall be operable or at least one valve shall be securely closed in each line penetrating the containment.

b. The containment vent and purge system, including the radiation monitors which initiate isolation shall be tested and verified to be operable immediately prior to refueling operations.

R1

c. Radiation levels in the containment and spent fuel storage areas shall be monitored continuously.

M3

[Applicability]
[LCO 3.9.2]

MODE 6

d. Whenever core geometry is being changed, core subcritical neutron flux shall be continuously monitored by at least two source range neutron monitors, each with continuous visual indication in the control room and one with audible

A11

LA4

ITS

3.8 REFUELING

A1

Applicability

Applies to operating limitations during refueling operations.

Objective

To minimize the possibility of an accident occurring during refueling operations that could affect public health and safety.

Specification

CORE ALTERATIONS, movement of irradiated fuel assemblies within containment

A4

[Applicability] 3.8.1 During refueling operations the following conditions shall be satisfied:

[LCO 3.9.3 a, b, c.1]

closed with 4 bolts

L2

a. The equipment door and at least one door in the personnel air lock shall be properly closed. For those systems which provide a direct path from containment atmosphere to the outside atmosphere, all automatic containment isolation valves shall be operable or at least one valve shall be securely closed in each line penetrating the containment.

L9

blind flange or equivalent

Manual or automatic

[SR 3.9.3.2]

b. The containment vent and purge system, including the radiation monitors which initiate isolation shall be tested and verified to be operable immediately prior to refueling operations. (18 mo)

See 3.3.6

L7

c. Radiation levels in the containment and spent fuel storage areas shall be monitored continuously.

R1

d. Whenever core geometry is being changed, core subcritical neutron flux shall be continuously monitored by at least two source range neutron monitors, each with continuous visual indication in the control room and one with audible

See 3.9.2

each valve actuates to isolation position on an actual or simulated actuation signal

A5

L8

ITS

A1

Suspend CORE ALTERATIONS

j. If any of the specified limiting conditions for refueling are not met, ~~refueling of the reactor shall cease~~ work shall be initiated to correct the conditions so that the specified limits are met and no operations which may increase the reactivity of the core shall be made.

[ACTION A]

L4

Suspend movement of irradiated fuel assemblies in containment

k. ~~The reactor shall be subcritical as required by 3.10.8.3.~~

A2

3.8.2 The Spent Fuel Building Filter system and the Containment Purge filter system shall satisfy the following conditions:

a. The results of the in-place cold DOP and halogenated hydrocarbon tests at greater than 20 percent design flows on HEPA filters and charcoal absorber banks shall show ≥ 99 percent DOP removal and ≥ 99 percent halogenated hydrocarbon removal.

See 5.5.11

b. Verification by way of laboratory carbon sample analysis from the Spent Fuel Building filter system carbon and the Containment Purge filter system carbon to show ≥ 90 percent radioactive methyl iodide removal in accordance with test 5.b of Table 5-1 of ANSI/ASME N509-1976 except that ≥ 70 percent relative humidity air is required.

c. 1. The Spent Fuel Building refueling filter fan shall be shown to operate within ± 10% of the design flow.

2. At least one Containment purge filter fan shall be shown to operate within ± 10% of the design flow and must be operable during core alterations or movement of irradiated fuel assemblies, or at least one automatic containment isolation valve in each line penetrating the containment which provides a direct path from the containment atmosphere to the outside atmosphere shall be securely closed.

See 3.9.7

d. During fuel handling operations, the relative humidity (R.H.) of the air processed by the refueling filter systems shall be ≤ 70 percent.

See 5.5.11.

e. ~~From and after the date that the Spent Fuel Building filter system~~ is made or found to be inoperable for any reason, fuel handling operations in the Spent Fuel Building shall be terminated immediately.

See 3.7.11

Add SR 3.9.3.1

M6

ITS

A1

indication available in the containment. When core geometry is not being changed at least one source range neutron flux monitor shall be in service.

e. At least one residual heat removal loop shall be operable, refueling cavity water level ≥ Plant elevation 272 ft. - 2 in. whenever fuel assemblies are being moved within the reactor pressure vessel, and Tave ≤ 140°F.

f. During reactor vessel head removal and while loading and unloading fuel from the reactor, the minimum boron concentration of 1950 ppm shall be maintained in the primary coolant system and verified by sampling once each shift.

g. Direct communication between the control room and the refueling cavity manipulator crane shall be available whenever changes in core geometry are taking place.

h. Movement of fuel within the core shall not be initiated prior to 100 hours after shutdown.

See 3.9.1 3.9.2 3.9.4 3.9.6

i. The Spent Fuel Building ventilation system shall be operating when handling irradiated fuel in this area. Prior to moving irradiated fuel assemblies in the spent fuel pool, the ventilation system exhaust shall be aligned to discharge through HEPA and impregnated charcoal filters. When in operation, the exhaust flow of the Containment Purge System shall discharge through HEPA and impregnated charcoal filters. When the Containment Purge System is not in operation at least one automatic containment isolation valve shall be secured in each line penetrating the containment which provides a direct path from the containment atmosphere to the outside atmosphere.

See 3.7.11

LA5

L9

[LCO 3.9.7]

[Required Action A1]

ITS

Suspend CORE ALTERATIONS
Suspend movement of irradiated fuel in containment

A.1

[Required Action A.2.1 & A.2.2]

j. If any of the specified limiting conditions for refueling are not met, refueling of the reactor shall cease; work shall be initiated to correct the conditions so that the specified limits are met; and no operations which may increase the reactivity of the core shall be made.

L4

k. The reactor shall be subcritical as required by 3.10.8.3.

See 3.9.1, 3.9.2, 3.9.3, 3.9.4, 3.5.6

3.8.2 The Spent Fuel Building Filter system and the Containment Purge filter system shall satisfy the following conditions:

a. The results of the in-place cold DOP and halogenated hydrocarbon tests at greater than 20 percent design flows on HEPA filters and charcoal absorber banks shall show ≥ 99 percent DOP removal and ≥ 99 percent halogenated hydrocarbon removal.

b. Verification by way of laboratory carbon sample analysis from the Spent Fuel Building filter system carbon and the Containment Purge filter system carbon to show ≥ 90 percent radioactive methyl iodide removal in accordance with test 5.b of Table 5-1 of ANSI/ASME N509-1976 except that ≥ 70 percent relative humidity air is required.

See 5.5.11

c. 1. The Spent Fuel Building refueling filter fan shall be shown to operate within ± 10% of the design flow.

2. At least one Containment purge filter fan shall be shown to operate within ± 10% of the design flow and must be operable during core alterations or movement of irradiated fuel assemblies, or at least one automatic containment isolation valve in each line penetrating the containment which provides a direct path from the containment atmosphere to the outside atmosphere shall be securely closed.

[LCO 3.9.7]

[APPLICABILITY]

[Required Action A.1]

L?

[SR 3.9.7.1]

d. During fuel handling operations, the relative humidity (R.H.) of the air processed by the refueling filter systems shall be ≤ 70 percent.

add frequency

M18

e. From and after the date that the Spent Fuel Building filter system is made or found to be inoperable for any reason, fuel handling operations in the Spent Fuel Building shall be terminated immediately.

See 3.7.11

Add SR 3.9.7.2
SR 3.9.7.3

M18

ADMINISTRATIVE CHANGES

A1 In the conversion of the H. B. Robinson Steam Electric Plant (HBRSEP), Unit 2 Current Technical Specifications (CTS) to the proposed plant specific Improved Technical Specifications (ITS) certain wording preferences or conventions are adopted which do not result in technical changes (either actual or interpretational). Editorial changes, reformatting, and revised numbering are adopted to make ITS consistent with the conventions in the Standard Technical Specifications, Westinghouse Plants, NUREG 1431, Revision 1 (i.e., Improved Standard Technical Specifications (ISTS)). These changes are administrative, and have no adverse impact on safety.

A2 CTS 3.8.1.k requires that the reactor be subcritical as required by CTS 3.10.8.3; and CTS 3.10.8.3 requires the shutdown margin to be at least 6 percent $\Delta k/k$ during refueling (relocation of CTS 3.10.8.3 is addressed in Discussion of Change LA3). CTS 3.8.1.f also requires a minimum boron concentration to be maintained during refueling operations. The requirement in CTS 3.8.1.f is included in ITS 3.9.1 with the actual value of minimum boron concentration relocated to the COLR. However, the Bases of ITS 3.9.1 states the minimum boron concentration requirement in the COLR ensures that core K_{eff} is maintained ≤ 0.94 (which is equivalent to 6 percent $\Delta k/k$). As a result, it is unnecessary to state that the reactor must be subcritical as required by CTS 3.10.8.3 since meeting the requirements of ITS 3.9.1 ensures that the reactor is subcritical with a shutdown margin equivalent to at least 6 percent $\Delta k/k$. Therefore, CTS 3.8.1.k is not retained in the ITS and the change is considered to be administrative with no adverse impact on safety.

A3 The CTS Bases are not retained in the ITS, but are replaced in their entirety. The ITS includes significantly expanded and improved Bases. The Bases do not define or impose any specific requirements but serve to explain, clarify and document the reasons (i.e., Bases) for the associated Specification. The Bases are not part of the Technical Specifications required by 10 CFR 50.36. This change is administrative, and has no adverse impact on safety.

A4 CTS Specification 3.8.1 has Applicability during "refueling operations." ITS Specification 3.9.3 has Applicability during "CORE ALTERATIONS," and "movement of irradiated fuel assemblies within containment." Since this change serves only to more clearly define the mode of Applicability, it is administrative and has no adverse impact on safety.

A5 CTS Specification 3.8.1.b requires the Containment Vent and Purge System be tested and verified to be OPERABLE. ITS Specification 3.9.3 requires that each valve actuate to the isolation position on an actual or simulated signal. The two tests actually accomplish the same objective. The addition of the allowance for actuation on an actual or simulated

signal is addressed in Discussion of Change L8. Therefore, this change is administrative, and has no adverse impact on safety.

A6 CTS 3.6.1.b requires that containment integrity not be violated when the reactor vessel head is removed unless a shutdown margin of at least 6% Δk/k is "constantly" maintained. ITS 3.9.1 requires a minimum boron concentration to be maintained at all times during MODE 6 (MODE 6 encompasses the condition with the reactor vessel head removed) with the actual value of minimum boron concentration relocated to the COLR. The Bases of ITS 3.9.1 states the minimum boron concentration requirement in the COLR ensures that core K_{eff} is maintained $\leq$ 0.94 (which is equivalent to 6% Δk/k). The requirement in CTS 3.6.1.b is unnecessary to be maintained in ITS and its elimination is considered to be administrative since entry into MODE 6 (and removal of the reactor vessel head) is precluded by ITS 3.9.1 Required Action A.3 when boron concentration is not within limits of ITS 3.9.1 (i.e., boron concentration is not sufficient to maintain a shutdown margin of at least 6% Δk/k). Since this change is administrative, it has no adverse impact on safety.

A7 CTS Specification 3.8.1.e requires the refueling cavity water level to be ≥ plant elevation 272 ft - 2 in. ITS Specifications 3.9.4 and 3.9.6 require the refueling cavity water level to be ≥ 23 feet above the top of the reactor vessel flange. The plant elevation datum relates directly to 23 feet above the top of the reactor vessel flange. This change is administrative, and has no adverse impact on safety.

A8 CTS Specification 3.8.1.e requires that the refueling cavity water level be ≥ plant elevation 272 feet 2 inches whenever fuel assemblies are being moved within the reactor pressure vessel. ITS Specification 3.9.6 requires that the refueling cavity water level be ≥ 23 feet above the top of the reactor vessel flange during CORE ALTERATIONS, except during latching and unlatching of control rod drive shafts. The definition of CORE ALTERATION includes movement of "reactivity control components." The CTS does not require level to be maintained during latching and unlatching operations; therefore, there is no change in requirements. This change is administrative, and has no adverse impact on safety.

A9 Not used.

A10 CTS Specification 3.8.1.j, which requires under certain circumstances, that work shall be initiated to correct the conditions so that the specified limits are met, is revised in ITS 3.9.4 Required Actions A.1, A.2, and A.3 to include a Completion Time of Immediately. Since the Completion Time of Immediately is implied in CTS 3.8.1.j, this change is administrative, and has no adverse impact on safety.

A11 CTS 3.8.1.d requires continuous monitoring of neutron flux to be performed by at least two source range monitors. ITS 3.9.2 requires two source range monitors to be OPERABLE. The CTS and ITS definition of

OPERABLE requires that the component be capable of performing its intended safety function. During refueling, the intended safety function of the source range monitors (as described in the HBRSEP Unit No. 2 UFSAR) is to monitor the reactivity of the core and to provide indication in the control room and the containment of any abnormal increase in core reactivity (i.e., neutron flux). Since ITS 3.9.2 requires the two source range monitors to be OPERABLE continuously during MODE 6 (refueling) and the definition of OPERABLE requires the source range monitors to be capable of monitoring neutron flux of the core, this change is considered to be administrative.

TECHNICAL CHANGES - MORE RESTRICTIVE

M1 CTS Specification 3.8.1.f has Applicability "during reactor vessel head removal and while loading and unloading fuel from the reactor." ITS Specification 3.9.1 has Applicability in MODE 6. This LCO is applicable in MODE 6 to ensure that the fuel in the reactor vessel will remain subcritical in this MODE. Since MODE 6 covers a much broader operational condition, this change is more restrictive and has no adverse impact on safety.

M2 CTS Specification 3.8.1.f requires a minimum boron concentration be maintained in the primary coolant system. ITS Specification 3.9.1 requires that a minimum boron concentration be maintained in the Reactor Coolant System, and in the refueling canal and refueling cavity, as well. This change is necessary, since in this MODE, the contents of the Reactor Coolant System, the refueling canal and refueling cavity are connected and intermixed. Since this change imposes new requirements, it is more restrictive and has no adverse impact on safety.

M3 CTS Specification 3.8.1.d has Applicability "whenever core geometry is being changed." ITS Specification 3.9.2 has Applicability in MODE 6. In MODE 6, the source range neutron flux monitors must be OPERABLE to determine changes in core reactivity. There are no other direct means available to check core reactivity levels. Since MODE 6 covers a much broader operational condition, this change is more restrictive and has no adverse impact on safety.

M4 The CTS is revised to adopt ISTS Specification 3.9.2, Required Action B.2, to provide assurance that any changes in boron concentration will be detected, since both source range flux monitors are inoperable. With no source range neutron flux monitor OPERABLE, there are no direct means of detecting changes in core reactivity. However, since CORE ALTERATIONS and positive reactivity additions are not to be made, the core reactivity condition is stabilized until the source range neutron flux monitors are OPERABLE. This stabilized condition is determined by performing SR 3.9.1.1 to ensure that the required boron concentration exists. The Completion Time of 4 hours is sufficient to obtain and

analyze a reactor coolant sample for boron concentration. The Frequency of once per 12 hours ensures that unplanned changes in boron concentration would be identified. The 12 hour Frequency is reasonable, considering the low probability of a change in core reactivity during this time period. Since this change imposes new requirements, it is more restrictive and has no adverse impact on safety.

M5 CTS Table 4.1-1, Item 3, which provides frequencies for checks and calibrations of Source Range Neutron Flux monitors, contains no requirements for performance of a CHANNEL CALIBRATION. ITS Specification 3.9.2 requires performance of a CHANNEL CALIBRATION every 18 months. A CHANNEL CALIBRATION requires adjustment of the channel such that channel output responds within a specified tolerance to a channel input. The 18 month Frequency is based on the need to perform this Surveillance under the conditions that apply during a plant outage. Operating experience has shown these components usually pass the Surveillance when performed at the 18 month Frequency. Since this change imposes new requirements, it is more restrictive and has no adverse impact on safety.

M6 The CTS is revised to adopt ITS SR 3.9.3.1, which requires a weekly verification that each required containment penetration is in the required status. This Surveillance demonstrates that each of the containment penetrations required to be in its closed position is in that position. This Surveillance ensures that a postulated fuel handling accident that releases fission product radioactivity within the containment will not result in a release of fission product radioactivity to the environment. Since this change imposes new requirements, it is more restrictive and has no adverse impact on safety.

M7 CTS Specification 3.8.1.e, which requires that at least one RHR loop be OPERABLE, is revised in ITS Specification 3.9.4 to require that at least one RHR train be OPERABLE, and in operation, and a NOTE is adopted which permits the required RHR train to be removed from operation for up to one hour in any 8 hour period. One RHR train must be in operation to provide:

a. Removal of decay heat;

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

c. Indication of reactor coolant temperature.

Since this change imposes new requirements, it is more restrictive and has no adverse impact on safety.

In addition, the term "loops" is revised to "trains" in reference to the RHR System for consistency with the HBRSEP Unit No. 2 plant specific description of the RHR System as reflected in UFSAR Section 5.4.4; UFSAR Section 5.4.4 describes that there is one loop of the RHR System. Therefore, to avoid confusion with the description of the RHR System in the UFSAR, the term "train" is used in place of the term "loop" in the ITS. The OPERABILITY requirements associated with a CTS RHR loop are the same as an ITS RHR train. Therefore, this portion of the change is administrative with no impact on safety.

M8 CTS Specification 3.8.1.j is revised in ITS 3.9.4 to require that, in addition to other actions, all penetrations providing direct access from containment atmosphere to outside atmosphere be closed within 4 hours. With the RHR train requirements not met, the potential exists for the coolant to boil and release radioactive gas to the containment atmosphere. Closing containment penetrations that are open to the outside atmosphere ensures dose limits are not exceeded. The Completion Time of 4 hours is reasonable, based on the low probability of the coolant boiling in that time. Since this change imposes new requirements, it is more restrictive and has no adverse impact on safety.

M9 The CTS is revised to adopt ITS SR 3.9.4.1 to require verification every 12 hours that one RHR train is in operation and circulating reactor coolant. This Surveillance requires verification every 12 hours that one RHR train is in operation. The Frequency of 12 hours is sufficient, considering the other indications and alarms available to the operator in the control room for monitoring the RHR System. Since no other similar Specification exists, this change is more restrictive and has no adverse impact on safety.

M10 The CTS is revised to adopt ITS Specification 3.9.5 to require that two RHR trains be OPERABLE, and one RHR train in operation when in MODE 6 with the water level < 23 feet above the top of the reactor vessel flange. If the reactor coolant temperature is not maintained below 200°F, boiling of the reactor coolant could result. This could lead to a loss of coolant in the reactor vessel. Additionally, boiling of the reactor coolant could lead to a reduction in boron concentration in the coolant due to the boron plating out on components near the areas of the boiling activity. The loss of reactor coolant and the reduction of boron concentration in the reactor coolant will eventually challenge the integrity of the fuel cladding, which is a fission product barrier. Two trains of the RHR System are required to be OPERABLE, and one train in operation, in order to prevent this challenge. Since no other similar Specification exists, this change is more restrictive and has no adverse impact on safety.

M11 CTS Specification 3.8.1.e has Applicability, "Whenever fuel assemblies are being moved within the reactor pressure vessel." ITS Specification

3.9.6 has Applicability, "during movement of irradiated fuel assemblies within containment." During CORE ALTERATIONS and movement of irradiated fuel assemblies, the water level in the refueling canal and the refueling cavity is an initial condition design parameter in the analysis of a fuel handling accident in containment. This change is appropriate since a fuel handling accident can also occur when handling irradiated fuel outside the reactor vessel. Since this change imposes new requirements, it is more restrictive and has no adverse impact on safety.

M12 The CTS is revised to adopt ITS SR 3.9.6.1, which requires verification every 12 hours that the refueling cavity water level is ≥ 23 feet above the top of the reactor vessel flange. Verification of a minimum water level of 23 ft above the top of the reactor vessel flange ensures that the design basis for the analysis of the postulated fuel handling accident during refueling operations is met. Water at the required level above the top of the reactor vessel flange limits the consequences of damaged fuel rods that are postulated to result from a fuel handling accident inside containment. The Frequency of 24 hours is based on engineering judgment and is considered adequate in view of the large volume of water and the normal procedural controls of valve positions, which make significant unplanned level changes unlikely. Since no similar Specification exists, this change is more restrictive and has no adverse impact on safety.

M13 CTS Specification 3.8.1.d is revised to add a Required Action to suspend positive reactivity additions in the event only one source range neutron flux monitor is OPERABLE, and this requirement is retained in ITS as LCO 3.9.2 Required Action A.2. CTS Specification 3.8.1.j, which requires that "refueling of the reactor" shall cease if any of the specifications are not met, is modified to restate "refueling of the reactor" as CORE ALTERATIONS. The incorporation of these CTS requirements into ITS Required Actions A.1 and A.2 is more restrictive because the actions now apply unequivocally to a single source range neutron flux monitor inoperable, rather than one or both monitors inoperable. With only one source range neutron flux monitor OPERABLE, redundancy has been lost. Since these instruments are the only direct means of monitoring core reactivity conditions, CORE ALTERATIONS and positive reactivity additions must be suspended immediately. This change has no adverse impact on safety.

M14 CTS Specification 3.8.1.a, which requires that all automatic containment isolation valves be operable or at least one valve be securely closed in each line penetrating the containment, is revised in ITS LCO 3.9.3.c.1 to require that at least one manual or automatic valve, blind flange, or equivalent be securely closed in each line penetrating the containment. Requiring closure of flowpaths without an automatic isolation valve is reasonable since releases can also occur via these pathways. This change is more restrictive and has no adverse impact on safety.

M15 CTS Specification 3.8.1.e, which applies the requirement for at least one RHR loop to be OPERABLE when fuel assemblies are being moved within the reactor pressure vessel, is revised in ITS for LCO 3.9.4 Applicability to MODE 6 when the water level is ≥ 23 ft. above the top of reactor vessel flange. The ITS Applicability is broader and more restrictive, and has no impact on safety.

M16 CTS Specification 3.8.1.e has Applicability, "Whenever fuel assemblies are being moved within the reactor pressure vessel." ITS Specification 3.9.6 has Applicability, "during CORE ALTERATIONS." The LCO minimizes the possibility of a fuel handling accident in containment that is beyond the assumptions of the safety analysis. Since this change imposes a broader Applicability to include movement of core and reactivity components, it is more restrictive and has no adverse impact on safety.

M17 CTS Specification 3.8.1.j is revised in ITS 3.9.6 to require that, in addition to other actions, that movement of irradiated fuel assemblies within containment be suspended. Suspending movement of irradiated fuel assemblies within the containment is necessary to ensure that a fuel handling accident cannot occur. Since this change imposes new requirements, it is more restrictive and has no adverse impact on safety.

M18 The CTS is revised to adopt ITS SR 3.9.7.1, SR 3.9.7.2, and SR 3.9.7.3 to require that the Containment Purge Filter System be verified OPERABLE and in operation. CTS 3.8.2.d requires that, during fuel handling operations, the relative humidity of air processed by the refueling filter systems (in this case the Containment Purge Filter System) shall be ≤ 70%. SR 3.9.7.1 requires verification, at a Frequency of once per hour, that the relative humidity of the containment atmosphere to be processed by the Containment Purge Filter System is ≤ 70%. Adding a Frequency for verification of relative humidity represents an additional restriction necessary to ensure that the testing performed to validate the safety analysis assumptions relative to charcoal filter efficiency, bounds actual plant conditions for relative humidity at the inlet of the Containment Purge Filter System charcoal filter. The one hour Frequency is based on engineering judgment considering the likelihood of changes in containment relative humidity during refueling outages. SR 3.9.7.2 requires verification that the Containment Purge Filter System is in operation and maintaining containment pressure negative relative to the adjacent auxiliary building areas once every 12 hours. This verification ensures that containment pressure is being maintained negative with respect to the outside atmosphere since the pressure of the auxiliary building areas is normally maintained negative with respect to the outside atmosphere. This change is necessary to ensure plant operation is consistent with the assumptions related to the capability of the Containment Purge Filter System to maintain a slight negative pressure in the containment. The Frequency of 12 hours is

sufficient considering other indications available to the operator to monitor Containment Purge Filter System operation. SR 3.9.7.3 verifies that the required Containment Purge Filter System testing is performed in accordance with the Ventilation Filter Testing Program (VFTP). The VFTP includes testing HEPA filter performance, charcoal adsorber efficiency, minimum system flow rate, and the physical properties of the activated charcoal (general use and following specific operations). This change is necessary to ensure the Containment Purge Filter System is maintained OPERABLE. Since these changes impose new requirements, they are more restrictive and have no adverse impact on safety.

TECHNICAL CHANGES - LESS RESTRICTIVE (GENERIC)

LA1 CTS Specification 3.8.1.f requires a minimum boron concentration of 1950 ppm. This detail is not retained in the ITS and is relocated to licensee controlled documents.

The details associated with this Specification are not required to be in the ITS to provide adequate protection of the public health and safety, since the ITS still retains system OPERABILITY requirements, including limitations on shutdown margin and/or boron concentration, where appropriate. This approach provides an effective level of regulatory control and provides for a more appropriate change control process. The level of safety of facility operation is unaffected by the change because there is no change in the overall operational requirements. Furthermore, NRC and licensee resources associated with processing license amendments to these requirements will be reduced. Therefore, relocation of these details is acceptable.

LA2 CTS Table 4.1-3 (Item 6), requires performance of functional checks on Refueling System Interlocks prior to each refueling shutdown. This detail is not retained in the ITS and is to be relocated to the Technical Requirements Manual (TRM). The TRM will be incorporated by reference into the UFSAR at ITS implementation.

The details associated with the involved Specification are not required to be in the ITS to provide adequate protection of the public health and safety, since the ITS still retains the requirement for OPERABILITY of systems required for refueling operations. The possibility of a fuel handling incident is remote because of the administrative controls and physical limitations imposed on fuel handling operations. All refueling operations are conducted in accordance with prescribed procedures, under direct supervision of a licensed SRO who has no other concurrent responsibilities during such operations. This approach provides an effective level of regulatory control and provides for a more appropriate change control process. The level of safety of facility operation is unaffected by the change because there is no change in the overall operational requirements. Furthermore, NRC and licensee

resources associated with processing license amendments to these requirements will be reduced. Therefore, relocation of these details is acceptable.

LA3 CTS Specification 3.10.8.3 requires the shutdown margin to be at least 6% Δk/k when the reactor is in the refueling operation mode. This detail is not retained in the ITS and is to be relocated to the CORE OPERATING LIMITS REPORT (COLR). Changes to COLR are controlled by the provisions of 10 CFR 50.59.

The details associated with this Specification are not required to be in the ITS to provide adequate protection of the public health and safety, since the ITS still retains the requirement that the boron concentration in the RCS, refueling cavity, and refueling canal be maintained within the limits specified in the COLR. This approach provides an effective level of regulatory control and provides for a more appropriate change control process. The level of safety of facility operation is unaffected by the change because there is no change in the overall operational requirements. Furthermore, NRC and licensee resources associated with processing license amendments to these requirements will be reduced. Therefore, relocation of this detail is acceptable.

LA4 CTS Specification 3.8.1.d requires the two Source Range Neutron Flux monitors to have continuous visual indication in the control room and one with audible indication available in containment. This detail is not retained in the ITS and is relocated to the Bases.

The details associated with this Specification are not required to be in the ITS to provide adequate protection of the public health and safety, since the ITS still retains the OPERABILITY requirements for the Source Range Neutron Flux instrumentation. This approach provides an effective level of regulatory control and provides for a more appropriate change control process. The level of safety of facility operation is unaffected by the change because there is no change in the operational requirements. Furthermore, NRC and licensee resources associated with processing license amendments to these requirements will be reduced. Therefore, relocation of this detail is acceptable.

LA5 CTS Specification 3.8.1.i requires, during refueling operations, that containment purge exhaust flow be discharged through HEPA and impregnated charcoal filters. This detail of the design and operation of the Containment Purge Filter System is to be relocated to the Bases. Changes to the Bases are to be controlled by the Bases Control Program in ITS Chapter 5.0.

ITS 3.9.7 requires the Containment Purge Filter System to be OPERABLE and in operation during CORE ALTERATIONS and during irradiated fuel movement in containment. In addition, the Containment Purge System design is such that, during refueling operations, purge exhaust flow

from containment is though the HEPA and impregnated charcoal filters. Therefore, the details associated with this Specification are not required to be in the ITS to provide adequate protection of the public health and safety. This approach provides an effective level of regulatory control and provides for a more appropriate change control process. The level of safety of facility operation is unaffected by the change because there is no change in the operational requirements. Furthermore, NRC and licensee resources associated with processing license amendments to these requirements will be reduced. Therefore, relocation of this detail is acceptable.

LA6 CTS Specification 3.8.1.e requires that during refueling operations, T_{avg} must be ≤ 140°F. This detail is not retained in the ITS and is to be relocated to the Technical Requirements Manual (TRM). The TRM will be incorporated by reference into the UFSAR at ITS implementation. The requirement to maintain reactor coolant temperature ≤ 140°F during refueling operations ensures that boiling of the reactor coolant does not occur. Boiling of the reactor coolant could lead to a reduction in reactor coolant in the reactor vessel and a reduction in boron concentration due to boron plating out on components near the areas of boiling activity. The reduction in reactor coolant and boron concentration could eventually challenge the integrity of the fuel cladding. During refueling operations, ITS 3.9.4 requires that one RHR train be in operation. With one RHR train in operation, decay heat is removed such that boiling of the reactor coolant and the potential challenge to the integrity of the fuel cladding are prevented. Therefore, the requirement to maintain reactor coolant temperature ≤ 140°F during refueling operations is not necessary to be maintained in ITS to provide adequate protection of the public health and safety.

This approach provides an effective level of regulatory control and provides for a more appropriate change control process. The level of safety of facility operation is unaffected by the change because there is no change in the operational requirements. Furthermore, NRC and licensee resources associated with processing license amendments to these requirements will be reduced. Therefore, relocation of this detail is acceptable.

LA7 CTS Specification 3.8.1.h requires that movement of fuel within the core not be initiated prior to 100 hours after shutdown. This detail is not retained in the ITS and is to be relocated to the Technical Requirements Manual (TRM). The TRM will be incorporated by reference into the UFSAR at ITS implementation.

Although CTS 3.8.1.h satisfies Criterion 2 of the Technical Specification Selection Criteria in 10 CFR 50.36(c)(2)(ii), the 100 hour decay time limit following subcriticality will be met for a refueling outage because of the activities required prior to moving irradiated fuel in the reactor vessel (e.g., plant cooldown to MODE 5 conditions,

removal of vessel head, flooding the reactor cavity, removal of vessel internals). Movement of irradiated fuel in the reactor vessel can not be conducted unless these activities are completed. These activities are normally not completed until after the 100 decay time limitation is satisfied due to the time required to complete each of the activities. Therefore, the requirement of CTS 3.8.1.h is not required to be in the Technical Specifications to provide adequate protection of the public health and safety. This approach provides an effective level of regulatory control and provides for a more appropriate change control process. The level of safety of facility operation is unaffected by the change because there is no change in the operational requirements. Furthermore, NRC and licensee resources associated with processing license amendments to these requirements will be reduced. Therefore, relocation of this detail is acceptable, and is consistent with NUREG-1431.

LA8 Not used.

TECHNICAL CHANGES - LESS RESTRICTIVE (SPECIFIC)

L1 CTS Specification 3.8.1.f requires that boron concentration be verified once each shift. ITS Specification 3.9.1 requires that boron concentration be verified at a Frequency of 72 hours. This is a relaxation of requirements, and is less restrictive. This change is acceptable, however, because industry operating experience has shown that 72 hours is a reasonable Frequency in which to verify the boron concentration of representative samples, considering that the limiting boron dilution event occurs in MODE 5, and the OPERABILITY requirements of the Source Range Neutron Flux instrumentation. In addition, a review of the surveillance test history was performed to validate that the impact, if any, on the capability to maintain boron concentration within the required limit is minimal as a result of the change in the surveillance test interval. This review of the surveillance test history, demonstrates that there are no failures that would invalidate the conclusion that the impact, if any, on the capability to maintain boron concentration within the required limit is minimal from a change to a 72 hour surveillance interval. This change is consistent with NUREG-1431.

L2 CTS Specification 3.8.1.a requires that the equipment door be properly closed during refueling operations. ITS Specification 3.9.3 requires that the equipment hatch be closed and held in place by 4 bolts. This is a relaxation of requirements, and is less restrictive. This change is acceptable, however, because the Applicability of this Specification is during a MODE when the RCS is cooled down and depressurized with the reactor head removed. In this MODE, the most severe radiological consequences result from a fuel handling accident. There are no accidents that could occur with the plant in this MODE that would produce sufficient pressure to require an air tight equipment hatch seal. This change is consistent with NUREG-1431.

L3 CTS Specification 3.8.1.i requires that, under certain conditions, one automatic containment isolation valve be securely closed in each line penetrating the containment. This requirement has been revised in ITS LCO 3.9.3.c.2 to require that each penetration be capable of being closed by an OPERABLE containment ventilation. This is a relaxation of requirements, and is less restrictive. This change is acceptable, however, because the requirements for isolation of the penetrations have not changed. This change in combination with the change to CTS Specification 3.8.1.a, which was incorporated into ITS LCO 3.9.3.a, 3.9.3.b, and 3.9.3.c.1, provides the same degree of protection required by the applicable safety analyses. This change is consistent with NUREG-1431.

L4 CTS Specification 3.8.1.j requires that, if the specified limiting conditions for refueling are not met, refueling of the reactor shall cease, work be initiated to correct the conditions so that the specified

limits are met, and no operations which may increase the reactivity of the core be made. ITS Specification 3.9.3 and 3.9.7 require that, under the same circumstances, that both CORE ALTERATIONS and movement of irradiated fuel assemblies be suspended. This is a relaxation of requirements because the CTS action to suspend operations which may increase the reactivity of the core is not retained in ITS, and is less restrictive. This change is acceptable, however, because taking these actions places the reactor in a MODE where the Specification no longer applies; and these actions provide the same degree of protection required by the applicable safety analyses. This change is consistent with NUREG-1431.

L5 CTS Specification 3.8.1.j, which requires that, in the evet that any of the specified LCOs for refueling are not met, refueling of the reactor shall cease, work shall be initiated to correct the conditions so that the specified limits are met, and no operations which may increase the reactivity of the core shall be made, is revised in ITS Required Action B.2 to apply only to the condition of two source range neutron flux monitors inoperable. This is a relaxation of requirements and is less restrictive. This change is acceptable because CTS Specification 3.8.1.d has been revised to incorporate ITS Required Actions A.1 and A.2, which also apply when both source range monitors are inoperable. This change is also acceptable because the Required Actions assure that operations that could result in a challenge to core reactivity due to refueling or boron concentration are ceased, with a Completion Time of Immediately, until a reactivity monitoring capability is restored. This change is consistent with NUREG-1431.

L6 CTS Specification 3.8.1.j, which requires that, in the event that any of the specified LCOs for refueling are not met refueling of the reactor shall cease, is revised in ITS Required Action A.2 to suspend loading irradiated fuel assemblies in the core immediately. This is a relaxation of requirements and is less restrictive because unloading of irradiated fuel assemblies is not prohibited. This change is acceptable because the ITS Required Action assures that operations that could result in a reduction in shutdown margin due to refueling operations are ceased, with a Completion Time of Immediately, until the RHR train requirements are met. This change is consistent with NUREG-1431.

L7 CTS 3.8.1.b requires the Containment Vent and Purge System... isolation shall be tested and verified to be OPERABLE immediately prior to refueling operations. ITS SR 3.9.3.2 requires verification that each required containment ventilation valve actuates to the isolation position on an actual or simulated actuation signal every 18 months. Therefore, the surveillance test interval of this Surveillance Requirement is being increased from prior to refueling operations to once every 18 months for a maximum interval of 22.5 months including the 25% grace period.

The subject SR ensures that the Containment Vent and Purge System will actuate as designed to isolate during an analyzed event. Extending the Frequency of the actuation test of this system is acceptable because the containment purge and vent valves are designed to be single failure proof and therefore are highly reliable. In addition, a review of the surveillance test history was performed to validate that the impact, if any, on system availability is minimal as a result of the change in the surveillance test interval to an 18 month actuation test frequency. This review of the surveillance test history demonstrates that there are no failures that would invalidate this conclusion.

L8 CTS 3.8.1.b requires Containment Vent and Purge System isolation capability to be tested and verified OPERABLE prior to refueling operations. ITS SR 3.9.3.2 allows verification by an actual or simulated actuation signal. This allows satisfactory actuations for other than Surveillance purposes to be used to fulfill the Surveillance Requirements. OPERABILITY is adequately demonstrated in either case since the Containment Vent and Purge System components cannot discriminate between an "actual" signal or a "test" signal.

L9 With a containment purge fan inoperable, CTS 3.8.2.c.2 requires at least one automatic containment isolation valve in each line penetrating the containment which provides a direct path from the containment atmosphere to the outside atmosphere to be securely closed. CTS Specification 3.8.1.a requires that all automatic containment isolation valves be operable or at least one valve be securely closed in each line penetrating the containment. ITS LCO 3.9.3.c.1 and LCO 3.9.3.c.2 require "Each penetration providing direct access from the containment atmosphere to the outside atmosphere to be closed by a manual or automatic isolation valve, blind flange, or equivalent, or capable of being closed by an OPERABLE Containment Ventilation Isolation System." ITS 3.9.7 Required Action A.1 requires closing "each penetration providing direct access from the containment atmosphere to the outside atmosphere by a manual or automatic valve, blind flange, or equivalent method."

CTS 3.8.1.a and 3.8.2.c.2 list acceptable containment isolation devices (automatic containment isolation valves which are closed or capable of being closed automatically) that may be used to satisfy the need to isolate a containment penetration during CORE ALTERATIONS and during movement of irradiated fuel assemblies in containment. ITS LCO 3.9.3.c.1, LCO 3.9.3.c.2, and ITS 3.9.7 Required Action A.1 provide a more complete list of acceptable isolation devices. The method of isolation must include the use of at least one isolation barrier, in each penetration, to ensure that a release of fission product radioactivity within containment will be restricted from escaping to the environment. Isolation barriers that meet this criterion are a closed manual or automatic isolation valve, a blind flange, or an equivalent method that can provide a temporary atmospheric pressure and ventilation

barrier during CORE ALTERATIONS or irradiated fuel movements in containment. In addition, isolation devices capable of being automatically closed by an OPERABLE Containment Ventilation Isolation System, when the containment purge fan is operable, also satisfy this criterion. Since the proposed isolation methods will continue to acceptably isolate the affected penetrations during the applicable conditions, the change does not adversely affect safe refueling operations.

RELOCATED SPECIFICATIONS

R1 3.8.1.c Continuous Monitoring of Radiation Levels

3.8.1.g Direct Communication (during refueling operations)

These Specifications, or Limiting Conditions for Operation (CTS Chapter 3.0), are not retained in the ITS because they have been reviewed against, and determined not to satisfy, the selection criteria for Technical Specifications provided in 10 CFR 50.36. The selection criteria were established to ensure that the Technical Specifications are reserved for those conditions or limitations on plant operation considered necessary to limit the possibility of an abnormal situation or event that could result in an immediate threat to the health and safety of the public. The rationale for relocation of each of these Specifications is provided in the report, "Application of Selection Criteria to the H. B. Robinson Steam Electric Plant Unit No. 2 Technical Specifications."

These Limiting Conditions for Operation, and their associated Surveillance Requirements (CTS Chapter 4.0), are relocated to licensee controlled documents. Relocation of the specific requirements for systems or variables contained in these Specifications to licensee documents will have no impact on the operability or maintenance of those systems or variables. The licensee will initially continue to meet the requirements contained in the relocated Specifications. The licensee is allowed to make changes to these requirements in accordance with the provisions of 10 CFR 50.59. Such changes can be made without prior NRC approval, if the change does not involve an unreviewed safety question, as defined in 10 CFR 50.59. These controls are considered adequate for assuring that structures, systems, and components in the relocated Specifications are maintained operable, and variables are maintained within limits. This change is consistent with the NRC Final Policy Statement on Technical Specification Improvements.

There are no margins of safety related to safety analyses that are dependent upon the proposed change. The requirements will continue to assure that limiting conditions for refueling are properly maintained. Therefore, this change does not involve a reduction in a margin of safety.

LESS RESTRICTIVE-SPECIFIC CHANGES
("L7" Labeled Comments/Discussions)

Carolina Power & Light Company 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 involves a change in surveillance testing intervals for Containment Vent and Purge System actuation test from prior to refueling to 18 months. The proposed change does not physically impact the plant nor does it impact any design or functional requirements of the associated systems. That is, the proposed change does not degrade the performance or increase the challenges of any safety systems assumed to function in the accident analysis. The proposed change does not impact the Surveillance Requirements themselves nor the way in which the Surveillances are performed. Additionally, the proposed change does not introduce any new accident initiators since no accidents previously evaluated have as their initiators anything related to the frequency of surveillance testing. The proposed change does not affect the availability of equipment or systems required to mitigate the consequences of an accident because of the availability of redundant systems or equipment and because other tests performed more frequently will identify potential equipment problems. Furthermore, a historical review of surveillance test results indicated that all failures identified were unique, non-repetitive, and not related to any time-based failure modes, and indicated no evidence of any failures that would invalidate the above conclusions. Therefore, the proposed change does not increase the probability or consequences of an accident previously evaluated.

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

   The proposed change involves a change in the surveillance testing intervals for Containment Vent and Purge System actuation test from prior to refueling to 18 months. The proposed change does not introduce any failure mechanisms of a different type than those previously evaluated since there are no physical changes being made to the facility. In addition, the Surveillance Requirements themselves and the manner in which Surveillances are performed will remain unchanged. Furthermore, a historical review of surveillance test results indicated no evidence of any failures that would invalidate the above conclusions. Therefore, the

proposed change does not create the possibility of a new or different kind of accident from any previously evaluated.

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

   The proposed change will not result in an increase in the interval between surveillance tests since refueling outages are conducted approximately every 18 months. Therefore, the impact on system availability is small. In addition, there is no evidence of any failures that would impact the availability of the systems. Therefore, the assumptions in the licensing basis are not impacted, and the proposed change does not involve a significant reduction in a margin of safety.

LESS RESTRICTIVE-SPECIFIC CHANGES
("L8" Labeled Comments/Discussions)

Carolina Power & Light Company 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 phrase "actual or," in reference to the automatic actuation signal, is added to the ITS Surveillance Requirements. This addition does not impose a requirement to create an "actual" signal, and does not eliminate restrictions on producing an "actual" signal. While creating an "actual" signal could increase the probability of an event, existing procedures and 10 CFR 50.59 control of revisions to them, dictate the acceptability of generating a test signal. The proposed change does not affect the procedures governing plant operations and the acceptability of creating test signals; it simply allows an actual signal to be utilized in evaluating the acceptance criteria associated with Surveillance Requirements. Therefore, the change does not involve a significant increase in the probability of an accident previously evaluated. Since the method of test initiation does not affect the acceptance criteria of the Surveillance Requirements, the change does not involve a significant increase in the 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 change does not introduce a new mode of plant operation and does not involve physical modification to the plant. Therefore, it does not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

   Use of an actual signal instead of the current Technical Specification requirement, which limits use to a simulated signal, will not affect the performance or acceptance criteria of the Surveillances. OPERABILITY is adequately demonstrated in either case (simulated or actual signal) since the system itself can not discriminate between "actual" or "simulated" signals. Therefore, the change does not involve a significant reduction in a margin of safety.

LESS RESTRICTIVE-SPECIFIC CHANGES
("L9" Labeled Comments/Discussions)

Carolina Power & Light Company 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 result in any hardware or operating procedure changes. The valves and devices that serve as containment isolation devices are not assumed to be initiators of any analyzed event. The role of these devices is to isolate containment during analyzed events, thereby limiting consequences. The proposed change does not allow continuous refueling operations when a fission product radioactivity release in containment could escape through an unisolated or unisolable path. The method of isolation proposed for ITS 3.9.3 and 3.9.7 must include the use of at least one isolation barrier, in each penetration, that is capable of restricting a fission product radioactivity release within containment from escaping to the environment. In addition, the change establishes allowances to use isolation barriers which are equivalent to those already included in the current Technical Specifications. Therefore, this change will 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 change does not introduce a new mode of plant operation and does not involve physical modification to the plant. The change still ensures the containment boundary is maintained. Therefore, it does not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

   The method of isolation proposed for ITS 3.9.3 and 3.9.7 must include the use of at least one isolation barrier, in each penetration, that is

capable of restricting a fission product radioactivity release within containment from escaping to the environment. In addition, the change establishes allowances to use isolation barriers which are equivalent to those already included in the current Technical Specifications. As a result, any reduction in a margin of safety will be insignificant and offset by the benefit gained through the use equivalent isolation devices that can provide atmospheric pressure and ventilation barriers to ensure fission product radioactivity within containment will be restricted from escaping to the environment during CORE ALTERATIONS or irradiated fuel movements in containment. Therefore, the change does not involve a significant reduction in a margin of safety.

RELOCATED CHANGES
("R" Labeled Comments/Discussions)

Relocating Requirements which do not meet the Technical Specification criteria to documents with an established control program allows the Technical Specifications to be reserved only for those conditions or limitations upon reactor operation which are necessary to adequately limit the possibility of an abnormal situation or event giving rise to an immediate threat to the public health and safety, thereby focusing the scope of Technical Specifications.

Therefore, requirements which do not meet the Technical Specification criteria in the NRC Final Policy Statement on Technical Specification Improvement for Nuclear Power Reactors (58FR 39132, dated 7/22/93) have been relocated to licensee controlled documents. This policy statement addresses the scope and purpose of Technical Specifications. In doing so, it establishes a specific set of objective criteria for determining which regulatory requirements and operating restrictions should be included in Technical Specifications. These criteria are as follows:

Criterion 1: Installed instrumentation that is used to detect and indicate in the control room, a significant abnormal degradation of the reactor coolant pressure boundary;

Criterion 2: A process variable that is an initial condition of a design basis accident (DBA) or transient analyses that either assumes the failure of or presents a challenge to the integrity of a fission product barrier;

Criterion 3: A structure, system or component that is part of the primary success path and which functions or actuates to mitigate a design basis accident or transient that either assumes the failure of or presents a challenge to the integrity of a fission barrier;

Criterion 4: A structure, system or component which operating experience or probabilistic safety assessment has shown to be significant to public health and safety.

The application of these criteria is provided in the "Application of Selection Criteria to the HBRSEP Unit No. 2 Technical Specifications." Requirements which met the criteria have been included in the proposed improved Technical Specifications. Carolina Power & Light (CP&L) proposes to remove the requirements which do not meet the criteria from the Technical Specifications and relocate the requirements to a suitable owner controlled document. The requirements in the relocated Specifications are not affected by this Technical Specification change. CP&L will initially continue to perform the required operation and maintenance to assure that the requirements are satisfied. Relocating specific requirements for systems or variables has no impact on the system's operability or the variable's maintenance, as applicable.

Licensee controlled programs will be utilized as the control mechanism for the relocated Specifications as they will be placed in plant procedures or other licensee controlled documents. CP&L is allowed to make changes to these requirements, without prior NRC approval, if the change does not involve an unreviewed safety question. These controls are considered adequate for assuring structures, systems and components in the relocated Specifications are maintained operable and variables in the relocated Specifications are maintained within limits.

Carolina Power & Light Company has evaluated each of the proposed Technical Specification changes identified as "Relocated" and has concluded that they do 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 proposed changes do 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 relocates requirements and surveillances for structures, systems, components or variables which did not meet the criteria for inclusion in Technical Specifications as identified in the "Application of Selection Criteria to the HBRSEP Unit No. 2 Technical Specifications." The affected structures, systems, components or variables are not assumed to be initiators of analyzed events and are not assumed to mitigate accident or transient events. The requirements and surveillances for these affected structures, systems, components or variables will be relocated from the Technical Specifications to an appropriate administratively controlled document under licensee control. Therefore, this change does not involve an increase in the probability or consequences of an accident previously evaluated.

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

   The proposed change does not necessitate a physical alteration of the plant (no new or different type of equipment will be installed) or change in parameters governing normal plant operation. The proposed change will not impose any different requirements and adequate control

of information will be maintained. Thus, this change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

The proposed change will not reduce a margin of safety because it has no impact on any safety analysis assumptions. In addition, the affected requirement will be relocated to an owner controlled document for which future changes will be evaluated pursuant to the requirements of licensee controlled programs. Therefore, this change does not involve a reduction in a margin of safety.

CTS

SURVEILLANCE REQUIREMENTS

	SURVEILLANCE		FREQUENCY
[M12]	SR 3.9.X.1	Verify refueling cavity water level is ≥ 23 ft above the top of reactor vessel flange.	24 hours

INSERT 3.9.7 - 1

3.9 REFUELING OPERATIONS

3.9.7 Containment Purge Filter System

LCO 3.9.7 The Containment Purge Filter System shall be OPERABLE and operating.

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

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. Containment Purge Filter System inoperable. OR Containment Purge Filter System not in operation.	A.1	Close each penetration providing direct access from the containment atmosphere to the outside atmosphere by a manual or automatic valve, blind flange, or equivalent method.	Immediately
	OR		
	A.2.1	Suspend CORE ALTERATIONS.	Immediately
	AND		
	A.2.2	Suspend movement of irradiated fuel assemblies within containment.	Immediately

ITS Insert 3.9.7-1 (Containment Purge Filter System)

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.9.7.1	Verify relative humidity of containment atmosphere to be processed by the Containment Purge Filter System is ≤ 70%.	1 hour
SR 3.9.7.2	Verify the Containment Purge Filter System is in operation and maintaining containment pressure negative relative to the adjacent auxiliary building areas.	12 hours
SR 3.9.7.3	Perform required Containment Purge Filter System filter testing in accordance with the Ventilation Filter Testing Program (VFTP).	In accordance with the VFTP

1 In the conversion of the HBRSEP current Technical Specifications (CTS) to the proposed plant specific Improved Technical Specifications (ITS), certain wording preferences or conventions are adopted which do not result in technical changes (either actual or interpretational). Editorial changes which involve the insertion of plant specific terms or parameters are used to preserve consistency with the CTS and licensing basis.

2 ISTS Specification 3.9.2, "Unborated Water Source Isolation Valves," is not applicable, because a boron dilution event has been analyzed in the UFSAR, Section 15.4.6, and the plant is considered to meet the applicable acceptance criteria, based on detection and termination prior to loss of shutdown margin. ITS Specification 3.9.3, "Nuclear Instrumentation," addresses the OPERABILITY requirements for the Source Range Neutron Flux instrumentation. Operability requirements include visual count rate indication in the control room and audible count rate indication inside containment, which is credited in the detection of a boron dilution event. Subsequent Specifications are renumbered accordingly.

3 In ITS LCO 3.9.3.b, the term, "each," is replaced by the term, "the," to reflect that the containment has only one air lock.

4 ITS Specifications 3.9.4 and 3.9.5 are modified by replacing the term "loop" with the term "train" when referring to the RHR System. The change from the ISTS term "loops" to "trains" in reference to the RHR System is done for consistency with the HBRSEP Unit No. 2 plant specific description of the RHR System as reflected in UFSAR Section 5.4.4; UFSAR Section 5.4.4 describes that there is one loop of the RHR System. Therefore, to avoid confusion with the description of the RHR System in the UFSAR, the term "train" is used in place of the term "loop" in the ITS. An RHR loop (as described in the ISTS Bases) consists of an RHR pump, a heat exchanger, valves, piping, and instruments and controls to ensure a flow path for decay heat removal is available. In the HBRSEP Unit No. 2 ITS, an RHR train consists of an RHR pump, a heat exchanger, valves, piping, and instruments and controls to ensure a flow path for decay heat removal is available.

5 ITS SR 3.9.5.1 is modified such that the RHR flow rate is not specified. It is necessary to have flexibility to control flow rate when the water level is ≥ 36 inches below the reactor vessel flange to avoid vortexing in the reactor vessel. In addition, the HBRSEP Unit No. 2 current licensing basis regarding RHR System Technical Specifications, approved in Amendment 64 dated March 8, 1982, does not include this requirement.

6 ISTS 3.9.7 (ITS 3.9.6) Required Action A.3 is deleted. Completion of Required Actions A.1 and A.2 result in exiting the MODE of Applicability. The deletion of ISTS 3.9.7 Required Action A.3 is consistent with generic change TSTF-20. TSTF-20 was approved by the NRC on March 13, 1997.

7 ITS Specification 3.9.4 contains a Note, permitting RCPs and RHR pumps to be de-energized for ≤1 hour per 8 hour period. This Note is modified by changing the phrase, "per 8 hour period," to "in any 8 hour period," to eliminate any interpretation that these pumps can be de-energized for consecutive 1 hour periods in two 8 hour periods.

8 ITS SR 3.9.4.1 is revised to delete the requirement for minimum RHR flow, consistent with ITS SR 3.4.8.1. There is no safety analysis that assumes a minimum RHR flow in this plant condition. In addition, the HBRSEP Unit No. 2 current licensing basis regarding RHR System Technical Specifications, approved in Amendment 64 dated March 8, 1982, does not include this requirement.

9 ITS 3.9.7, Containment Purge Filter System, is added to require the Containment Purge Filter System to be OPERABLE and operating during CORE ALTERATIONS and during movement of irradiated fuel in containment. ITS 3.9.7 encompasses the Containment Purge Filter System requirements in CTS 3.8.1.i, 3.8.2.c.2, and 3.8.2.d. This specification is necessary to ensure the assumptions of the fuel handling accident in containment (regarding the Containment Purge Filter System) are maintained. In order to reduce the consequences of a fuel handling accident in containment, the Containment Purge Filter System is assumed to be operating during the release and filtration of the activity is assumed for 5 minutes prior to isolation of the containment purge supply and exhaust penetrations. Filtration of the release is assumed due to the negative pressure maintained in the containment, relative to the outside atmosphere, by the Containment Purge Filter System.

BASES (continued)

SURVEILLANCE REQUIREMENTS

SR 3.9.~~8~~ 6.1

Verification of a minimum water level of 23 ft above the top of the reactor vessel flange ensures that the design basis for the analysis of the postulated fuel handling accident during refueling operations is met. Water at the required level above the top of the reactor vessel flange limits the consequences of damaged fuel rods that are postulated to result from a fuel handling accident inside containment (Ref. ~~8~~ 1).

The Frequency of 24 hours is based on engineering judgment and is considered adequate in view of the large volume of water and the normal procedural controls of valve positions, which make significant unplanned level changes unlikely.

REFERENCES

~~1. Regulatory Guide 1.25, March 23, 1972~~ 16

1 ~~2.~~ FSAR, Section ~~15.4.5~~ 15.7.4

~~3. NUREG-0800, Section 15.7.4~~ 10

2 ~~4.~~ 10 CFR 100.10.

3 ~~5.~~ Malinowski, D. D., Bell, M. J., Duhn, E., and Locante, J., WCAP-828, Radiological Consequences of a Fuel Handling Accident, December 1971.

INSERT B.3.9.7.1 - 1 11

B 3.9 REFUELING OPERATIONS

B 3.9.7 Containment Purge Filter System

BASES

BACKGROUND

The Containment Purge Filter System filters airborne radioactivity released to the containment atmosphere following a fuel handling accident in the containment. During refueling outages, the Containment Purge Filter System, in conjunction with other normally operating systems, also provides environmental control of temperature and humidity in the containment.

The Containment Purge Filter System is a single train system which consists of a prefilter, a high efficiency particulate air (HEPA) filter, an activated charcoal adsorber section for removal of gaseous activity (principally iodines), and two fans (only one of the fans is required, the second fan is a spare). Ductwork, valves or dampers, and instrumentation also form part of the system.

The Containment Purge Filter System is a manually intitiated system, which may also be operated during normal plant operations.

The Containment Purge Filter System is discussed in the UFSAR, Sections 6.5.1, 9.4.3, and 15.7.4 (Refs. 1, 2, and 3, respectively) because it may be used for normal, as well as post accident, atmospheric cleanup functions.

APPLICABLE SAFETY ANALYSES

The Containment Purge Filter System design basis is established by the consequences of the limiting Design Basis Accident (DBA), which is a fuel handling accident in the containment. The analysis of the fuel handling accident, given in Reference 3, assumes that all fuel rods in an assembly are damaged and the fission product inventory in the gap is released. The Containment Purge Filter System is assumed to be operating during the release and filtration of the activity is assumed for 5 minutes prior to isolation of the containment purge supply and exhaust penetrations. Filtration of the release is assumed due to the negative pressure maintained in the containment, relative to the outside atmosphere, by the Containment Purge Filter System.

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

The integrated dose is calculated using assumptions in Reference 3, which are consistent with the methodology utilized throughout the UFSAR, Chapter 15.

The Containment Purge Filter System satisfies Criterion 3 of the NRC Policy Statement.

LCO

The Containment Purge Filter System is required to be OPERABLE and operating. When the Containment Purge Filter System is in operation, the exhaust flow from containment shall discharge through the HEPA and impregnated charcoal filters. Total system failure could result in the atmospheric release from the containment exceeding the 10 CFR 100 (Ref. 4) limits in the event of a fuel handling accident.

The Containment Purge Filter System is considered OPERABLE when the individual components necessary to control exposure in the containment are OPERABLE. The Containment Purge Filter System is considered OPERABLE when:

a. One fan is OPERABLE;

b. HEPA filter and charcoal adsorber are not excessively restricting flow, and are capable of performing their filtration function; and

c. Ductwork, valves, and dampers are OPERABLE, and air flow can be maintained.

APPLICABILITY

During CORE ALTERATIONS and during movement of irradiated fuel in the containment, the Containment Purge Filter System is required to be OPERABLE and operating to alleviate the consequences of a fuel handling accident.

ACTIONS

A.1

When the Containment Purge Filter System is inoperable or not in operation during CORE ALTERATIONS or during movement

(continued)

BASES

ACTIONS

A.1 (continued)

of irradiated fuel assemblies in containment, Required Action A.1 requires each penetration providing direct access from the containment atmosphere to the outside atmosphere to be immediately closed. Closure may be achieved by a closed manual or automatic valve, blind flange, or equivalent method. Equivalent closure methods must be approved and may include use of a material that can provide a temporary atmospheric pressure, ventilation barrier for the penetration during fuel movements. Alternately, Required Actions A.2.1 and A.2.2 may be taken to place the unit in a condition in which the LCO does not apply. Required Actions A.2.1 and A.2.2 require immediate suspension of CORE ALTERATIONS and movement of irradiated fuel assemblies in containment. Suspension of these activities does not preclude the movement of fuel to a safe position.

SURVEILLANCE REQUIREMENTS

SR 3.9.7.1

This SR verifies that the relative humidity of the containment atmosphere to be processed by the Containment Purge Filter System is ≤ 70%. This ensures that the testing performed to validate the safety analysis assumptions relative to charcoal filter efficiency, bounds actual plant conditions for relative humidity at the inlet of the Containment Purge Filter System charcoal filter. The one hour Frequency is based on engineering judgment considering the likelihood of changes in containment relative humidity during refueling outages.

SR 3.9.7.2

This SR verifies that the Containment Purge Filter System is in operation and maintaining containment pressure negative relative to the adjacent auxiliary building areas once every 12 hours. This verification ensures that containment pressure is being maintained negative with respect to the outside atmosphere since the pressure of the auxiliary building areas is normally maintained negative with respect to the outside atmosphere. The Containment Purge Filter

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.9.7.2 (continued)

System is assumed to maintain a slight negative pressure in the containment, to prevent unfiltered leakage to the outside atmosphere. The Frequency of 12 hours is sufficient considering other indications available to the operator to monitor Containment Purge Filter System operation.

SR 3.9.7.3

This SR verifies that the required Containment Purge Filter System filter testing is performed in accordance with the Ventilation Filter Testing Program (VFTP). The VFTP includes testing HEPA filter performance, charcoal adsorber efficiency, system flow rate, and the physical properties of the activated charcoal (general use and following specific operations). Specific test frequencies and additional information are discussed in detail in the VFTP.

REFERENCES

1. UFSAR, Section 6.5.1.
2. UFSAR, Section 9.4.3.
3. UFSAR, Section 15.7.4.
4. 10 CFR 100.

1 In the conversion of the HBRSEP current Technical Specifications (CTS) to the proposed plant specific Improved Technical Specifications (ITS), certain wording preferences or conventions are adopted which do not result in technical changes (either actual or interpretational). Editorial changes or clarifications which involve the insertion of plant specific terms, parameters, or descriptions are used to preserve consistency with the CTS and licensing basis.

2 Bases for ITS 3.9.1 are modified to reflect 6% Δk/k refueling shutdown margin, which is current licensing basis.

3 HBRSEP was designed and licensed to the proposed Appendix A to 10 CFR 50, which was published in the Federal Register on July 11, 1967 (32FR10213). Appendix A to 10 CFR 50, which became effective in 1971, and was subsequently amended, is somewhat different from the proposed 1967 criteria. UFSAR section 3.1 includes an evaluation of HBRSEP with respect to the proposed 1967 criteria. ISTS statements concerning the general design criteria are modified in the ITS to reference the current licensing basis description in UFSAR Section 3.1.

4 Bases for ITS 3.9.1 are modified to reflect that refueling canal and refueling cavity cannot be flooded by gravity feed; and that safety injection pumps are normally used to flood up.

5 ISTS Specification 3.1.2 is not retained as a separate specification in the ITS. Since the specific shutdown margin requirements for various plant conditions are relocated to the Core Operating Limits Report (COLR), there is no need for separate specifications. Consequently, shutdown margin requirements applicable to MODE 5 are included in ITS Specification 3.1.1.

6 ISTS Specification 3.9.2, "Unborated Water Source Isolation Valves," is not applicable, because a boron dilution event has been analyzed in the UFSAR, and the plant is considered to meet the applicable acceptance criteria, based on detection and termination prior to loss of shutdown margin. ITS Specification 3.9.3, "Nuclear Instrumentation," addresses the OPERABILITY requirements for the Source Range Neutron Flux instrumentation. Operability requirements include visual count rate indication in the control room and audible count rate indication inside containment, which is credited in the detection of a dilution event. Subsequent Specifications are renumbered accordingly.

7 Bases for ITS 3.9.3 are modified to reflect that the containment has only one air lock.

8 Bases for ITS 3.9.3 are modified to reflect that ESFAS is disabled when in MODE 6. Containment isolation functions are taken from the Containment Isolation System.

9 Bases for ITS 3.9.3 is modified to reflect reference to GPU Nuclear safety evaluation is not needed. References renumbered accordingly.

10 HBRSEP is not a "Standard Review Plan" plant. Therefore, references to NUREG-0800 are deleted.

11 The Bases are revised to be consistent with the changes made to the Specifications.

12 Bases for ITS 3.9.4 and 3.9.5 are modified by deleting phrase, "and to determine the low end temperature," as it is not related to OPERABILITY of an RHR train.

13 Bases for ITS 3.9.5 are modified to allow both RHR pumps to be aligned to the RWST to fill the refueling cavity or to perform the RHR full flow test. This is necessary, as the LCO requires both RHR trains to be OPERABLE, and OPERABLE is described in the Bases as a flow path from the RCS hot leg, through the RHR pump and RHR heat exchanger, to the RCS cold leg. However, in order to fill the refueling cavity in preparation for refueling, the suction of the RHR pumps is aligned to the RWST and the water is pumped into the refueling cavity through the RCS hot legs. A similar situation occurs during the RHR full flow test when both pumps are aligned to the RWST and pump water into the core. This change to the Bases acknowledges these operational conditions.

14 Bases for ITS 3.9.5, Required Action B.3, are modified to reflect that the completion time to close all penetrations is reasonable, based on operating experience, rather than the low probability of the coolant boiling in that time.

15 Bases for ITS SR 3.9.5.1 are modified such that the RHR flow rate is not specified. It is necessary to have flexibility to control flow rate when the water level is ≥ 36 inches below the reactor vessel flange to avoid vortexing in the reactor vessel.

16 HBRSEP is not committed to Regulatory Guide 1.25, and therefore references to the Regulatory Guide are deleted in the ITS.

17 Bases for ITS 3.9.6, Required Action A.3, is deleted. Completion of Required Actions A.1 and A.2 result in exiting the MODE of Applicability

18 Bases for ITS SR 3.9.5.2 are modified to delete reference to an RCS pump. The SR requires verification of OPERABILITY of an RHR pump.

3.9 REFUELING OPERATIONS

3.9.7 Containment Purge Filter System

LCO 3.9.7 The Containment Purge Filter System shall be OPERABLE and operating.

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

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Containment Purge Filter System inoperable. OR Containment Purge Filter System not in operation.	A.1 Close each penetration providing direct access from the containment atmosphere to the outside atmosphere by a manual or automatic valve, blind flange, or equivalent method.	Immediately
	OR	
	A.2.1 Suspend CORE ALTERATIONS.	Immediately
	AND	
	A.2.2 Suspend movement of irradiated fuel assemblies within containment.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.9.7.1	Verify relative humidity of containment atmosphere to be processed by the Containment Purge Filter System is ≤ 70%.	1 hour
SR 3.9.7.2	Verify the Containment Purge Filter System is in operation and maintaining containment pressure negative relative to the adjacent auxiliary building areas.	12 hours
SR 3.9.7.3	Perform required Containment Purge Filter System filter testing in accordance with the Ventilation Filter Testing Program (VFTP).	In accordance with the VFTP

B 3.9 REFUELING OPERATIONS

B 3.9.7 Containment Purge Filter System

BASES

BACKGROUND

The Containment Purge Filter System filters airborne radioactivity released to the containment atmosphere following a fuel handling accident in the containment. During refueling outages, the Containment Purge Filter System, in conjunction with other normally operating systems, also provides environmental control of temperature and humidity in the containment.

The Containment Purge Filter System is a single train system which consists of a prefilter, a high efficiency particulate air (HEPA) filter, an activated charcoal adsorber section for removal of gaseous activity (principally iodines), and two fans (only one of the fans is required, the second fan is a spare). Ductwork, valves or dampers, and instrumentation also form part of the system.

The Containment Purge Filter System is a manually intitiated system, which may also be operated during normal plant operations.

The Containment Purge Filter System is discussed in the UFSAR, Sections 6.5.1, 9.4.3, and 15.7.4 (Refs. 1, 2, and 3, respectively) because it may be used for normal, as well as post accident, atmospheric cleanup functions.

APPLICABLE SAFETY ANALYSES

The Containment Purge Filter System design basis is established by the consequences of the limiting Design Basis Accident (DBA), which is a fuel handling accident in the containment. The analysis of the fuel handling accident, given in Reference 3, assumes that all fuel rods in an assembly are damaged and the fission product inventory in the gap is released. The Containment Purge Filter System is assumed to be operating during the release and filtration of the activity is assumed for 5 minutes prior to isolation of the containment purge supply and exhaust penetrations. Filtration of the release is assumed due to the negative pressure maintained in the containment, relative to the outside atmosphere, by the Containment Purge Filter System.

(continued)

BASES

APPLICABLE SAFETY ANALYSES (continued)

The integrated dose is calculated using assumptions in Reference 3, which are consistent with the methodology utilized throughout the UFSAR, Chapter 15.

The Containment Purge Filter System satisfies Criterion 3 of the NRC Policy Statement.

LCO

The Containment Purge Filter System is required to be OPERABLE and operating. When the Containment Purge Filter System is in operation, the exhaust flow from containment shall discharge through the HEPA and impregnated charcoal filters. Total system failure could result in the atmospheric release from the containment exceeding the 10 CFR 100 (Ref. 4) limits in the event of a fuel handling accident.

The Containment Purge Filter System is considered OPERABLE when the individual components necessary to control exposure in the containment are OPERABLE. The Containment Purge Filter System is considered OPERABLE when:

a. One fan is OPERABLE;

b. HEPA filter and charcoal adsorber are not excessively restricting flow, and are capable of performing their filtration function; and

c. Ductwork, valves, and dampers are OPERABLE, and air flow can be maintained.

APPLICABILITY

During CORE ALTERATIONS and during movement of irradiated fuel in the containment, the Containment Purge Filter System is required to be OPERABLE and operating to alleviate the consequences of a fuel handling accident.

ACTIONS

A.1

When the Containment Purge Filter System is inoperable or not in operation during CORE ALTERATIONS or during movement

(continued)

BASES

ACTIONS

A.1 (continued)

of irradiated fuel assemblies in containment. Required Action A.1 requires each penetration providing direct access from the containment atmosphere to the outside atmosphere to be immediately closed. Closure may be achieved by a closed manual or automatic valve, blind flange, or equivalent method. Equivalent closure methods must be approved and may include use of a material that can provide a temporary atmospheric pressure, ventilation barrier for the penetration during fuel movements. Alternately, Required Actions A.2.1 and A.2.2 may be taken to place the unit in a condition in which the LCO does not apply. Required Actions A.2.1 and A.2.2 require immediate suspension of CORE ALTERATIONS and movement of irradiated fuel assemblies in containment. Suspension of these activities does not preclude the movement of fuel to a safe position.

SURVEILLANCE REQUIREMENTS

SR 3.9.7.1

This SR verifies that the relative humidity of the containment atmosphere to be processed by the Containment Purge Filter System is ≤ 70%. This ensures that the testing performed to validate the safety analysis assumptions relative to charcoal filter efficiency, bounds actual plant conditions for relative humidity at the inlet of the Containment Purge Filter System charcoal filter. The one hour Frequency is based on engineering judgment considering the likelihood of changes in containment relative humidity during refueling outages.

SR 3.9.7.2

This SR verifies that the Containment Purge Filter System is in operation and maintaining containment pressure negative relative to the adjacent auxiliary building areas once every 12 hours. This verification ensures that containment pressure is being maintained negative with respect to the outside atmosphere since the pressure of the auxiliary building areas is normally maintained negative with respect to the outside atmosphere. The Containment Purge Filter

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.9.7.2 (continued)

System is assumed to maintain a slight negative pressure in the containment, to prevent unfiltered leakage to the outside atmosphere. The Frequency of 12 hours is sufficient considering other indications available to the operator to monitor Containment Purge Filter System operation.

SR 3.9.7.3

This SR verifies that the required Containment Purge Filter System filter testing is performed in accordance with the Ventilation Filter Testing Program (VFTP). The VFTP includes testing HEPA filter performance, charcoal adsorber efficiency, system flow rate, and the physical properties of the activated charcoal (general use and following specific operations). Specific test frequencies and additional information are discussed in detail in the VFTP.

REFERENCES

1. UFSAR, Section 6.5.1.
2. UFSAR, Section 9.4.3.
3. UFSAR, Section 15.7.4.
4. 10 CFR 100.

SUPPLEMENT 5
CONVERSION PACKAGE SECTION 5.0
PAGE INSERTION INSTRUCTIONS

Remove and insert the following pages into Enclosure 19 to Serial: RNP-RA/96-0141.

	Remove Page	Insert Page
a.	Part 1, "Markup of Current Technical Specifications (CTS)"	
	4.2-6 & 4.1-10a	4.2-6 & 4.1-10a
b.	Part 2, "Discussion of Changes (DOCs) for CTS Markup"	
	6 & 7	6 & 7
	-	7a
	9b	9b
c.	Part 3, " No Significant Hazards Consideration (NSHC), And Basis for Categorical Exclusion from 10 CFR 51.22	
	NA	
d.	Part 4, "Markup of NUREG-4131, Revision 1, Standard Technical Specifications-Westinghouse Plants, (ISTS)"	
	5.0-10	5.0-10
e.	Part 5, "Justification of Differences (JFDs) to ISTS"	
	NA	
f.	Part 6, "Markup of ISTS Bases"	
	NA	
g.	Part 7, "Justification for Differences (JFDs) to ISTS Bases"	
	NA	
h.	Part 8, "Proposed HBRSEP, Unit No. 2 ITS"	
	5.0-11	5.0-11
i.	Part 9. "Proposed Bases to HBRSEP, Unit No. 2 ITS Bases"	
	NA	
j.	Part 10. "ISTS Generic Changes"	
	NA	

ITS

A1

Class 2 and Class 3 components were chosen based on Regulatory Guide 1.26 and ANSI N18.2 and N18.2a "Nuclear Safety Criteria for the Design of Stationary Pressurized Water Reactor Plants."

A17

The Surveillance Requirements for inspection of the steam generator tubes ensure that the structural integrity of this portion of the RCS will be maintained. The program for inservice inspection of steam generator tubes is based on a modification of Regulatory Guide 1.83, Revision 1. Inservice inspection of steam generator tubing is essential in order to maintain surveillance of the conditions of the tubes for evidence of mechanical damage or progressive degradation. Inservice inspection of steam generator tubing also provides a means of characterizing the nature and cause of any tube degradation so that corrective measures can be taken.

Wastage-type defects will be minimized with proper chemistry treatment of the secondary coolant. If defects or significant degradations should develop in service, this condition is expected to be detected during inservice steam generator tube examinations. Plugging will be required for all tubes with imperfections exceeding the plugging limit. Steam generator tube inspections by means of eddy current testing have demonstrated the capability to reliably detect degradation that has penetrated 20% of the original tube wall thickness.

Whenever the results of any steam generator tubing inservice inspection fall into Category C-3, these results will be reported to the Commission prior to resumption of plant operation. Such cases will be considered by the Commission on a case-by-case basis and may result in a requirement for analysis, laboratory examinations, tests, additional eddy-current inspection, and revision of the Technical Specifications.

4.2.2 Materials Irradiation Surveillance Specimens

A18

The reactor vessel material surveillance specimens shall be removed and examined to determine changes in their material properties, as required by Appendix H to 10CFR50.

4.2.3 Primary Pump Flywheels

[5.5.7]

LA8

The flywheels shall be visually examined at the first refueling after each ten year inspection. At the fourth refueling after each ten year inspection and at each fourth refueling thereafter, the outside surfaces shall be examined by ultrasonic methods. The examinations scheduled for Refueling Outage 17, in 1996, may be deferred to Refueling Outage 18.

A32

This program provides controls for the inspection of each reactor coolant pump flywheel in accordance with the Inservice Inspection Program.

ITS

Add general program statement for ITS 5.5.13 (A38)

(A1)

TABLE 4.1.2 (Continued)

FREQUENCIES FOR SAMPLING TESTS

		Check	Frequency	Maximum Time Between Tests	(M15)
[5.5.13.b]	11. U-2 Diesel Generator Fuel Oil Storage Tank	API or Specific Gravity, Water and Sediment, Viscosity and cloud point (M15)	Monthly	38.75 days per SR 3.0.2 ~~45 days~~	
[5.5.13.b]	12. U-1 I-C Turbine Fuel Oil Storage Tanks or Tank Truck	API or Specific Gravity, Water and Sediment, Viscosity	Prior to transfer to U-2 (31 days) (M15)	N/A	

Add 5.5.13.a (M15)

The provisions of SR 3.0.2 and SR 3.0.3 are applicable to the Diesel Fuel Oil Testing Program surveillance frequencies. (A38)

Supplement 5

Specification 5.5

A28 CTS Specification 6.13.2, which requires that locked doors be provided to prevent unauthorized entry into high radiation areas, is revised in ITS Specification 5.7.2, "High Radiation Area." The term, "unauthorized," is replaced with the term, "inadvertent," to clarify that the purpose of locked doors in high radiation is to prevent inadvertent entry for personnel safety, rather than to prevent unauthorized entry from a security perspective. This change is administrative, and has no adverse impact on safety.

A29 CTS Specifications 6.6.1.a and 6.6.2.a, which contain requirements regarding notification and submittal of reports to the NRC pursuant to the requirements of 10 CFR 50.72 and 10 CFR 50.73, are not retained in the ITS. These reporting requirements are specified within the cited regulations and need not be repeated in the ITS. This change is administrative, and has no adverse impact on safety.

A30 CTS Specification 6.2.3.a is revised in ITS Specification 5.2.2 to incorporate the current plant practice regarding the function of the shift technical advisor. Since no technical requirements are modified, this change is administrative, and has no adverse impact on safety.

A31 CTS Specification 6.9.1.2.1, which specifies occupational radiation exposure reporting requirements, is modified in ITS Specification 5.6.1, where examples of work and job functions identified as, "routine maintenance, special maintenance [describe maintenance]," are condensed to simply read, "maintenance," to be consistent with other examples given. This change is administrative, and has no adverse impact on safety.

A32 CTS Specifications 4.2.3 (Primary Pump Flywheels) and 6.12 (Containment Leakage Rate Testing Program), are revised in ITS Specifications 5.5.7 and 5.5.16, respectively to modify the presentation of text to be consistent with the presentation of purpose statements of other programs in this Chapter. In addition, for the reactor coolant pump flywheel inspection requirements, a general program statement has been added consistent with current plant practice which includes reactor coolant pump flywheel inspection requirements in the Inservice Inspection Program. Since no technical requirements are modified, this change is administrative and has no adverse impact on safety.

A33 The CTS is revised to adopt ISTS Specification 5.5.12, "Explosive Gas and Storage Tank Radioactivity Monitoring Program," in the ITS. This program captures the existing requirements for explosive gas and storage tank radioactivity contained in CTS Specifications 3.16.2, 3.16.4, 3.16.5,4.20.2, 4.20.4, and 4.20.5. Consequently, the adoption of this program is an administrative change, and is consistent with NUREG-1431.

A34 CTS 4.4.4.3.c requires submitting a report and evaluation of a Containment Tendon Test within 6 months of completing the test. CTS

4.4.4.1 requires performance of a containment tendon inspection after 5 years of operation and 25 years of operation. CTS 6.9.3 refers to submittal of the Containment Tendon Surveillance Report upon completion of the inspection at 25 years of operation. (The CTS 6.9.3 reference reporting the inspection upon completion of the tendon test at 25 years of operation is an imprecise restatement of the actual test requirement specified in CTS 4.4.4.1 and reporting requirement specified in 4.4.4.3.c.) ITS 5.6.7 requires reporting containment tendon test results within 6 months after completion of the test. This is consistent with the current licensing basis for HBRSEP Unit No. 2 and the comparable STS requirement which also does not require performance of the test at any specific time (i.e., after 25 years of operation). CTS 4.4.4.1 which includes schedular requirements for performance of the test is relocated to the Pre-Stressed Concrete Containment Tendon Surveillance Program. Relocation of test schedular requirements to the Pre-Stressed Concrete Containment Tendon Surveillance Program is consistent with ISTS 5.5.6 and ITS 5.5.6.

A35 CTS 6.5.1 is a general statement of objectives which is actually implemented by the more prescriptive subsections 6.5.1.1, "Procedures, Tests and Experiments"; and 6.5.1.2, "Modifications"; 6.5.1.3, "Technical Specification and License Changes, 6.5.1.4, "Review of Technical Specification Violations"; 6.5.1.5, Nuclear Safety Review Qualification"; and 6.5.1.6, "Plant Nuclear Safety Committee (PNSC)." This statement is not separately retained in the ITS since actual implementation is accomplished in accrodance with the more prescriptive subsections listed above. Therefore, deletion of this general statement is considered administrative, and has no impact on safety.

A36 The personnel exposure and monitoring reporting requirements of CTS 6.9.1.2.1 are modified in ITS 5.6.1 to reflect the revised 10 CFR 20 requirements. The revised 10 CFR 20 requirements are currently applicable to HBRSEP Unit No. 2. As a result, the change is considered to be administrative in nature in order to make the existing wording of CTS 6.9.1.2.1 consistent with the wording of the revised 10 CFR 20.

A37 CTS Specification 6.2.1.e, CTS Specification 6.2.3.b, and the *Note to CTS Specification 6.13.1, refer to requirements related to "health physics" individuals. ITS Specification 5.2.1.d, ITS Specification 5.2.2.e, and ITS Specificaiton 5.7.1 modifies this generic title to "radiation control" individuals. This change is being made to be consistent with the generic title used in the HBRSEP, Unit No. 2 UFSAR. Per UFSAR Section 12.1, "radiation control" individuals implement health physics programs and principles. Therefore, this change is considered administrative, and has no adverse impact on safety.

A38 The diesel fuel oil testing requirements (CTS Table 4.1.2) are placed in a program in the Administrative Controls Chapter (ITS Specification 5.5.13 (Diesel Fuel Oil Testing Program)). As such, a

general program statement is added. Also, a statement of applicability of SR 3.0.2 and SR 3.0.3 is needed to clarify that the allowances for surveillance frequency extensions do apply, since these SRs are not normally applied to frequencies identified in the Administrative Controls Chapter of the Technical Specifications. Since this change is a clarification needed to maintain provisions that would be allowed in the LCO sections of the Technical Specifications, it is considered administrative.

provided by the diesel fuel supplier for each new fuel oil delivery. The certificate of compliance includes certification of each of the new fuel oil properties included in ISTS 5.5.13.a. Therefore, to maintain consistency with this current practice an additional requirement is provided. HBRSEP ITS 5.5.13 (Diesel Fuel Oil Testing Program) states the purpose of the program is to establish the acceptability of new fuel oil for use prior to addition to the storage tanks by determining that the new oil has not become contaminated with other products during transit, thus altering the quality of the fuel oil. This is an additional restriction upon unit operation to assure the quality of the fuel oil is maintained.

In addition, CTS items 11 and 12 of Table 4.1.2 require testing of stored diesel fuel oil monthly with a maximum time between tests of 45 days (for the Unit 2 diesel fuel oil storage tank) and prior to transfer to Unit 2 (for the Unit 1 I-C turbine fuel oil storage tanks or tank truck). The CTS requires tests for API or specific gravity, water and sediment, and viscosity. ITS 5.5.13.b requires the same tests to be performed at a 31-day frequency for the required Unit 1 and the Unit 2 diesel fuel oil storage tanks and adds a requirement for cloud point testing to be performed. The ITS 5.5.13.b frequency requirement may only be extended by 25% in accordance with SR 3.0.2 (for a maximum time between tests of 38.75 days versus the CTS maximum time of 45 days). These changes represent additional restrictions on plant operation necessary to ensure that fuel oil quality is maintained and to achieve consistency with NUREG-1431 philosophy regarding Surveillance Frequency extensions.

CTS

5.5 Programs and Manuals

(1)

[A15] 5.5.4 Radioactive Effluent Controls Program (continued)

i. Limitations on the annual and quarterly doses to a member of the public from iodine-131, iodine-133, tritium, and all radionuclides in particulate form with half lives > 8 days in gaseous effluents released ~~from each unit~~ to areas beyond the site boundary, conforming to 10 CFR 50, Appendix I; and (4)

j. Limitations on the annual dose or dose commitment to any member of the public due to releases of radioactivity and to radiation from uranium fuel cycle sources, conforming to 40 CFR 190.

[M5] 5.5.5 Component Cyclic or Transient Limit

This program provides controls to track the ~~FSAR, Section 5.5~~ UFSAR Table 3.9.1-1 cyclic and transient occurrences to ensure that components are maintained within the design limits.

(24)

[4.4.4.1] [4.4.4.3.a] 5.5.6 Pre-Stressed Concrete Containment Tendon Surveillance Program

This program provides controls for monitoring any tendon degradation in pre-stressed concrete containments, including effectiveness of its corrosion protection medium, to ensure containment structural integrity. The program shall include ~~baseline measurements prior to initial operations. The Tendon Surveillance Program, inspection frequencies, and acceptance criteria shall be in accordance with Regulatory Guide 1.35, Revision 3, 1989.~~

The provisions of SR 3.0.2 and SR 3.0.3 are applicable to the Tendon Surveillance Program inspection frequencies.

[4.2.3] 5.5.7 Reactor Coolant Pump Flywheel Inspection Program

This program ~~shall provide~~ provides controls for the inspection of each reactor coolant pump flywheel ~~per the recommendations of Regulatory Position C.4.b of Regulatory Guide 1.14, Revision 1, August 1975~~ in accordance with the Inservice Inspection Program (9) (12)

(continued)

Supplement 5

5.5 Programs and Manuals (continued)

5.5.7 Reactor Coolant Pump Flywheel Inspection Program

This program provides controls for the inspection of each reactor coolant pump flywheel in accordance with the Inservice Inspection Program.

5.5.8 Inservice Testing Program

This program provides controls for inservice testing of ASME Code Class 1, 2, and 3 pumps and valves. The program shall include the following:

a. Testing frequencies specified in Section XI of the ASME Boiler and Pressure Vessel Code and applicable Addenda as follows:

ASME Boiler and Pressure Vessel Code and applicable Addenda terminology for inservice testing activities	Required Frequencies for performing inservice testing activities
Weekly	At least once per 7 days
Monthly	At least once per 31 days
Quarterly or every 3 months	At least once per 92 days
Semiannually or every 6 months	At least once per 184 days
Every 9 months	At least once per 276 days
Yearly or annually	At least once per 366 days
Biennially or every 2 years	At least once per 731 days

b. The provisions of SR 3.0.2 are applicable to the above required Frequencies for performing inservice testing activities;

c. The provisions of SR 3.0.3 are applicable to inservice testing activities; and

d. Nothing in the ASME Boiler and Pressure Vessel Code shall be construed to supersede the requirements of any TS.

(continued)

Page Insertion Instruction for inserting pages into Enclosure 21 to Serial: RNP-RA/96-0141, "Compilation of CTS Pages."

Remove Page	Insert Page
3.1-11 (ITS 3.1.8)	3.1-11 (ITS 3.1.8)
3.3-1 (ITS 3.5.3)	3.3.1 (ITS 3.5.3)
3.3-2 (ITS 3.5.2)	3.3-2 (ITS 3.5.2)
3.3-2 (ITS 3.5.3)	3.3-2 (ITS 3.5.3)
3.3-3 (ITS 3.5.1)	3.3-3 (ITS 3.5.1)
3.3-3 (ITS 3.5.2)	3.3-3 (ITS 3.5.2)
3.3-4 (ITS 3.5.1)	3.3-4 (ITS 3.5.1)
3.3-4 (ITS 3.5.2)	3.3-4 (ITS 3.5.2)
3.4-1 (ITS 3.7.1)	3.4-1 (ITS 3.7.1)
3.4-2 (ITS 3.7.1)	3.4-2 (ITS 3.7.1)
3.5-14 (ITS 3.3.2)	3.5-14 (ITS 3.3.2)
3.5-15a (ITS 3.3.2)	3.5-15a (ITS 3.3.2)
3.5-17 (ITS 3.3.2)	3.5-17 (ITS 3.3.2)
3.8-1 (ITS 3.9.2)	3.8-1 (ITS 3.9.2)
3.8-1 (ITS 3.9.3)	3.8-1 (ITS 3.9.3)
3.8-3 (ITS 3.9.3)	3.8-3 (ITS 3.9.3)
3.8-2 (ITS 3.9.7)	3.8-2 (ITS 3.9.7)
3.8-3 (ITS 3.9.7)	3.8-3 (ITS 3.9.7)
3.10-1 (ITS 3.1.8)	3.10-1 (ITS 3.1.8)
3.10-2 (ITS 3.1.4)	3.10-2 (ITS 3.1.4)
3.10-2 (ITS 3.2.1)	3.10-2 (ITS 3.2.1)
3.10-3 (ITS 3.2.1)	3.10-3 (ITS 3.2.1)
3.10-3 (ITS 3.2.2)	3.10-3 (ITS 3.2.2)
3.10-5a (ITS 3.2.2)	3.10-5a (ITS 3.2.2)
3.10-7a (ITS 3.2.3)	3.10-7a (ITS 3.2.3)
3.10-8 (ITS 3.1.4)	3.10-8 (ITS 3.1.4)
4.1-8 (ITS 3.3.3)	4.1-8 (ITS 3.3.3)
4.1-10a (ITS 3.8.3)	4.1-10a (ITS 3.8.3)
4.1-10a (ITS 5.5)	4.1-10a (ITS 5.5)
4.1-11 (ITS 3.4.16)	4.1-11 (ITS 3.4.16)
4.1-14 (ITS 3.4.14)	4.1-14 (ITS 3.4.14)
4.2-6 (ITS 5.5)	4.2-6 (ITS 5.5)
4.6-3 (ITS 3.8.4)	4.6-3 (ITS 3.8.4)
4.9-1 (ITS 3.1.2)	4.9-1 (ITS 3.1.2)
4.8-1 (ITS 3.7.4)	4.8-1 (ITS 3.7.4)

3.1.3 Minimum Conditions for Criticality

3.1.3.1 Except during low power physics tests, the reactor shall not be made critical at any temperature, at which the moderator temperature coefficient is outside the limits specified in the CORE OPERATING LIMITS REPORT (COLR). The maximum upper limits shall be less than or equal to:

a) +5.0 pcm/°F at less than 50% of rated power, or

b) 0 pcm/°F at 50% of rated power and above.

See 3.1.3

3.1.3.2 In no case shall the reactor be made critical above and to the left of the criticality limit shown on Figure 3.1-1.

See 3.4.2

3.1.3.3 When the reactor coolant temperature is in a range where the moderator temperature coefficient is outside the limits specified in the COLR, the reactor shall be made subcritical by an amount equal to or greater than the potential reactivity insertion due to depressurization.

See 3.1.3

3.1.3.4 The reactor shall be maintained subcritical by at least 1% until normal water level is established in the pressurizer.

See 3.4.9

Basis

During the early part of fuel cycle, the moderator temperature coefficient may be slightly positive at low power levels. The moderator temperature coefficient at low temperatures or powers will be most positive at the beginning of the fuel cycle, when the boron concentration in the coolant is the greatest. At all times, the moderator temperature coefficient is calculated to be negative in the high power operating range, and after a very brief period of power operation, the coefficient will be negative in all circumstances due to the reduced boron concentration as xenon and fission products build into the core. The requirement that the reactor is not to be made critical when the moderator temperature coefficient outside the limits specified in the COLR has been imposed to prevent any unexpected power excursion during normal operations as a result of either an increase in moderator temperature or decrease in coolant pressure. This requirement is

A7

M22

Add Applicability: MODE 2 during PHYSICS TESTS

Specification 3.5.3

ITS

A1

3.3 EMERGENCY CORE COOLING SYSTEM, AUXILIARY COOLING SYSTEMS, AIR RECIRCULATION FAN COOLERS, CONTAINMENT SPRAY, POST ACCIDENT CONTAINMENT VENTING SYSTEM, AND ISOLATION SEAL WATER SYSTEM

Applicability

Applies to the operating status of the Emergency Core Cooling System, Auxiliary Cooling Systems, Air Recirculation Fan Coolers, Containment Spray, Post Accident Containment Venting System, and Isolation Seal Water System.

Objective

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

A4

Specification

[Applicability]

3.3.1 Safety Injection and Residual Heat Removal Systems

IN MODE 4

3.3.1.1 The reactor shall not be made critical unless the following conditions are met:

a. The refueling water tank contains not less than 300,000 gallons of water with a boron concentration of at least 1950 ppm.

See 3.5.4

Add LCO Note

A12

M11

Add SR 3.5.3.1

Amendment No. 106

Supplements

ITS

Specification 3.5.

A1

b. Each accumulator is pressurized to at least 600 psig and contains at least 825 ft^3 and no more than 841 ft^3 of water with a boron concentration of at least 1950 ppm. No accumulator may be isolated. — see 3.5.1

[LCO 3.5.2] c. Two ~~safety injections~~ pumps are operable, ~~each capable of automatic initiation from a separate emergency bus.~~ ECCS trains shall be — A2

d. ~~Two residual heat removal pumps are operable.~~

e. ~~Two residual heat exchangers are operable.~~

f. ~~All essential features including valves, interlocks, and piping associated with the above components are operable.~~ — EA2

[SR 3.5.2.1] g. During conditions of operation with reactor coolant pressure in excess of 1000 psig the A.C. control power shall be removed from the following motor operated valves with the valve in the specified position: — M22

Valves	Position
MOV 862 A&B	Open
MOV 864 A&B	Open
MOV 865 A,B,&C	Open — see 3.5.1
MOV 878 A&B	Open
MOV 863 A&B	Closed
MOV 866 A&B	Closed

M22

[SR 3.5.2.7] h. During conditions of operation with reactor coolant pressure in excess of 1000 psig, the air supply to air operated valves 605 and 758 shall be shut off with valves in the closed position. — M22

Add SR 3.5.2.3 — A6

Add SR 3.5.2.6 — M11

Add SR 3.5.2.8 — M12

Supplements

ITS

Specification 3.5.3

A1

L6

ONE

b. Each accumulator is pressurized to at least 600 psig and contains at least 825 ft^3 and no more than 841 ft^3 of water with a boron concentration of at least 1950 ppm. No accumulator may be isolated. — See 3.5.1

[LCO 3.5.3] c. ~~Two~~ ECCS train shall be ~~safety injections pumps are~~ operable, ~~each capable of automatic initiation from a separate emergency bus.~~ — A2

d. ~~Two residual heat removal pumps are operable.~~

e. ~~Two residual heat exchangers are operable.~~

f. ~~All essential features including valves, interlocks, and piping associated with the above components are operable.~~

LA2

g. During conditions of operation with reactor coolant pressure in excess of 1000 psig the A.C. control power shall be removed from the following motor operated valves with the valve in the specified position: — See 3.5.2

Valves	Position
MOV 862 A&B	Open
~~MOV 864 A&B~~	~~Open~~
~~MOV 865 A,B,&C~~	~~Open~~ — See 3.5.1
MOV 878 A&B	Open
MOV 863 A&B	Closed
MOV 866 A&B	Closed — see 3.5.2

h. During conditions of operation with reactor coolant pressure in excess of 1000 psig, the air supply to air operated valves 605 and 758 shall be shut off with valves in the closed position. — See 3.5.2

Supplement 5

Specification 3.5.1

ITS

A1

See 3.4.4

A4

i. Power operation with less than three loops in service is prohibited.

3.3.1.2 During power operation, the requirements of 3.3.1.1 may be modified to allow any one of the following components to be inoperable. If the system is not restored to meet the requirements of 3.3.1.1 within the time period specified, the reactor shall be placed in the hot shutdown condition utilizing normal operating procedures. If the requirements of 3.3.1.1 are not satisfied within an additional 48 hours, the reactor shall be placed in the cold shutdown condition utilizing normal operating procedures.

Require Action and associated completion times of Condition A or B not met

M4

Mode 3 within 6 hours

[RA D.1]

[RA.D.2]

12

M5

Reduce pressurizer pressure to ≤ 1000 psig

M5

a. One accumulator may be isolated or otherwise inoperable relative to the requirements of 3.3.1.1.b for a period not to exceed four hours.

[RA C.1]

for reasons other than Condition A

b. If one safety injection pump becomes inoperable during normal reactor operation, the reactor may remain in operation for a period not to exceed 24 hours.

c. If one residual heat removal pump becomes inoperable during normal reactor operation, the reactor may remain in operation for a period not to exceed 24 hours.

See 3.5.2

Add RA A.1

L1

Add RA E.1

M7

Amendment No. ~~97~~, ~~115~~, ~~119~~, ~~146~~, 153

Supplement 5

Specification 3.5.2

A1

ITS

see 3.4

i. Power operation with less than three loops in service is prohibited

3.3.1.2 During power operation, the requirements of 3.3.1.1 may be modified to allow any one of the following components to be inoperable. If the system is not restored to meet the requirements of 3.3.1.1 within the time period specified, the reactor shall be placed in the hot shutdown condition utilizing normal operating procedures. If the requirements of 3.3.1.1 are not satisfied within an additional 48 hours, the reactor shall be placed in the cold shutdown condition utilizing normal operating procedures.

[RA A.1]

[RA C.1]

[RA C.2]

Required Action and associated completion time are not met

Mode 3 within 6 hours and Mode 4 within 12 hours

a. One accumulator may be isolated or otherwise inoperable relative to the requirements of 3.3.1.1.b for a period not to exceed four hours.

See 3.5.1

L2

b. If one safety injection pump becomes inoperable during normal reactor operation, the reactor may remain in operation for a period not to exceed 24 hours.

c. If one residual heat removal pump becomes inoperable during normal reactor operation, the reactor may remain in operation for a period not to exceed 24 hours.

L2

with one or more train inoperable and at least 100% of the ECCS flow equivalent to a single OPERABLE train available, Restore train(s) to OPERABLE status within 72 hours

L2

Amendment No. ~~97~~, ~~115~~, ~~119~~, ~~146~~, 153

Supplements

Specification 3.5.1

A 1

d. If one residual heat exchanger becomes inoperable during normal reactor operation, the reactor may remain in operation for a period not to exceed 24 hours.

e. If any one flow path including valves of the safety injection or residual heat removal system is found to be inoperable during normal reactor operation, the reactor may remain in operation for a period not to exceed 24 hours. The hot leg injection paths of the Safety Injection System, including valves, are not subject to the requirements of this specification.

See 3.5.2 and 3.5.3

~~f. Power or air supply may be restored to any valve referenced in 3.3.1.1.g. and 3.3.1.1.h. for the purpose of valve testing or maintenance providing no more than one valve has power restored and provided that testing and maintenance is completed and power removed within 24 hours except for accumulator isolation valves (MOV 865 A,B,&C) which will have this time period limited to four hours.~~

A10

Add RA B.1
B.2

Supplement 5

ITS

Specification 3.5.2

A1

d. If one residual heat exchanger becomes inoperable during normal reactor operation, the reactor may remain in operation for a period not to exceed 24 hours.

L2

e. If any one flow path including valves of the safety injection or residual heat removal system is found to be inoperable during normal reactor operation, the reactor may remain in operation for a period not to exceed 24 hours. The hot leg injection paths of the Safety Injection System including valves, are not subject to the requirements of this specification.

LA1

f. Power or air supply may be restored to any valve referenced in 3.3.1.1.g. and 3.3.1.1.h. for the purpose of valve testing or maintenance providing no more than one valve has power restored and provided that testing and maintenance is completed and power removed within 24 hours except for accumulator isolation valves (MOV 865 A,B,&C) which will have this time period limited to four hours.

A10

See 3.5.1

Add RA B.1 B.2

A10

Add Applicability Note 1

m23

Add Applicability Note 2

L8

Supplement 5

ITS

A1

3.4 SECONDARY STEAM AND POWER CONVERSION SYSTEM

Applicability

Applies to the operating status of turbine cycle.

Objective

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

Specification

A2

MODES 1, 2, 3

[Applicability] 3.4.1 The reactor coolant shall not be heated above 350°F unless the following conditions are met:

[LCO 3.7.1]

a. A minimum turbine cycle steam relieving capability of twelve (12) main steam safety valves operable (shall be) (as specified in Tables 3.7.1-1 and 3.7.1-2) — L1

b. Three auxiliary feedwater pumps must be operable — See 3.7.4

c. A minimum of 35,000 gallons of water in the condensate storage tank and an unlimited water supply from the lake via either leg of the plant Service Water System. — see 3.7.5, 3.7.8

d. Essential features including system piping and valves directly associated with the above components are operable. — A3

e. The main steam stop valves are operable and capable of closing in five seconds or less. — see 3.7.2

Add Table 3.7.1-1 and ACTIONS A & B — L1

Add ACTIONS "Note" — L2

ITS

A1

3.4.2 The specific activity of the secondary coolant system shall be ≤ 0.10 μCi/gram dose equivalent I-131 under all modes of operation from cold shutdown through power operation. When the specific activity of the secondary coolant system is > 0.10 μCi/gram dose equivalent I-131, be in at least HOT SHUTDOWN within 6 hours and cold shutdown within the following 30 hours.

See 3.7.15

The specific activity of the secondary coolant system shall be determined to be within the limit by performance of the sampling and analysis program of Table 4.1-2.

MODES 1,2,3

M1

[ACTION C] 3.4.3 If, during power operations, any of the specifications in 3.4.1, with the exception of 3.4.1.b and 3.4.1.d as it applies to 3.4.1.b above, cannot be met within 24 hours, the operator shall initiate procedures to put the plant in the hot shutdown condition. If any of these specifications cannot be met within an additional 48 hours, the operator shall cool the reactor below 350°F using normal procedures.

See 3.7.4

M2

4

MODE 3 - 6hrs
MODE 4 - 12hrs

3.4.4 With one auxiliary feedwater pump and/or essential features inoperable, restore that auxiliary feedwater pump and/or essential features to operable status within 72 hours, or:

a. Submit a Special Report to the Commission within 30 days outlining the cause of the inoperability and the action taken to return the pump and/or essential features to operable status, and;

b. Restore all three auxiliary feedwater pumps and their essential features to operable status within 7 days or be in at least hot shutdown within 6 hours.

See 3.7.4

3.4.5 With two auxiliary feedwater pumps inoperable, restore at least one inoperable auxiliary feedwater pump to operable status within 24 hours or be in at least hot shutdown within 6 hours.

or 1 or more SGs w/ ≥ 3 MSSVs inoperable

M3

Specification 3.3.2

A1

ITS

TABLE 3.5-3

ENGINEERED SAFETY FEATURES INSTRUMENTATION LIMITING OPERATING CONDITIONS

A27

NO.	FUNCTIONAL UNIT	1 TOTAL NO. OF CHANNELS	2 MINIMUM CHANNELS OPERABLE	3 OPERABLE ACTION IF COLUMN 1 OR 2 CANNOT BE MET	APPLICABLE CONDITIONS
1.	SAFETY INJECTION				MODE 1,2,3,4
[T3.3.2-1(1a)] A.	Manual	2	2	ACTION 11 B	≥200°F
[T3.3.2-1(1c)] B.	High Containment Pressure (Hi Level)	3	2	ACTION 12 E	≥200°F
[T3.3.2-1(1d)] C.	High Differential Pressure between Any Steam Line and the Steam Header	3/Steam Line	2/Steam Line	ACTION 12 D	MODE 1,2,3(a) #
[T3.3.2-1(1e)] D.	Pressurizer Low Pressure	3	2	ACTION 12 D	MODES 1,2,3(a) #
[T3.3.2-1(1f)] E.	High Steam Flow in 2/3 Steam Lines Coincident with Low T_{avg} in 2/3 loops	2/Steam Line and 1 T_{avg} Loop	1/Steam Line and 1 T_{avg} in 2 Loops OR 2/Steam Line and 1 T_{avg}	ACTION 12 D	A11 ≥350°F ## MODES 1,2(b),3(b)
[T3.3.2-1(1g)] F.	High Steam Flow in 2/3 Steam Lines Coincident with Low Steam Pressure in 2/3 lines	2/Steam Line and 1 Press/Line	1/Steam Line and 1 Press in 2 Lines OR 2/Steam Line and 1 Press	ACTION 12 E	≥350°F ##
2.	CONTAINMENT SPRAY				A24 MODES 1,2,3,4
[T3.3.2-1(2a)] A.	Manual	2	2	ACTION 13 I	≥200°F
[T3.3.2-1(2c)] B.	High Containment Pressure (Hi Hi Level)	3/Set	2/Set	ACTION 12 E	≥200°F

Amendment No. ~~85~~, 175

Supplement 5

A1

TABLE 3.5-3 (Continued)

ITS

ENGINEERED SAFETY FEATURES INSTRUMENTATION LIMITING OPERATING CONDITIONS

TABLE NOTATIONS

[T3.3.2-1 NOTE A] # Above Low Pressure SI Block Permit interlock.

[T3.3.2-1 Note (b)] ## Trip function may be blocked below Low T_{avg} Interlock setpoint.

The reactor may remain critical below the Power Operating conditions with this feature inhibited for the purpose of starting reactor coolant pumps.

See 3.3.5

[ACTION B] ACTION 11 With the number of OPERABLE channels one less than the Minimum Channels OPERABLE requirement, restore the inoperable channel to OPERABLE status within 48 hours or be in at least the Hot Shutdown Condition within the next 8 hours and the Cold Shutdown Condition within the following 30 hours.

M22

6

[ACTION C,G]

[ACTION D ACTION E] ACTION 12 With the number of OPERABLE channels one less than the Total Number of Channels. Power Operation may proceed ~~until performance of the next required operational test~~ provided the inoperable channel is placed into the tripped condition within 1 hour.

or restore OPERABLE in 6 hours

6

L21

[ACTION I] ACTION 13 With the number of OPERABLE channels one less than the Minimum Channels OPERABLE requirement, restore the inoperable channel to OPERABLE status within 1 hour or be in ~~at least the Hot Shutdown Condition within the next 8 hours~~ and the ~~Cold Shutdown Condition within the following 30 hours~~.

M24

MODE 3 in 7 hrs, MODE 4 in 13 hrs, MODE 5 in 37 hrs

ACTION 14 With the number of OPERABLE channels one less than the Total Number of Channels; place the inoperable channel into the blocked condition within 1 hour, and restore the inoperable channel to OPERABLE status within 48 hours or be in at least the Hot Shutdown Condition within the next 8 hours and the Cold Shutdown Condition within the following 30 hours.

See 3.3.5

M23

[ACTION C] or be in MODE 3 in 12 hours and MODE 5 in 42 hours

[ACTION D/G] or be in MODE 3 in 12 hours and MODE 4 in 18 hours

[ACTION E] or be in MODE 3 in 12 hours, MODE 4 in 18 hours and MODE 5 in 42 hours

Add ACTIONS "Note1"

A5

Add ACTIONS Note 2

L50

Supplement 5

A1

TABLE 3.5-4 (Continued)

ISOLATION FUNCTIONS INSTRUMENTATION LIMITING OPERATING CONDITIONS

ITS

A27

NO.	FUNCTIONAL UNIT	1 TOTAL NO. OF CHANNELS	~~2 MINIMUM CHANNELS OPERABLE~~	3 OPERABLE ACTION IF COLUMN 1 OR 2 CANNOT BE MET	APPLICABLE CONDITIONS
	2. STEAM LINE ISOLATION				
[T3.3.2-1(4d)] A.	High Steam Flow in 2/3 Steam Lines Coincident with Low T_{avg} in 2/3 loops	See Item No. 1.E of Table 3.5-3 for initiating functions and requirements			
[T3.3.2-1(4e)] B.	High Steam Flow in 2/3 Steam Lines Coincident with Low Steam Pressure in 2/3 lines	See Item No. 1.F of Table 3.5-3 for initiating functions and requirements			
[T3.3.2-1(4c)] C.	High Containment Pressure (Hi Hi Level)	See Item No. 2.B of Table 3.5-3 for initiating functions and requirements			
[T3.3.2-1(4a)] D.	Manual	1/Line	~~1/Line~~	ACTION ~~16~~ F	
	3. FEEDWATER LINE ISOLATION				
[T3.3.2-1(5)] A.	Safety Injection	See Item No. 1 of Table 3.5-3 for all Safety Injection initiating functions and requirements			

L42 Add Note (e) to Applicability

A1 ACTION D

L42 Add Note (e) to MODE 2 and 3 APPLICABILITY

L49 Delete MODE 4

A27 M25 ≥350°F

MODES 1, 2(e), 3(e) L42

L50 Add Note 2 to Surveillance Requirements

M27 Add ACTION H
SR 3.3.2.1 SR 3.3.3.5
SR 3.3.2.3 SR 3.3.2.7
SR 3.3.2.4
T 3.3.2-1 Item 6

M12 Add T3.3.2-1 "Allowable Value" Column

Specification 3.9.2

ITS

A1

3.8 REFUELING

Applicability

Applies to operating limitations during refueling operations.

Objective

To minimize the possibility of an accident occurring during refueling operations that could affect public health and safety.

Specification

3.8.1 During refueling operations the following conditions shall be satisfied:

See 3.9.3

a. The equipment door and at least one door in the personnel air lock shall be properly closed. For those systems which provide a direct path from containment atmosphere to the outside atmosphere, all automatic containment isolation valves shall be operable or at least one valve shall be securely closed in each line penetrating the containment.

b. The containment vent and purge system, including the radiation monitors which initiate isolation shall be tested and verified to be operable immediately prior to refueling operations.

R1

c. Radiation levels in the containment and spent fuel storage areas shall be monitored continuously.

M3

[Applicability]
[LCO 3.9.2]

MODE 6

d. Whenever core geometry is being changed, core subcritical neutron flux shall be continuously monitored by at least two source range neutron monitors, each with continuous visual indication in the control room and one with audible

A11

LA4

Supplement 5

ITS

3.8 REFUELING

A1

Applicability

Applies to operating limitations during refueling operations.

Objective

To minimize the possibility of an accident occurring during refueling operations that could affect public health and safety.

Specification

CORE ALTERATIONS, movement of irradiated fuel assemblies within containment

A4

[Applicability] 3.8.1 During refueling operations the following conditions shall be satisfied:

[LCO 3.9.3 a, b, c.1]

closed with 4 bolts

L2

a. The equipment door and at least one door in the personnel air lock shall be properly closed. For those systems which provide a direct path from containment atmosphere to the outside atmosphere, all automatic containment isolation valves shall be operable or at least one valve shall be securely closed in each line penetrating the containment.

blind flange or equivalent

L9

[SR 3.9.3.2]

Manual or automatic

b. The containment vent and purge system, including the radiation monitors which initiate isolation shall be tested and verified to be operable immediately prior to refueling operations.

See 3.3.6

18 mo

L7

c. Radiation levels in the containment and spent fuel storage areas shall be monitored continuously.

R1

d. Whenever core geometry is being changed, core subcritical neutron flux shall be continuously monitored by at least two source range neutron monitors, each with continuous visual indication in the control room and one with audible

See 3.9.2

each valve actuates to isolation position on an actual or simulated actuation signal

A5

L8

ITS

A1

Suspend CORE ALTERATIONS

j. If any of the specified limiting conditions for refueling are not met, refueling of the reactor shall cease work shall be initiated to correct the conditions so that the specified limits are met and no operations which may increase the reactivity of the core shall be made.

[ACTION A]

L4

Suspend movement of irradiated Fuel Assemblies in Containment

k. The reactor shall be subcritical as required by 3.10.8.3.

A2

3.8.2 The Spent Fuel Building Filter system and the Containment Purge filter system shall satisfy the following conditions:

a. The results of the in-place cold DOP and halogenated hydrocarbon tests at greater than 20 percent design flows on HEPA filters and charcoal absorber banks shall show ≥ 99 percent DOP removal and ≥ 99 percent halogenated hydrocarbon removal.

b. Verification by way of laboratory carbon sample analysis from the Spent Fuel Building filter system carbon and the Containment Purge filter system carbon to show ≥ 90 percent radioactive methyl iodide removal in accordance with test 5.b of Table 5-1 of ANSI/ASME N509-1976 except that ≥ 70 percent relative humidity air is required.

See 5.5.11

c. 1. The Spent Fuel Building refueling filter fan shall be shown to operate within ± 10% of the design flow.

2. At least one Containment purge filter fan shall be shown to operate within ± 10% of the design flow and must be operable during core alterations or movement of irradiated fuel assemblies, or at least one automatic containment isolation valve in each line penetrating the containment which provides a direct path from the containment atmosphere to the outside atmosphere shall be securely closed.

See 3.9.7

d. During fuel handling operations, the relative humidity (R.H.) of the air processed by the refueling filter systems shall be ≤ 70 percent.

See 5.5.11.

e. From and after the date that the Spent Fuel Building filter system is made or found to be inoperable for any reason, fuel handling operations in the Spent Fuel Building shall be terminated immediately.

See 3.7.11

Add SR 3.9.3.1

M6

Supplement 5

ITS

(A1)

indication available in the containment. When core geometry is not being changed at least one source range neutron flux monitor shall be in service.

e. At least one residual heat removal loop shall be operable, refueling cavity water level ≥ Plant elevation 272 ft. - 2 in. whenever fuel assemblies are being moved within the reactor pressure vessel, and Tave ≤ 140°F.

f. During reactor vessel head removal and while loading and unloading fuel from the reactor, the minimum boron concentration of 1950 ppm shall be maintained in the primary coolant system and verified by sampling once each shift.

g. Direct communication between the control room and the refueling cavity manipulator crane shall be available whenever changes in core geometry are taking place.

h. Movement of fuel within the core shall not be initiated prior to 100 hours after shutdown.

See 3.9.1 3.9.2 3.9.4 3.9.6

i. The Spent Fuel Building ventilation system shall be operating when handling irradiated fuel in this area. Prior to moving irradiated fuel assemblies in the spent fuel pool, the ventilation system exhaust shall be aligned to discharge through HEPA and impregnated charcoal filters. When in operation, the exhaust flow of the Containment Purge System shall discharge through HEPA and impregnated charcoal filters. When the Containment Purge System is not in operation at least one automatic containment isolation valve shall be secured in each line penetrating the containment which provides a direct path from the containment atmosphere to the outside atmosphere.

See 3.7.11

LA5

L9

[LCO 3.9.7]

[Required Action A1]

ITS

Suspend CORE ALTERATIONS
Suspend movement of irradiated fuel in containment

A.1

[Required Action A.2.1 & A.2.2]

j. If any of the specified limiting conditions for refueling are not met, refueling of the reactor shall cease; work shall be initiated to correct the conditions so that the specified limits are met; and no operations which may increase the reactivity of the core shall be made.

L4

k. The reactor shall be subcritical as required by 3.10.8.3.

See 3.9.1 3.9.2 3.9.3 3.9.4 3.9.6

3.8.2 The Spent Fuel Building Filter system and the Containment Purge filter system shall satisfy the following conditions:

a. The results of the in-place cold DOP and halogenated hydrocarbon tests at greater than 20 percent design flows on HEPA filters and charcoal absorber banks shall show ≥ 99 percent DOP removal and ≥ 99 percent halogenated hydrocarbon removal.

b. Verification by way of laboratory carbon sample analysis from the Spent Fuel Building filter system carbon and the Containment Purge filter system carbon to show ≥ 90 percent radioactive methyl iodide removal in accordance with test 5.b of Table 5-1 of ANSI/ASME N509-1976 except that ≥ 70 percent relative humidity air is required.

See 5.5.11

c. 1. The Spent Fuel Building refueling filter fan shall be shown to operate within ± 10% of the design flow.

2. At least one Containment purge filter fan shall be shown to operate within ± 10% of the design flow and must be operable during core alterations or movement of irradiated fuel assemblies, or at least one automatic containment isolation valve in each line penetrating the containment which provides a direct path from the containment atmosphere to the outside atmosphere shall be securely closed.

[LCO 3.9.7]

[APPLICABILITY]

[Required Action A.1]

L9

[SR 3.9.7.1] d. During fuel handling operations, the relative humidity (R.H.) of the air processed by the refueling filter systems shall be ≤ 70 percent.

add frequency

M18

e. From and after the date that the Spent Fuel Building filter system is made or found to be inoperable for any reason, fuel handling operations in the Spent Fuel Building shall be terminated immediately.

See 3.7.11

Add SR 3.9.7.2
SR 3.9.7.3

M18

Amendment No. ~~78~~, 172

Supplement 5

Specification 3.1.8

A1

3.10 REQUIRED SHUTDOWN MARGINS, CONTROL ROD, AND POWER DISTRIBUTION LIMITS

Applicability

Applies to the required shutdown margins, operation of the control rods, and power distribution limits.

Objective

To ensure (1) core subcriticality after a reactor trip and during normal shutdown conditions, (2) limited potential reactivity insertions from a hypothetical control rod ejection, and (3) an acceptable core power distribution during power operation.

Specification

3.10.1 Full Length Control Rod Insertion Limits

3.10.1.1 (Deleted by Change No. 21 issued 7/6/73)

[LCO 3.1.8]

3.10.1.2 When the reactor is critical, except for physics tests ~~and~~ full length control rod exercises, the shutdown control rods shall be limited in physical insertion as specified in the CORE OPERATING LIMITS REPORT (COLR).

See 3.1.5 3.1.6

[LCO 3.1.8]

3.10.1.3 When the reactor is critical, except for physics tests ~~and~~ full length control rod exercises, the control rods shall be limited in physical insertion as specified in the COLR. Control rod bank insertion beyond the limits specified in the COLR shall be corrected within the time criteria established by the axial power distribution methodology or within one (1) hour, whichever occurs sooner. If bank insertion is not restored to the specified limits (i.e., within one (1) hour or within the time criteria established by the axial power distribution methodology, whichever is sooner) the reactor shall be placed in the hot shutdown condition utilizing normal operating procedures within six (6) hours.

3.10.1.4 At 50 percent of the cycle as defined by burnup, the limits shall be adjusted to the end-of-core values as specified in the COLR.

Add RAs A1, A.2, B.1, C.1, D.1 — M20

Add SR 3.1.8.1
SR 3.1.8.2
SR 3.1.8.3
SR 3.1.8.4 — M21

Add LCO 3.1.8 requirements a, b, and c — M22

Amendment No. ~~141~~, 167

Supplement 5

Specification 3.1.4 (A1)

ITS

See - 3.1.8

All shutdown and

shall be OPERABLE with all individual indicated rod position

3.10.1.5 Except for physics tests, if a full length control rod is withdrawn as follows:

[LCO 3.1.4]

- at positions ≥ 200 steps and is > 15 inches out of alignment with its bank position, or (A4)

- at positions < 200 steps and is > 7.5 inches out of alignment with the average of its bank position

M7

ONE

then within two hours, perform the following:

[RAB.1] a. Correct the situation, or

within 72 hours perform SR 3.2.1.1 + 3.2.2.1

[RAB2.4] [RAB.2.5] b. Determine by measurement the hot channel factors and apply Specification 3.10.2.1, or and (M8)

Reduce Thermal

[RAB.2.2] c. Limit power to ≤ 70 percent of rated power

3.10.1.6 Insertion limits do not apply during physics tests or during period exercise of individual rods. However, the shutdown margin indicated in Figure 3.10-2 must be maintained, except during the low power physics test to measure control rod worth and shutdown margin. For this test, the reactor may be critical with all but one full length control rod inserted.

[NOTE TO LCO 3.1.5 AND LCO 3.1.6]

See 3.1.5 3.1.6

See 3.1.8

3.10.2 Power Distribution Limits

3.10.2.1 At all times except during low power physics tests, the hot channel factors, $F_Q(Z)$ and $F_{\Delta H}$, defined in the basis, must meet the following limits:

$$F_Q(Z) \leq (F_Q^{RTP}/P) \times K(Z) \text{ for } P > 0.5$$

$$F_Q(Z) < (F_Q^{RTP}/0.5) \times K(Z) \text{ for } P \leq 0.5$$

$$F_{\Delta H} < F_{\Delta H}^{RTP} \ (1 + PF_{\Delta H} \ (1-P) \ )$$

See 3.2.1 + 3.2.2

Add MODES 1 + 2 (M28)

Add RA B.2.1.1, B.2.1.2
RA B.2.3, B2.6
RA C.1, D.1.1, D.1.2
RA D.2

(M9)

Supplement 5

ITS

A1

3.10.1.5 Except for physics tests, if a full length control rod is withdrawn as follows:

- at positions ≥ 200 steps and is > 15 inches out of alignment with its bank position, or

- at positions < 200 steps and is > 7.5 inches out of alignment with the average of its bank position

then within two hours, perform the following:

a. Correct the situation, or

b. Determine by measurement the hot channel factors and apply Specification 3.10.2.1, or

c. Limit power to 70 percent of rated power

3.10.1.6 Insertion limits do not apply during physics tests or during period exercise of individual rods. However, the shutdown margin indicated in Figure 3.10-2 must be maintained, except during the low power physics test to measure control rod worth and shutdown margin. For this test, the reactor may be critical with all but one full length control rod inserted.

See 3.1.4 3.1.5 3.1.7 3.1.8

L1

3.10.2 Power Distribution Limits

MODE

[applicability]

3.10.2.1 At all times except during low power physics tests, the hot channel factors, $F_Q(Z)$ and $F_{\Delta H}$, defined in the basis, must meet the following limits:

[LCO 3.2.1]

M

limits specified in the COLR.

LA1

$F_Q(Z) \leq (F_Q^{RTP}/P) \times K(Z)$ for $P > 0.5$

$F_Q(Z) < (F_Q^{RTP}/0.5) \times K(Z)$ for $P \leq 0.5$

as approximated by $F_Q^V(Z)$

A2

LA1

$F_{\Delta H} < F_{\Delta H}^{RTP} (1 + PF_{\Delta H} (1-P))$

See 3.2.2

LA1

Add RA A.2.2
RA A.2.4
RA B.1

See 3.2.2

M2

Supplement 5

Specification 3.2.1

ITS

A1

where P is the fraction of rated power (2300 Mwt) at which the core is operating. $F_Q(Z)$ is the measured $F_Q^N(Z)$ multiplied by the measurement uncertainty factor $F_u^N = 1.05$ and the engineering factor $F_Q^E = 1.03$. $F_{\Delta H}$ is the measured $F_{\Delta H}^N$ multiplied by a 1.04 measurement uncertainty factor. K(Z) is the normalized $F_Q(Z)$ as a function of core height specified in the CORE OPERATING LIMITS REPORT (COLR). F_Q^{RTP} is the F_Q limit at RATED THERMAL POWER (RTP). $F_{\Delta H}^{RTP}$ is the $F_{\Delta H}$ limit at RATED THERMAL POWER. $PF_{\Delta H}$ is the Power Factor Multiplier for $F_{\Delta H}^{RTP}$, F_Q^{RTP}, $F_{\Delta H}^{RTP}$ and $PF_{\Delta H}$ are specified in the COLR.

LA1 See 3.2.2 LA1 See 3.2.2 LA1

3.10.2.1.1 Following ~~initial loading~~ refueling and prior to exceeding 75% RTP, or ~~upon~~ achieving equilibrium conditions after exceeding by 10% or more of rated power, the power $F_Q(Z)$ was last determined, and at least once per ~~effective full power month~~ (within 12 hours) 3 EFPDs power distribution maps using the movable detector system shall be made to confirm that the ~~hot channel factor~~ F_Q limits of Specification 3.10.2.1 are satisfied and to establish the target axial flux difference as a function of power level (called the target flux difference).*

[SR 3.2.1.1]

M5 A2 See 3.2.2

If either measured hot channel factor exceeds the specified limit, the reactor power shall be reduced so as not to exceed a fraction equal to the ratio of the $F_Q(Z)$ or $F_{\Delta H}$ limit to the measured value, whichever is less, and the high neutron flux trip setpoint shall be reduced by the same ratio.

A4

If subsequent incore mapping cannot, within a 24 (72) hour period, demonstrate that the hot channel factors are met, the overpower ΔT and overtemperature ΔT trip setpoints shall be similarly reduced.

[RA A.2.3]

See 3.2.2 L2 A2 M28

During power escalation at the beginning of each cycle, the design target may be used until a power level for extended operation has been achieved.

See 3.2.3

Supplement 5

Specification 3.2

ITS

where P is the fraction of rated power (2300 Mwt) at which the core is operating. $F_Q(Z)$ is the measured $F_Q^N(Z)$ multiplied by the measurement uncertainty factor $F_Q^N = 1.05$ and the engineering factor $F_Q^E = 1.03$. $F_{\Delta H}$ is the measured $F_{\Delta H}^N$ multiplied by a 1.04 measurement uncertainty factor. K(Z) is the normalized $F_Q(Z)$ as a function of core height specified in the CORE OPERATING LIMITS REPORT (COLR). F_Q^{RTP} is the F_Q limit at RATED THERMAL POWER (RTP). $F_{\Delta H}^{RTP}$ is the $F_{\Delta H}$ limit at RATED THERMAL POWER. $PF_{\Delta H}$ is the Power Factor Multiplier for $F_{\Delta H}$. F_Q^{RTP}, $F_{\Delta H}^{RTP}$, and $PF_{\Delta H}$ are specified in the COLR.

3.10.2.1.1 31 EFPDs (refueling prior to exceeding 75% RTP) Following initial loading, or upon achieving equilibrium conditions after exceeding by 10% or more of rated power, the power $F_Q(Z)$ was last determined, and at least once per effective full power month, power distribution maps using the movable detector system shall be made to confirm that the hot channel factor limits of Specification 3.10.2.1 are satisfied and to establish the target axial flux difference as a function of power level (called the target flux difference).*1

[SR 3.2.2.1] LA5

[RA A.1.1] [RA A.1.2.1] If either measured hot channel factor ($F_{\Delta H}$) exceeds the specified limit, the reactor power shall be reduced so as not to exceed a fraction equal to the ratio of the $F_Q(Z)$ or $F_{\Delta H}$ limit to the measured value, whichever is less, and the high neutron flux trip setpoint shall be reduced by the same ratio.

[RA A.1.2.2] (to < 55% RTP) If subsequent incore mapping cannot, within a 24-hour period, demonstrate that the hot channel factors are met, the overpower ΔT and overtemperature ΔT trip setpoints shall be similarly reduced.

Then restore $F_{\Delta H}^N$ to within limits within 4 hours OR

Add Note to Condition A

RA A.2
RA A.3
Note to RA A.3
RA B.1

During power escalation at the beginning of each cycle, the design target may be used until a power level for extended operation has been achieved.

Supplements
Amendment No. 87, 115, 119, 141, 151

A1

3.10.2.2.3 With successive measurements indicating the enthalpy rise hot channel factor, $F^N_{\Delta H}$, to be increasing with exposure, the total peaking factor, $F_Q(Z)$, shall be further increased by two percent over that specified in Specifications 3.10.2.2, 3.10.2.2.1, and

SR 3.2.2.1 NOTE

$F^V_Q(z)$

A2

For each OPERABLE ex core channel — A24 — A1

b. at power levels less than 90 percent of rated power or 0.9 x APL (whichever is less) provided the indicated axial flux difference does not exceed the limits specified in the COLR. — L7

3.10.2.10 Alarms shall normally be used to indicate non-conformance with the flux difference requirement of 3.10.2.6 or the flux difference-time requirement of 3.10.2.7.a. If the alarms are temporarily out of service, the axial flux difference shall be logged and conformance with the limits assessed, every hour for the first 24 hours, and half-hourly thereafter. — A15

[SR 3.2.3.2 Frequency Note] [SR 3.2.3.2]

Insert 3.2.3.4

3.10.2.11 The axial flux difference target band about the target axial flux difference shall be determined in conjunction with the measurement of $F_Q(Z)$ as specified in 3.10.2.1.1. The allowable values of the target band are specified in the COLR. Redefinition of the target band from more restrictive to less restrictive ranges between determinations of the target axial flux difference is allowed when appropriate redefinitions of APL are made. Redefinition of the target band from less restrictive to more restrictive ranges is allowed only in conjunction with the determination of a new target axial flux difference. — M20 — L8 — A16

[LCO 3.2.3.a]

3.10.3 Quadrant Power Tilt Limits

3.10.3.1 Except for physics tests and during power increases below 50 percent of rated power, whenever the indicated quadrant power tilt ratio exceeds 1.02, the tilt condition shall be eliminated within two hours or the following actions shall be taken:

a. Restrict core power level and reset the power range high flux setpoint to be less than two percent of rated values for every percent of indicated power tilt ratio exceeding 1.0, and — LA5

See 3.2.4

Add Note to SR 3.2.3.2 — A15

Supplement 5

ITS

M29

A1

3.10.4 Rod Drop Time

verify the rod

≤

[SR 3.1.4.3] 3.10.4.1 The drop time of each control rod shall be not greater than 1.8 seconds at full flow and operating temperature from the beginning of rod motion to dashpot entry.

>540°F

with all reactor coolant pumps operating

3.10.5 Reactor Trip Breakers

3.10.5.1 The reactor shall not be made critical unless the following conditions are met:

a. Two reactor trip breakers are operable.

b. Reactor trip bypass breakers are racked out or removed.

c. Two trains of automatic trip logic are operable.

Decay of stationary gripper coil Voltage

M24

3.10.5.2 During power operation, the requirements of 3.10.5.1 may be modified to allow the following components to be inoperable. If the system is not restored to meet the requirements of 3.10.5.1, the reactor shall be placed in the hot shutdown condition utilizing normal operating procedures within the next 8 hours.

a. One reactor trip breaker may be inoperable for up to 12 hours.

b. One train of automatic trip logic may be inoperable for up to 12 hours.

c. One reactor trip bypass breaker may be racked in and closed for up to 12 hours.

See 3.3.1

3.10.5.3 With one of the diverse trip features inoperable (shunt trip attachment/undervoltage trip attachment) on one of the reactor trip breakers, power operation may continue for up to 48 hours. If the

M.26

Add Condition A, associated actions, and completion times for control rod drop time not within limits

Supplement 5

TABLE 4.1-1 (Continued)

MINIMUM FREQUENCIES FOR CHECKS, CALIBRATIONS AND TEST OF INSTRUMENT CHANNELS

	Channel Description	Check	Calibration	Test	Remarks
32.	Loss of Power				See 3.3.5
	a. 480 Emerg. Bus Undervoltage (Loss of Voltage)	N.A.	R	R	
	b. 480 Emerg. Bus Undervoltage (Degraded Voltage)	N.A.	R	R	
33.	Auxiliary Feedwater Flow**** Indication	M [SR 3.3.3.1]	R [SR 3.3.3.2]	N.A.	
34.	Reactor Coolant System** Subcooling Monitor	M	R	N.A.	R1
35.	PORV Position Indicator***	N.A.	N.A.	R [SR 3.3.3.3]	A39
36.	PORV Blocking Valve*** Position Indicator	N.A.	N.A.	R [SR 3.3.3.3]	
37.	Safety Relief Valve Position*** Indicator	N.A.	N.A.	R [SR 3.3.3.3]	L27
38.	Noble Gas Effluent Monitors*****				
	a. Main Steam Line	Q	R	Q	R1

** Instrument for Detection of Inadequate Core Cooling - NUREG 0578 Item 2.1.3.b.
*** Direct Indication of Power Operated Relief Valve and Safety Valve Position - NUREG 0578 Item 2.1.3.a.
**** Auxiliary Feedwater Flow Indication to Steam Generator - NUREG 0578 Item 2.1.7.b.
***** Noble Gas Effluent Monitors - NUREG-0737 Item II.F.1.1.

A19

Supplement 5

Specification 3.3.3

D

ITS

A1

In accordance with the Diesel Fuel oil Testing program — A24

TABLE 4.1.2 (Continued)

FREQUENCIES FOR SAMPLING TESTS

LA6

[SR3.8.3.2]

	Check	Frequency	Maximum Time Between Tests
~~11. U-2 Diesel Generator Fuel Oil Storage Tank~~	API or Specific Gravity, Water and Sediment, Viscosity	Monthly	45 days
~~12. U-1 I-C Turbine Fuel Oil Storage Tanks or Tank Truck~~	API or Specific Gravity, Water and Sediment, Viscosity	Prior to transfer to U-2	N/A

See ITS 5.5

Verify fuel oil properties of stored fuel oil are tested in accordance with, and maintained within the limits of the Diesel fuel oil Testing Program — A24

Supplement 5

Specification 3.8.3

ITS

Add general program statement for ITS 5.5.13 (A38)

(A1)

TABLE 4.1.2 (Continued)

FREQUENCIES FOR SAMPLING TESTS

		Check	Frequency	Maximum Time Between Tests	(M15)
[5.5.13.b]	11. U-2 Diesel Generator Fuel Oil Storage Tank	API or Specific Gravity, Water and Sediment, Viscosity, and cloud point (M15)	Monthly	38.75 days per SR 3.0.2 ~~45 days~~	
[5.5.13.b]	12. U-1 I-C Turbine Fuel Oil Storage Tanks or Tank Truck	API or Specific Gravity, Water and Sediment, Viscosity	~~Prior to transfer to U-2~~ 31 days (M15)	N/A	

Add 5.5.13.a (M15)

The provisions of SR 3.0.2 and SR 3.0.3 are applicable to the Diesel Fuel Oil Testing Program surveillance frequencies. (A38)

Supplement 5

Specification 5.5

A1

ITS

NOTES TO TABLE 4.1-2

(1) A gross activity analysis shall consist of the quantitative measurement of the total radioactivity of the primary coolant of units of μCi/gram.

(2) A radiochemical analysis shall consist of the quantitative measurement of each radionuclide with half life greater than 30 minutes making up at least 95% of the total activity of the primary coolant.

LA6

A27

Add SR 3.4.16.3 Note

(3) When iodine or particulate radioactivity levels exceed 10% of the limit in Specification 3.9.2.1, the sampling frequency shall be increased to a minimum of once each day.

LA10

(5) Deleted.

[SR 3.4.16.3] (6) Sample to be taken after a minimum of 2EFPD and 20 days of power operation have elapsed since the reactor was last subcritical for 48 hours or longer.

[SR 3.4.16.2] [SR 3.4.16.3] [RA A.1] (7) Samples are to be taken in the power operating condition.

(8) Sample taken at all operating conditions whenever the specific activity exceed 1.0 μCi/gram DOSE EQUIVALENT I-131 or 100/E Ci/gram. These samples are to be taken until the specific activity of the reactor coolant system is restored within its limits.

[SR 3.4.16.2] (9) One sample between 2 and 6 hours following a thermal power change exceeding 15 percent of the rated thermal power within a one-hour period. Samples are required when in the hot shutdown or power operating modes.

(10) Sample whenever that gross activity determination indicates iodine concentrations are greater than 10% of the allowable limit.

(11) Sample whenever the gross activity determination indicates iodine concentrations are below 10 percent of the allowable limit.

See 3.7.15

NA* - Not applicable.

Specification 3.4.14

A1

TABLE 4.1-3 (Continued)
FREQUENCIES FOR EQUIPMENT TESTS

	Check	Frequency	Maximum Time Between Test
		~~2. Whenever integrity of a pressure isolation valve listed in Table 3.1-1 cannot be demonstrated, the integrity of the remaining valve in each high pressure line having a leaking valve shall be determined and recorded daily. In addition, the position of the other closed valve located in the high pressure piping shall be recorded daily.~~	
18. Automatic Bus Transfers a) Auxiliary Feedwater Header Discharge Valve to Steam Generator A, V2-16A b) Turbine Building, Cooling Water Isolation Valve, V6-16C	Test thermal and magnetic trip elements of respective molded case circuit breakers	Each refueling shutdown.	NA

L24

See 3.8.9

A1

Class 2 and Class 3 components were chosen based on Regulatory Guide 1.26 and ANSI N18.2 and N18.2a "Nuclear Safety Criteria for the Design of Stationary Pressurized Water Reactor Plants."

A17

The Surveillance Requirements for inspection of the steam generator tubes ensure that the structural integrity of this portion of the RCS will be maintained. The program for inservice inspection of steam generator tubes is based on a modification of Regulatory Guide 1.83, Revision 1. Inservice inspection of steam generator tubing is essential in order to maintain surveillance of the conditions of the tubes for evidence of mechanical damage or progressive degradation. Inservice inspection of steam generator tubing also provides a means of characterizing the nature and cause of any tube degradation so that corrective measures can be taken.

Wastage-type defects will be minimized with proper chemistry treatment of the secondary coolant. If defects or significant degradations should develop in service, this condition is expected to be detected during inservice steam generator tube examinations. Plugging will be required for all tubes with imperfections exceeding the plugging limit. Steam generator tube inspections by means of eddy current testing have demonstrated the capability to reliably detect degradation that has penetrated 20% of the original tube wall thickness.

Whenever the results of any steam generator tubing inservice inspection fall into Category C-3, these results will be reported to the Commission prior to resumption of plant operation. Such cases will be considered by the Commission on a case-by-case basis and may result in a requirement for analysis, laboratory examinations, tests, additional eddy-current inspection, and revision of the Technical Specifications.

4.2.2 Materials Irradiation Surveillance Specimens

A18

The reactor vessel material surveillance specimens shall be removed and examined to determine changes in their material properties, as required by Appendix H to 10CFR50.

4.2.3 Primary Pump Flywheels

[5.5.7]

LA8

The flywheels shall be visually examined at the first refueling after each ten year inspection. At the fourth refueling after each ten year inspection and at each fourth refueling thereafter, the outside surfaces shall be examined by ultrasonic methods. The examinations scheduled for Refueling Outage 17, in 1996, may be deferred to Refueling Outage 18.

A32

This program provides controls for the inspection of each reactor coolant pump flywheel in accordance with the Inservice Inspection Program.

Verify battery capacity is ≥ 80% for the A battery and ≥ 90% for the B battery when (M11) (A5)

4.6.3.5 The batteries shall be subjected to a performance test once every 1 (one) year. (A21) [SR 3.8.4.6] (CTS insert 4.6.3.5A) discharge

4.6.3.6 The batteries shall be subjected to a service test at least once per 18 months during a shutdown (when) verify (required) that the battery capacity is adequate to supply and maintain in OPERABLE status (and) (required) the actual or simulated emergency loads for the design duty cycle. Surveillance 4.6.3.5 may be performed at five year intervals in lieu of this test (A15) [SR 3.8.4.5 and Note 2] [SR 3.8.4.5 Note 1]

4.6.4 Pressurizer Heaters' Emergency Power Supply

The emergency power supply for the pressurizer heaters shall be demonstrated operable each refueling shutdown by transferring power from normal to the emergency power supply and energizing the heaters. See 3.4.9

4.6.5 Battery Chargers (ISI 25.7) (Verify) (terminal) (M12)

[SR 3.8.4.1] Demonstrate the in-service battery charger is operable by monitoring the output voltage daily five days per week and during normal equalizing charge. (7 days) (L4)

Basis

The tests specified are designed to demonstrate that the diesel generators will provide power for operation of equipment. They also assure that the emergency generator system controls and the control systems for the safety features equipment will function automatically in the event of a loss of all normal 480 V AC station service power. (A6)

The test to ensure proper operation of engineered safety features upon loss of AC power is initiated by tripping the breakers supplying normal power to the 480 volt buses and initiating a safety injection signal. This test demonstrates the proper tripping of motor feeder breakers, main supply and tie breakers on the affected bus, operation of the diesel generators, and sequential starting of essential equipment.

Add SR 3.8.4.2
SR 3.8.4.3
SR 3.8.4.4 (M13)

Add SR 3.8.4.6 Note (A16)

ITS

A1

4.9 REACTIVITY ANOMALIES

Applicability

Applies to potential reactivity anomalies.

Objective

To require evaluation of reactivity anomalies within the reactor.

Specification

M2

[SR 3.1.2.1] Following a normalization of the computed boron concentration as a function of burnup, the actual boron concentration of the coolant shall be periodically compared with the predicted value. If the difference between the observed and predicted steady-state concentrations reaches the equivalent of one percent in reactivity, submit a Special Report to the Commission within 30 days.

L7

Basis

To eliminate possible errors in the calculations of the initial reactivity of the core and the reactivity depletion rate, the predicted relation between fuel burn-up and the boron concentration, necessary to maintain adequate control characteristics, must be adjusted (normalized) to accurately reflect actual core conditions. When full power is reached initially, and with the control rod groups in the desired positions, the boron concentration is measured and the predicted curve is adjusted to this point. As power operation proceeds, the measured boron concentration is compared with the predicted concentration and the slope of the curve relating burnup and reactivity is compared with that predicted. This process of normalization should

A7

Add LCO 3.1.2 + Applicability
RAs A.1, A.2 + B.1

M3

ITS

4.8 AUXILIARY FEEDWATER SYSTEM

Applicability

Applies to periodic testing requirements of the turbine-driven and motor-driven auxiliary feedwater pumps.

Objective

To verify the operability of the auxiliary feedwater system and its ability to respond properly when required.

Specification

31 days on a STAGGERED TEST BASIS

[SR 3.7.4.2] 4.8.1 Each motor driven auxiliary feedwater pump will be started at monthly intervals, run for 15 minutes, and determined that it is operable developed head ≥ req'd head and Note 2

[SR 3.7.4.2] 4.8.2 The steam turbine driven auxiliary feedwater pump by using motor operated steam admission valves will be started at monthly intervals, run for 15 minutes, and determined that it is operable when the reactor coolant system is above the cold shutdown condition. When periods of reactor cold shutdown extend this interval beyond one month, the test shall be performed within 24 hours of achieving stable plant conditions at ≥1000 psig in the steam generator following plant heatup.

[SR 3.7.4.2 NOTE 1]

automatic

[SR 3.7.4.3] 4.8.3 The auxiliary feedwater pumps discharge valves will be tested by operator action at monthly intervals.

18

4.8.4 These tests shall be considered satisfactory if control board indication and subsequent visual observation of the equipment demonstrate that all components have operated properly.

actual or simulated actuation signal

that are not locked, sealed, or otherwise secured in position

Add SR 3.7.4.1
SR 3.7.4.4
SR 3.7.4.5
SR 3.7.4.6

L6 L20 M1 L20 A11 M12 L7 A12 A13 M12 M13