United States Nuclear Regulatory Commission
Attachment to Serial: RNP-RA/97-0011

2. Enclosure 9 to Serial RNP-RA/96-0141, "Conversion Package Section 3.1"

   a. Part 1, "Markup of Current Technical Specifications (CTS)"

Remove	Insert
CTS Page 3.1-11 for ITS Specification 3.1.3	CTS Page 3.1-11 for ITS Specification 3.1.3

Specification 3.1.3

ITS

(A1)

3.1.3 Minimum Conditions for Criticality

See 3.1.8

MODE 1 + MODE 2 with KeFF ≥ 1.0

[LCO 3.1.3]
[Applicability]

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.

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

[RA B.1]
[RA A.1]

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.

(L2)

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)

Add Applicabity for MTC lower limit (M4)

(E)

Add RA C.1 (L5)

Add SR 3.1.3.1
SR 3.1.3.2

(M6)

3. Enclosure 11 to Serial RNP-RA/96-0141, "Conversion Package Section 3.3"

   a. Part 1, "Markup of Current Technical Specifications (CTS)"

Remove	Insert
CTS Page 3.5-12 for ITS Specification 3.3.1	CTS Page 3.5-12 for ITS Specification 3.3.1
CTS Page 3.5-13b for ITS Specification 3.3.1	CTS Page 3.5-13b for ITS Specification 3.3.1

   b. Part 8, "Proposed HBRSEP, Unit No. 2 ITS"

Remove	Insert
Page 3.3-17	Page 3.3-17

Specification 3.3.1

A1

TABLE 3.5-2

REACTOR TRIP INSTRUMENTATION LIMITING OPERATING CONDITIONS

A27

ITS	NO.	FUNCTIONAL UNIT	1 TOTAL NO. OF CHANNELS	2 MINIMUM CHANNELS OPERABLE	3 OPERATOR ACTION IF COLUMN 1 OR 2 CANNOT BE MET	APPLICABLE CONDITIONS
[T3.3.1-1 (1)]	1.	Manual	2	~~2~~	ACTION ~~1~~ B	Reactor Critical (MODES 1, 2)
			2	~~2~~	ACTION ~~8~~ C	Hot/Cold Shutdown* (a) (MODES 3, 4, 5)
[T3.3.1-1 (2)]	2.	Nuclear Flux Power Range				
		A. High Setpoint	4	~~3~~	ACTION ~~2~~ D	Reactor Critical (MODES 1, 2)
		B. Low Setpoint	4	~~3~~	ACTION ~~2~~ E	Reactor Critical** (b) (MODES 1, 2)
[T3.3.1-1 (3)]	3.	Nuclear Flux Intermediate Range	2	~~2~~	ACTION ~~8~~ F, G, H	Reactor Critical*** (c, d)
[T3.3.1-1 (4)]	4.	Nuclear Flux Source Range				
		A. Startup	2	~~2~~	ACTION ~~4~~ I	Reactor Critical*** (MODE 2) (d)
		B. Shutdown	~~2~~ 1	~~1~~	ACTION ~~5~~ L	Hot/Cold Shutdown (e)
		C. Shutdown	2	~~2~~	ACTION ~~8~~ K	Hot/Cold Shutdown* (a) (MODES 3, 4, 5)
[T3.3.1-1 (5)]	5.	Overtemperature ΔT	3	~~2~~	ACTION ~~6~~ E	Reactor Critical (MODES 1, 2)
[T3.3.1-1 (6)]	6.	Overpower ΔT	3	~~2~~	ACTION ~~6~~ E	Reactor Critical (MODES 1, 2)
[T3.3.1-1 (7.a)]	7.	Low Pressurizer Pressure	3	~~2~~	ACTION ~~6~~ M	***** (MODE 1 (f))
[T3.3.1-1 (7.b)]	8.	Hi Pressurizer Pressure	3	~~2~~	ACTION ~~6~~ E	Reactor Critical (MODES 1, 2)
[T3.3.1-1 (8)]	9.	Pressurizer Hi Water Level	3	~~2~~	ACTION ~~6~~ M	***** (MODE 1 (f))
[T3.3.1-1 (9)]	10.	Low Reactor Coolant Flow				
		A. Single Loop	3/loop	~~2/loop~~	ACTION ~~6~~ N	>45% of rated power (MODE 1 (g))
		B. Two Loop	3/loop	~~2/loop~~	ACTION ~~6~~ M	**** (MODE 1 (h))

A27 A28 E A6 L10

Amendment No. 85

License Amendment Request, 12/10/95

A1

ITS

TABLE 3.5-2 (Continued)

REACTOR TRIP INSTRUMENTATION LIMITING OPERATING CONDITIONS

TABLE NOTATIONS

(a) * With the reactor trip breakers closed. and rods not fully inserted or Rod Control system capable of rod withdrawal. L35

(b) ** Below the P-10 (Low Setpoint Power Range Neutron Flux Interlock) setpoint.

(d) *** Below the P-6 (Intermediate Range Neutron Flux Interlock) setpoint.

(h) **** Above the ~~P-10 (Low Setpoint Power Range Neutron Flux Interlock) setpoint or~~ P-7 (Turbine First Stage Pressure Interlock) setpoint and below the P-8 (Low Setpoint Power Range Neutron Flux Interlock) setpoint.

(f) ***** Above the ~~P-10 (Low Setpoint Power Range Neutron Flux Interlock) setpoint or~~ P-7 (Turbine First Stage Pressure Interlock) setpoint. L5

Add Notes (c), (e), (i) A26

ACTION STATEMENTS

[ACTION B] ~~ACTION 1~~ - With the number of OPERABLE channels one less than the Minimum Channels OPERABLE requirement, restore the inoperable channel to OPERABLE status within 48 ~~12~~ hours, or be in MODE 3 ~~the Hot Shutdown Condition~~ within 54 ~~the next 8~~ hours and open RTBs in 55 hours. L6

[ACTION D] ~~ACTION 2~~ - With the number of OPERABLE channels one less than the Total Number of Channels, Startup and/or Power Operation may proceed provided the following Conditions are satisfied: Add RA D.2.2 "NOTE" L7

a. The inoperable channel is placed in the tripped condition within 6 ~~1~~ hour.

b. Either, thermal power is restricted to less than or equal to 75% of rated power ~~and the Power Range Neutron Flux trip setpoint is reduced to less than or equal to 85% of rated power within 4 hours~~, or, the Quadrant Power Tilt Ratio is monitored within 12 hours and every 12 hours thereafter, using the movable incore detectors to confirm that the normalized symmetric power distribution is consistent with the indicated Quadrant Power Tilt Ratio. or be in MODE 3 in 12 hours L8 M6

~~ACTION 3~~ - With the number of channels OPERABLE one less than the Minimum Channels OPERABLE requirement and with the thermal power level:

[ACTION H] a. Below the P-6 (Intermediate Range Neutron Flux Interlock) setpoints, restore the inoperable channel to OPERABLE status prior to increasing thermal power above the P-6 setpoint.

[ACTION F] b. Above the P-6 (Intermediate Range Neutron Flux Interlock) setpoint but below 10% of rated power, ~~restore the inoperable channel to OPERABLE status prior to increasing thermal power above 10% of rated power~~. Reduce power to < P6 in 2 hrs or increase power to > P10 in 2 hrs M7 L36 E

~~Amendment No.~~

License Amendment Request, 12/10/95

Table 3.3.1-1 (page 5 of 7)
Reactor Protection System Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS	CONDITIONS	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE	TRIP SETPOINT
17. Reactor Protection System Interlocks						
a. Intermediate Range Neutron Flux, P-6	2(d)	2	S	SR 3.3.1.11 SR 3.3.1.13	≥ 7.29 E-11 amp	≥ 1 E-10 amp
b. Low Power Reactor Trips Block, P-7	1	1 per train	T	SR 3.3.1.11 SR 3.3.1.13	NA	NA
c. Power Range Neutron Flux, P-8	1	4	T	SR 3.3.1.11 SR 3.3.1.13	≤ 41.00% RTP	≤ 40% RTP
d. Power Range Neutron Flux, P-10	1,2	4	S	SR 3.3.1.11 SR 3.3.1.13	≥ 9.00% RTP and ≤ 11.00% RTP	≥ 10% RTP
e. Turbine Impulse Pressure, P-7 input	1	2	T	SR 3.3.1.1 SR 3.3.1.10 SR 3.3.1.13	≤ 10.71% turbine power	≤ 10% turbine power
18. Reactor Trip Breakers(i)	1,2	2 trains	R,V	SR 3.3.1.4	NA	NA
	3(a), 4(a), 5(a)	2 trains	C,V	SR 3.3.1.4	NA	NA
19. Reactor Trip Breaker Undervoltage and Shunt Trip Mechanisms	1,2	1 each per RTB	U	SR 3.3.1.4	NA	NA
	3(a), 4(a), 5(a)	1 each per RTB	C	SR 3.3.1.4	NA	NA
20. Automatic Trip Logic	1,2	2 trains	Q,V	SR 3.3.1.5	NA	NA
	3(a), 4(a), 5(a)	2 trains	C,V	SR 3.3.1.5	NA	NA

(a) With Reactor Trip Breakers (RTBs) closed and rods not fully inserted or Rod Control System capable of rod withdrawal.
(b) Below the P-10 (Power Range Neutron Flux) interlock.
(c) Above the P-6 (Intermediate Range Neutron Flux) interlock.
(d) Below the P-6 (Intermediate Range Neutron Flux) interlock.
(e) With the RTBs open. In this condition, source range Function does not provide reactor trip but does provide indication and alarm.
(f) Above the P-7 (Low Power Reactor Trips Block) interlock.
(g) Above the P-8 (Power Range Neutron Flux) interlock.
(h) Above the P-7 (Low Power Reactor Trips Block) interlock and below the P-8 (Power Range Neutron Flux) interlock.
(i) Including any reactor trip bypass breakers that are racked in and closed for bypassing an RTB.

4. Enclosure 12 to Serial RNP-RA/96-0141, "Conversion Package Section 3.4"

a. Part 1, "Markup of Current Technical Specifications (CTS)"

Remove	Insert
CTS Page 3.3-5 for ITS Specification 3.4.8	CTS Page 3.3-5 for ITS Specification 3.4.8
- - -	CTS Page 4.1-10 for ITS Specification 3.4.13 (After CTS Page 3.1-16 for ITS Specification 3.4.13)

b. Part 2, "Discussion of Changes (DOCs) for CTS Markup"

Remove	Insert
Page 1	Page 1
- - -	Pages 14a (After Page 14)

ITS

A1

See 3.5.1 3.5.2 3.5.3 3.5.4

3.3.1.3 When the reactor is in the hot shutdown condition, the requirements of 3.3.1.1 and 3.3.1.2 shall be met. Except that the accumulators may be isolated or otherwise inoperable relative to the requirements of 3.3.1.1.b. In addition, any one component as defined in 3.3.1.2 may be inoperable for a period equal to the time period specified in the subparagraphs of 3.3.1.2 plus 48 hours, after which the plant shall be placed in the cold shutdown condition utilizing normal operating procedures. The safety injection pump power supply breakers must be racked out when the reactor coolant system temperature is below 350°F and the system is not vented to containment atmosphere.

See 3.4.12

[LCO 3.4.8] 3.3.1.4 When the reactor is in the cold shutdown condition (except refueling operation when Specification 3.8.1.e applies), both residual heat removal ~~loops~~ train must be operable. Except that either the normal or emergency power source to both residual heat removal loops may be inoperable. and 1 RHR train in operation

MODE 5 Loops not filled

A7 A8 M15

[ACTION A] a. If one residual heat removal ~~loop~~ train becomes inoperable during cold shutdown operation, within 24 hours verify the existence of a method to add make-up water to the reactor coolant system such as charging pumps, safety injection pumps (under adequate operator control to prevent system overpressurization), or primary water (if the reactor coolant system is open for maintenance) as back-up decay heat removal method. Initiate action to Restore the inoperable RHR ~~loop~~ train to operable status within 14 days or immediately prepare and submit a Special Report to the Commission within the next 30 days outlining the action taken, the cause of the inoperability, and the plans and schedule for restoring the loop to operable status.

A12 L17 L16 A12

[ACTION B] b. If both residual heat removal ~~loops~~ train become inoperable during cold shutdown operation, or no RHR train in operation, close all containment penetrations providing direct access from the containment atmosphere to the outside atmosphere prior to the reactor coolant average temperature exceeding 200°F, immediately initiate action to restore at least one residual

L4 A10 M14

Add LCO "NOTES"

L5

E

)

TABLE 4.1.2

FREQUENCIES FOR SAMPLING TESTS

	Check	Frequency	Maximum Time Between Tests	
1. Reactor Coolant Samples	- Gross Activity[1]	Minimum 1 Per 72 hrs.	3 days	See 3.4.16
	- Radiochemical[2]	Monthly	45 days	
	- Radiochemical for $\bar{E}$ Determination	1 per 6 mos.[6][7]	6 months	
	- Isotopic Analysis for Dose Equivalent I-131 Concentration	1 per 14 days[7]	14 days	
	- Isotopic Analysis for Iodine Including I-131, I-133 and I-135	a) Once per 4 hours[8] b) One sample[9]		
	- Tritium Activity	Weekly	10 days	
	- Cl & O_2	5 day/week	3 days	R1
2. Reactor Coolant Boron	Boron concentration	Twice/week	5 days	See 3.4.16
3. Refueling Water Storage Tank Water Sample	Boron concentration	Weekly	10 days	See 3.5.4
4. Boric Acid Tank	Boron concentration	Twice/week	5 days	See 3.4.16
5. Spray Additive Tank	NaOH concentration	Monthly	45 days	See 3.6.7
6. Accumulator	Boron concentration	Monthly	45 days	See 3.5.1
7. Spent Fuel Pit	Boron concentration	Prior to Refueling or New Fuel Movement in the Spent Fuel Pit	NA*	See 3.7.13
8. Secondary Coolant	Gross activity Isotopic Analysis for Dose Equivalent I-131 Concentration	Minimum 1 Per 72 hrs. a) 1 per 31 days[10] b) 1 per 6 months[11]	3 days	See 3.7.15
9. Stack Gas Iodine & Particulate Samples	I-131 and particulate radioactivity releases	Weekly[3]	10 days	See 3.4.16
~~10. Steam Generator Samples~~	~~Primary to secondary tube leakage~~	~~5 days/week~~	~~3 days~~	LA9

SPECIFICATION 3.4.13

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 The CTS Bases (and References) 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.

A3 CTS Specification 2.1.b, which permits operation at power levels of ≤20% RTP with one reactor coolant pump in operation, and CTS Specification 2.1.c, which permits operation at power levels of ≤12% RTP on natural circulation, are not retained in the ITS. These Specifications were used during initial startup and physics testing, and have not been valid with recent core designs, which assume three reactor coolant pumps are in operation at all power levels in the safety analysis. This change is administrative since CTS 3.1.1.1.b prohibits power operation with less than three RCS loops in service, and has no adverse impact on safety.

A4 Not Used.

A5 Not Used.

A6 CTS Specification 3.1.1.1.d is revised in the ITS by deleting the parenthetical term "(or jogged)." This Specification provides requirements for "starting" a reactor coolant pump. "Starting" and "jogging" a pump have the same meaning, with the perceived difference being in the purpose and length of time the pump is energized. This change is administrative, and has no adverse impact on safety.

LA9 CTS Table 4.1-2, Item 10 requires sampling of the steam generators for primary to secondary leakage five days per week with a maximum interval between tests of 3 days. This test requirement is not retained in the ITS and is relocated to licensee controlled documents.

The test specification is not required to be in the ITS to provide adequate protection of the public health and safety, since the LCO requirement for primary to secondary leakage is retained in ITS LCO 3.4.13, and the surveillance requirement SR 3.4.13.1 includes primary to secondary leakage among all pathways to assess when performing the required RCS inventory balance. The bases to ITS SR 3.4.13.1 states, "Primary to secondary LEAKAGE is also measured by performance of an RCS water inventory balance in conjunction with effluent monitoring within the secondary steam and feedwater systems." Since the Frequency for SR 3.4.13.1 is 72 hours and the maximum allowable time between tests in the CTS is 3 days, the Frequency requirement for performing the test five days per week is a detail that is also relocated to licensee controlled documents.

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 surveillance requirements is acceptable.

5. Enclosure 13 to Serial RNP-RA/96-0141, "Conversion Package Section 3.5"

a. Part 2, "Discussion of Changes (DOCs) for CTS Markup"

Remove	Insert
Page 6	Page 6

b. Part 4, "Markup of NUREG-1431, Revision 1, `Standard Technical Specifications - Westinghouse Plants,' (ISTS)"

Remove	Insert
- - -	Page 3.5-3a (After 3.5-3)
- - -	Page 3.5-5a (After 3.5-5)
- - -	Page 3.5-6a (After 3.5-6)

c. Part 5, "Justification of Differences (JFDs) to ISTS"

Remove	Insert
Pages 1, 2	Pages 1, 2

d. Part 7, "Justification for Differences (JFDs) to ISTS Bases"

Remove	Insert
Pages 1, 2	Pages 1, 2

e. Part 8, "Proposed HBRSEP, Unit No. 2 ITS"

Remove	Insert
Page 3.5-3	Page 3.5-3
Page 3.5-5	Page 3.5-5
Page 3.5-7	Page 3.5-7

SR 3.5.4.2 requires a verification of this parameter every 7 days. 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). Therefore, 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. Therefore, 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. Therefore, these changes are additional restrictions on plant operation and are consistent with NUREG-1431.

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.

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

Insert 3.5.1-1

--------------------------------------NOTE--------------------------------------
Control power or air may be restored to no more than one valve identified in SR 3.5.1.5, SR 3.5.2.1 and SR 3.5.2.7 for the purposes of testing or maintenance. A valve identified in SR 3.5.1.5 may have control power restored for no more than four hours. A valve identified in SR 3.5.2.1 and SR 3.5.2.7 may have control power or air restored for no more than 24 hours.

Insert 3.5.2-2

------------------------------------NOTE------------------------------------
Control power or air may be restored to no more than one valve identified in SR 3.5.1.5, SR 3.5.2.1 and SR 3.5.2.7 for the purposes of testing or maintenance. A valve identified in SR 3.5.1.5 may have control power restored for no more than four hours. A valve identified in SR 3.5.2.1 and SR 3.5.2.7 may have control power or air restored for no more than 24 hours.

Insert 3.5.2-3

--------------------------------------NOTE--------------------------------------
Control power or air may be restored to no more than one valve identified in SR 3.5.1.5, SR 3.5.2.1 and SR 3.5.2.7 for the purposes of testing or maintenance. A valve identified in SR 3.5.1.5 may have control power restored for no more than four hours. A valve identified in SR 3.5.2.1 and SR 3.5.2.7 may have control power or air restored for no more than 24 hours.

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

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

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.

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, a note was added to SR 3.5.1.5, SR 3.5.2.1 and SR 3.5.2.7 to permit restoration of control power or air to one valve identified in these SRs for testing or maintenance.

4 A note is added to ITS 3.5.2 Actions to permit delaying entry into Conditions or Required Actions for pressure isolation valve testing. This note is similar to Applicability Note 1 (not used). The HBRSEP design is not conducive to performing this testing, requiring up to 24 hours to complete the testing. A Note to the Actions is considered more appropriate than a Note to applicability since the intent of the Note is to permit delaying compliance with the Required Actions and not to modify the overall Applicability of the Specification.

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. Consistent with the current licensing basis, a note was added to SR 3.5.1.5, SR 3.5.2.1 and SR 3.5.2.8 to permit restoration of control power or air to one valve identified in these SRs for testing or maintenance.

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.

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 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 An appropriate bases for the Note to the Actions for specification 3.5.2 is provided.

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

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.5.1.5	---NOTE--- Control power or air may be restored to no more than one valve identified in SR 3.5.1.5, SR 3.5.2.1 and SR 3.5.2.7 for the purposes of testing or maintenance. A valve identified in SR 3.5.1.5 may have control power restored for no more than four hours. A valve identified in SR 3.5.2.1 and SR 3.5.2.7 may have control power or air restored for no more than 24 hours. ------	
	Verify control power is removed from each accumulator isolation valve operator.	31 days

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.5.2.1	---NOTE--- Control power or air may be restored to no more than one valve identified in SR 3.5.1.5, SR 3.5.2.1 and SR 3.5.2.7 for the purposes of testing or maintenance. A valve identified in SR 3.5.1.5 may have control power restored for no more than four hours. A valve identified in SR 3.5.2.1 and SR 3.5.2.7 may have control power or air restored for no more than 24 hours. ------ 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 3.5.2.7	------------------NOTE------------------ Control power or air may be restored to no more than one valve identified in SR 3.5.1.5, SR 3.5.2.1 and SR 3.5.2.7 for the purposes of testing or maintenance. A valve identified in SR 3.5.1.5 may have control power restored for no more than four hours. A valve identified in SR 3.5.2.1 and SR 3.5.2.7 may have control power or air restored for no more than 24 hours. -- Verify the following valves in the listed position: <u>Number</u> / <u>Position</u> / <u>Function</u> 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 <u>Number</u> / <u>Position</u> / <u>Function</u> RHR-764 / Locked Open / LHSI	92 days

6. Enclosure 14 to Serial RNP-RA/96-0141, "Conversion Package Section 3.6"

   a. Part 4, "Markup of NUREG-1431, Revision 1, `Standard Technical Specifications - Westinghouse Plants,' (ISTS)"

Remove	Insert
Page 3.6-12	Page 3.6-12

CTS

E

5

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~~ 25 21
[3.6.4.1]	SR 3.6.3.~~2~~ 1 Verify each ~~[8]~~ 42 inch purge supply + exhaust valve is closed, + each 6 inch pressure + Vacuum relief valve ~~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~~ 2 -------------------NOTE------------------- Valves and blind flanges in high radiation areas may be verified by use of administrative controls. --	
[M13]	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

(continued)

and not locked, sealed or otherwise secured 16

7. Enclosure 16 to Serial RNP-RA/96-0141, "Conversion Package Section 3.8"

a. Part 1, "Markup of Current Technical Specifications (CTS)"

Remove	Insert
CTS Page 4.6-1 for ITS Specification 3.8.1	CTS Page 4.6-1 for ITS Specification 3.8.1

b. Part 6, "Markup of ISTS Bases"

Remove	Insert
Page B 3.8-52	Page B 3.8-52

Specification 3.8.1

A1

4.6 EMERGENCY PC_R SYSTEM PERIODIC TESTS

~~Applicability~~

~~Applies to periodic testing and surveillance requirements of the emergency power system.~~

~~Objective~~

~~To verify that the emergency power system will respond promptly and properly when required.~~

Specification

The following tests and surveillance shall be performed as stated:

M25

4.6.1 Diesel Generators

[SR 3.8.1.2]
[SR 3.8.1.3]

4.6.1.1 On a monthly basis, verify each diesel generator ~~shall be tested by manually initiated~~ start from standby conditions and achieves steady state voltage ≥ 467V and ≤ 498V and frequency ≥ 58.8 Hz & ≤ 61.2 Hz. followed by ~~manua~~l synchronization with other power sources, and verification that each diesel generator is loaded and operates for ≥ 60 minutes at a load ≥ 2350 kW and ≤ 2500 kW.

A7

[SR 3.8.14]

4.6.1.2 Automatic start of each diesel generator, load shedding and restoration to operation of particular vital equipment, initiated by an actual or a simulated loss of all normal A-C station service power supplies together with a ~~simulated~~ safety injection signal. This test ~~will~~ be conducted 18 months ~~at each refueling interval~~. ~~to assure that the diesel will start and assume required load within 50 seconds after the initial starting signal.~~ Insert 3.8.1-1

M5

[SR 3.8.1.10]

except engine overspeed

4.6.1.3 ~~Each diesel generator shall be inspected at least once every refueling interval.~~ Verify The diesel ~~protective~~ automatic trips are bypassed ~~listed in Specification 3.7.1.0 shall be demonstrated to be operable by simulating a trip signal to each of the trip devices that is bypassed and observing that the diesel does not receive a trip signal.~~ (d)

LA4

LA1

LA3

E

M10

~~4.6.1.4 The following diesel generator load limits shall be observed:~~

~~a. The continuous load rating for the diesel generator is 2500 kW. Continuous operation above this limit shall not be permitted, except as defined within Technical Specification 4.6.1.4.b.~~

~~b. The short-term, overload rating of the diesel generator is 2750 kW. Operation at this load shall not exceed 2 hours in any 24 hour period. Operation above the short-term overload rating shall not be permitted.~~

LA5

A8

Add SR 3.8.1.2, Notes 1, 3
Add SR 3.8.1.3, Notes 1, 2, 3 & 4
Add SR 3.8.1.14 Note 1, 2

Add SR 3.8.1.1
SR 3.8.1.4
SR 3.8.1.5
SR 3.8.1.6
SR 3.8.1.7
SR 3.8.1.8
SR 3.8.1.9
SR 3.8.1.12
SR 3.8.1.13

M6

Add SR 3.8.1.15
SR 3.8.1.16

M6

LAR 1/30/96

~~Amendment No. 147~~

BASES 1

APPLICABLE SAFETY ANALYSES (continued)

electrical power system provides normal and emergency DC electrical power for the DGs, emergency auxiliaries, and control and switching during all MODES of operation.

The OPERABILITY of the DC sources is consistent with the initial assumptions of the accident analyses and is based upon meeting the design basis of the unit. This includes maintaining the DC sources OPERABLE during accident conditions in the event of:

a. An assumed loss of all offsite AC power or all onsite AC power; ~~and~~ or 8

An assumed loss of offsite AC power and

b. A worst case single active failure.

The DC sources satisfy ~~Criterion 3 of the NRC Policy Statement.~~ 52

35

LCO

The DC electrical one power subsystems, each subsystem consisting of ~~[two]~~ batter~~ies~~y, battery charger ~~[for each battery]~~ and the corresponding control equipment and interconnecting cabling supplying power to the associated bus within the train are required to be OPERABLE to ensure the availability of the required power to shut down the reactor and maintain it in a safe condition after an anticipated operational occurrence (AOO) or a postulated DBA. Loss of any train DC electrical power subsystem does not prevent the minimum safety function from being performed (Ref. 4). E

An OPERABLE DC electrical power subsystem requires ~~all~~ ~~required~~ the batter~~ies~~y and ~~respective~~ one of the two associated chargers to be operating and connected to the associated DC bus(es). 35 4

APPLICABILITY

The DC electrical power sources are required to be OPERABLE in MODES 1, 2, 3, and 4 to ensure safe unit operation and 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

(continued)

8. Enclosure 17 to Serial RNP-RA/96-0141, "Conversion Package Section 3.9"

   a. Part 2, "Discussion of Changes (DOCs) for CTS Markup"

Remove	Insert
Pages 1 through 8	Pages 1 through 10

   b. Part 3, "No Significant Hazards Consideration (NSHC), And Basis For Categorical Exclusion From 10 CFR 51.22"

Remove	Insert
Pages 1 through 9	Pages 1 through 11

   c. Part 5, "Justification of Differences (JFDs) to ISTS"

Remove	Insert
Page 1	Page 1

   d. Part 7, "Justification for Differences (JFDs) to ISTS Bases"

Remove	Insert
Pages 1,2	Pages 1 through 3

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 3.8.1.k, which requires that the reactor be subcritical as required by CTS Specification 3.10.8.3, is not retained in the ITS, since the Specification only states that another Specification must be met when it has applicability. This change is administrative, and has 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, however the addition of the allowance for actuating on an actual or simulated signal only provides clarity with respect to test initiation. This change is administrative, and has no adverse impact on safety.

A6 CTS Specification 3.6.1.b, which requires that containment integrity not be violated when the reactor vessel head is removed unless a shutdown margin of at least 6% Dk/k is constantly maintained, is not retained in the ITS. ITS Specification 3.9.1 requires that the RCS boron concentration be as specified in the COLR when the reactor is in MODE 6, and the current licensing basis requires a shutdown margin of 6% Dk/k, which is retained in the COLR. Since the reactor vessel head is only removed when in MODE 6, and a shutdown margin of 6% Dk/k is required when in MODE 6, there is no change to any requirements. This change is administrative, and 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 With a containment purge fan inoperable (and therefore not operating), 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 atmosphere to be securely closed. This specification duplicates similar requirements in CTS 3.8.1.i, it is not separately retained in the ITS. The elimination of this requirement is administrative in nature since it duplicates similar requirements located elsewhere in the CTS.

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.

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. 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. 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. 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 3.9.3, Required Action B.2, to provide assurance that any changes in boron concentration will be detected, since both source range flux monitors are inoperable. 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. 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. 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. Since this change imposes new requirements, it is more restrictive and has no adverse impact on safety.

M8 CTS Specification 3.8.1.j is revised in the ITS to require that, in addition to other actions, all penetrations providing direct access from containment atmosphere to outside atmosphere be closed within 4 hours. 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 at a flow rate of ≥ 2800 gpm. 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. 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." 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. 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 unequivocably to a single source range neutron flux monitor inoperable, rather than one or both monitors inoperable. 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. 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 to when fuel assemblies are being moved within the reactor pressure level, 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." 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 the ITS to require that, in addition to other actions, that movement of irradiated fuel assemblies within containment be suspended. Since this change imposes new requirements, it is more restrictive and has 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 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, 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% Dk/k when the reactor is in the refueling operation mode. 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 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 that containment purge exhaust flow be discharged through HEPA and impregnated charcoal filters. 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 the OPERABILITY requirements for the Containment Purge System. 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 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 the RCS temperature requirements for MODE 6 operation. 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 relocated to licensee controlled documents.

Although this Specification satisfies criterion 2 of the Technical Specification Selection Criteria in 10 CFR 50.36(c)(2)(ii), 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 activities necessary prior to commencing movement of irradiated fuel ensure that there will normally be greater than the 100 hours of subcriticality before movement of any irradiated fuel takes place. 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 CTS 3.8.2.c.2 includes a detail that requires at least one Containment Purge Filter Fan to be OPERABLE during core alterations or movement of irradiated fuel assemblies. The requirement that the purge fan is OPERABLE is implicit in the USFAR requirement that the ventilation systems are in operation during refueling operations. Therefore, the explicit requirement that at least one fan be OPERABLE is relocated to licensee controlled documents.

This detail 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 requirements that penetration pathways providing direct access between containment attmosphere and outside atmosphere be capable of being closed by an OPERABLE Containment Ventilation Isolation System. 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 3.8.1.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. 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 requires 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.

<u>RELOCATED SPECIFICATIONS</u>

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.

ADMINISTRATIVE CHANGES
("A" Labeled Comments/Discussions)

In the conversion of the HBRSEP Unit No. 2 Technical Specifications to the proposed plant specific Improved Technical Specifications certain wording preferences or conventions are being adopted which do not result in technical changes (either actual or interpretational). Editorial changes, clarification, reformatting, rewording and revised numbering are being adopted to make the improved Technical Specifications consistent with NUREG 1431, Revision 1, the improved Standard Technical Specifications for Westinghouse plants, including approved generic changes.

Carolina Power & Light Company has evaluated each of the proposed Technical Specification changes identified as "Administrative" 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 the 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 changes consist of editorial changes and clarification, reformatting, rewording and renumbering of the current Technical Specifications. This process does not involve any technical changes to existing requirements. As such, these changes are administrative in nature and do not impact initiators of analyzed events or alter any assumptions relative to mitigation of accident or transient events. Therefore, these changes do not involve any 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 changes do not involve any physical alteration of plant systems, structures, or components or changes in parameters governing normal plant operation. The proposed changes do not impose or eliminate any requirements. Therefore, these changes do 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 changes will not reduce a margin of safety because they do not impact any safety analysis assumptions. These changes are administrative in nature and, as such, do not impact any technical requirements. Therefore, these changes do not involve any reduction in a margin of safety.

MORE RESTRICTIVE CHANGES
("M" Labeled Comments/Discussions)

The HBRSEP Unit No. 2 Technical Specifications are proposed to be modified in some areas to impose more restrictive requirements than currently exist. These more restrictive changes are being imposed to be consistent with NUREG 1431, Revision 1, the improved Standard Technical Specifications for Westinghouse plants, including approved generic changes.

Carolina Power & Light Company has evaluated each of the proposed Technical Specification changes identified as "More Restrictive" 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 the 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 changes provide requirements determined to be more restrictive than the current Technical Specifications requirements for operation of the facility. These more restrictive requirements are not assumed to be initiators of analyzed events and will not alter assumptions relative to mitigation of accident or transient events. These changes have been confirmed to ensure that no previously evaluated accident has been adversely affected. The more restrictive requirements being proposed enhance assurance that process variables, structures, systems, and components are maintained consistent with the safety analyses and licensing basis of the unit. Therefore, these changes do not involve any 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 changes do not involve any physical alteration of plant systems, structures, or components or changes in parameters governing normal plant operation. These changes do impose new or additional requirements which are consistent with assumptions made in the safety analysis and licensing basis. The additional requirements include new Surveillance Requirements, more restrictive Frequencies and Completion Times, new LCOs, more restrictive Required Actions and Applicabilities, and other operational restrictions that enhance safe operation. Therefore, these changes do not create the possibility of a new or different kind of accident from any accident previously evaluated.

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

   The imposition of more restrictive requirements either has no impact or increases the margin of plant safety. Each of the changes in this category, while providing new or additional requirements designed to enhance plant safety, is consistent with the safety analyses and licensing basis. Therefore, these changes do not involve a reduction in a margin of safety.

LESS RESTRICTIVE-GENERIC CHANGES
("LA" Labeled Comments/Discussions)

In the conversion of the HBRSEP Unit No. 2 Technical Specifications to the proposed plant specific Improved Technical Specifications, portions of some Specifications which are descriptive in nature regarding equipment, systems, actions, surveillances or programs are proposed to be relocated from the Specifications to the Bases, Updated Final Safety Analysis Report, procedures or other licensee 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 compliance with the applicable specifications. Changes to the Bases are controlled in accordance with the proposed Bases Control Program described in Chapter 5 of the Improved Technical Specifications. Changes to the UFSAR and administrative procedures which control revisions to these relocated requirements are controlled in accordance with licensee controlled programs.

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 Technical Specification requirements. Furthermore, NRC and utility resources associated with processing license amendments to these requirements will be reduced. Therefore, relocation of these details is acceptable.

Carolina Power & Light Company has evaluated each of the proposed Technical Specification changes identified as "Less Restrictive-Generic" 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 changes relocate requirements from the Technical Specifications to the Bases, Updated Final Safety Analysis Report, procedures or other licensee controlled documents. The documents containing the relocated requirements are subject to the change control of licensee controlled programs. Since any changes to these documents will be evaluated in accordance with the requirements of licensee controlled programs, no increase in the probability or consequences of an accident previously evaluated will be permitted without further NRC review. Therefore, these changes do not involve any 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 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 ofa 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 proposed change revises the frequency for verification of boron concentration from once per shift to 72 hours. The Frequency for performing a surveillance is not considered to be an initiator of accidents. Therefore, this change does not involve an increase in the probability or consequences of an accident previously evaluated.

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

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

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

There are no margins of safety related to safety analyses that are dependent upon the Frequency of performing a surveillance. The requirements will continue to assure that shutdown margin requirements are maintained during fuel handling operations. Therefore, this change does not involve a significant reduction in a margin of safety. The extension of a surveillance Frequency does, however, decrease the implied margin of safety associated with verification of OPERABILITY by surveillance.

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 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. Therefore, this change does not involve an increase in the probability or consequences of an accident previously evaluated.

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

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

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

There are no accidents that could occur with the plant in this MODE that would produce sufficient pressure to require an air tight equipment hatch seal. Therefore, this change does not involve a reduction in a margin of safety.

LESS RESTRICTIVE-SPECIFIC CHANGES
("L3" 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 requirements for isolation of the penetrations have not changed. This change restates the options for isolating a penetration, but the restated requirements provide the same degree of protection required by the applicable safety analyses. Therefore, this change does not involve an increase in the probability or consequences of an accident previously evaluated.

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

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

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

   There are no margins of safety related to safety analyses that are dependent upon the proposed change. The requirements will continue to assure that penetrations are properly isolated when required. Therefore, this change does not involve a reduction in a margin of safety.

LESS RESTRICTIVE-SPECIFIC CHANGES
("L4" 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. This change simply limits the Required Actions to those necessary to place the reactor in a MODE or condition where the LCO no longer applies. Therefore, this change does not involve an increase in the probability or consequences of an accident previously evaluated.

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

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

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

   There are no margins of safety related to safety analyses that are dependent upon the proposed change. The requirements will continue to assure that limiting conditions for refueling are properly maintained. Therefore, this change does not involve a reduction in a margin of safety.

LESS RESTRICTIVE-SPECIFIC CHANGES
("L5" 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. This change limits the Required Actions to those necessary to ensure that reactivity changes during refueling operations are monitored by source range neutron flux monitors, or in the absence of adequate monitoring, CORE ALTERATIONS and operations to increase reactivity of the core are ceased and the LCO no longer applies. Therefore, this change does not involve an increase in the probability or consequences of an accident previously evaluated.

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

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

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

   There are no margins of safety related to safety analyses that are dependent upon the proposed change. The requirements will continue to assure that limiting conditions for refueling are properly maintained. Therefore, this change does not involve a reduction in a margin of safety.

LESS RESTRICTIVE-SPECIFIC CHANGES
("L6" 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. This change limits the Required Actions to those necessary to ensure that reactivity changes during refueling operations are within the applicable safety analyses, or operations that could result in a reduction in shutdown margin are ceased and the LCO no longer applies. Therefore, this change does not involve an increase in the probability or consequences of an accident previously evaluated.

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

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

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

There are no margins of safety related to safety analyses that are dependent upon the proposed change. The requirements will continue to assure that limiting conditions for refueling are properly maintained. Therefore, this change does not involve a reduction in a margin of safety.

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.

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. Plant design basis consists of 2 RHR pumps and heat exchangers (and attendant power, instrumentation and control functions), arranged in parallel in a single piping circuit, thereby not having full redundancy for passive failures, as the term "loop" would imply.

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.

6 ITS Specification 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.

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.

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 for ITS 3.9.1 are modified to reflect 6% $\Delta k/k$ 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 Bases for ITS 3.9.5 and 3.9.6 are modified by replacing the term "loop" with the term "train" when referring to the RHR System. The HBRSEP design consists of 2 RHR pumps and heat exchangers (and attendant power, instrumentation and control functions), arranged in parallel in a single piping circuit, thereby not having full redundancy for passive failures, as the term "loop" would imply.

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 $\geq$ 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.

19 Bases for 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.

20 Bases for ITS SR 3.9.4.1 are modified to delete the requirement for minimum RHR flow, and describe the SR to be consistent with ITS SR 3.4.8.1. There is no safety analysis that assumes a minimum RHR flow in this plant condition.

9. Enclosure 19 to Serial RNP-RA/96-0141, "Conversion Package Section 5.0"

   a. Part 4, "Markup of NUREG-1431, Revision 1, `Standard Technical Specifications - Westinghouse Plants,' (ISTS)"

Remove	Insert
5.0-17	5.0-17

   b. Part 5, "Justification of Differences (JFDs) to ISTS"

Remove	Insert
Pages 1 through 3	Pages 1 through 3

CTS

5.5 Programs and Manuals (continued)

1

[M7] 5.5.15 Safety Function Determination Program (SFDP)

This program ensures provides controls to loss of safety function is detected and appropriate actions taken. Upon entry into LCO 3.0.6, an evaluation shall be made to determine if loss of safety function exists. Additionally, other appropriate actions may be taken as a result of the support system inoperability and corresponding exception to entering supported system Condition and Required Actions. This program implements the requirements of LCO 3.0.6.

9

a. The SFDP shall contain the following:

1 ~~a~~. Provisions for cross train checks to ensure a loss of the capability to perform the safety function assumed in the accident analysis does not go undetected;

2 ~~b~~. Provisions for ensuring the plant is maintained in a safe condition if a loss of function condition exists;

3 ~~c~~. Provisions to ensure that an inoperable supported system's Completion Time is not inappropriately extended as a result of multiple support system inoperabilities; and

4 ~~d~~. Other appropriate limitations and remedial or compensatory actions.

8

b. A loss of safety function exists when, assuming no concurrent single failure, a safety function assumed in the accident analysis cannot be performed. For the purpose of this program, a loss of safety function may exist when a support system is inoperable, and:

1 ~~a~~. A required system redundant to the system(s) supported by the inoperable support system is also inoperable; or

2 ~~b~~. A required system redundant to the system(s) in turn supported by the inoperable supported system is also inoperable; or

[6.12]

3 ~~c~~. A required system redundant to the support system(s) for the supported systems ~~(a) and (b)~~ described in b.1 and b.2 above is also inoperable.

2

INSERT 5.5.16-1

c. The SFDP identifies where a loss of safety function exists. If a loss of safety function is determined to exist by this program, the appropriate Conditions and Required Actions of the LCO in which the loss of safety function exists are required to be entered.

E

22

INSERT T 5.5-1

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. Such changes are considered to be administrative, as neither technical content nor overall intent has been altered, and therefore have no impact on safety.

2 Specification presentation is modified for clarity, or to correct a typographical or grammatical error.

3 ISTS Specification 5.1.2 is not adopted in the ITS, consistent with current licensing basis. The control room command function is adequately addressed in 10 CFR 50.54(m).

4 HBRSEP is a single unit site. Information related to dual unit sites is either deleted or modified to reflect a single unit.

5 ISTS Specification 5.2.2.g, related to the shift technical advisor (STA) position, is modified in the ITS to reflect the current licensing basis regarding the function of the position. Qualification requirements are identified in ITS Specification 5.3, "Unit Staff Qualifications." The modified text assures that the STA provides advisory technical support to the shift superintendent.

6 ISTS Specification 5.3, "Unit Staff Qualifications," is modified in the ITS to reflect that the manager of the radiation protection function meet or exceed the minimum qualifications of ANSI/ANS 3.1-1981. ANSI/ANS 3.1-1981 reflects the currently acceptable qualification requirements for nuclear power plant personnel, and is updated as deemed necessary, based on operating experience and lessons learned throughout the commercial nuclear industry. The qualification requirements for the STA are also added, consistent with current licensing basis. The qualifications of other unit staff personnel are retained consistent with the current licensing basis.

7 ISTS Specification 5.4.1.b is modified in the ITS by replacing the term, "requirements of," with "commitments to," to be more specific with regard to NUREG-0737, since not all the NUREG requirements have been committed to by HBRSEP.

8 ISTS Specifications 5.5.1, "Offsite Dose Calculation Manual," and 5.5.15, "Safety Function Determination Program," are renumbered in the ITS to maintain consistency with the Writer's Guide for the Restructured Technical Specifications.

9 The text presentation in ISTS Specifications 5.5.3, "Post Accident

Sampling," 5.5.7, "Reactor Coolant Pump Flywheel Inspection Program," 5.5.11, "Ventilation Filter Testing Program," 5.5.13, "Diesel Fuel Oil Testing Program," 5.5.14, "Technical Specifications (TS) Bases Control Program, and 5.5.15, "Safety Function Determination Program (SFDP)," is modified in the ITS to be consistent with the presentation of purpose statements of other programs in this Chapter.

10 ISTS Specifications 5.5.4, "Radioactive Effluent Controls Program," 5.5.12, "Explosive Gas and Storage Tank Radioactivity Monitoring Program," and 5.6.1, "Occupational Radiation Exposure Report," are revised in the ITS to be consistent with the new 10 CFR 20 requirements.

11 ISTS Specification 5.5.4.f requires limitations on the functional capability and use of the liquid and gaseous effluent treatment systems to ensure that appropriate portions of these systems are used to reduce releases of radioactivity when the projected doses in a period of 31 days would exceed 2% of the guidelines for the annual dose or dose commitment, conforming to 10 CFR 50, Appendix I. However, CTS Specifications 3.16.1 and 3.16.3 require that the Liquid Radwaste Treatment System and Gaseous Radwaste Treatment System, respectively, be maintained and used whenever the projected dose commitments exceed specified quarterly limits. Therefore, to maintain the current licensing basis, ISTS Specification 5.5.4.f is modified in the ITS to replace the reference to 2% of the guidelines or dose commitment over 31 days with a reference to the specified limits. This change conforms with the design dose objectives specified in Appendix I of 10 CFR 50 for liquid and gaseous effluents.

12 ISTS Specification 5.5.7, "Reactor Coolant Pump Flywheel Inspection Program," is modified in the ITS to be consistent with current licensing basis, which includes visual and ultrasonic inspections conducted in accordance with the Inservice Inspection Program.

13 ISTS Specification 5.5.8, "Inservice Testing (IST) Program," is modified in the ITS to state that the IST Program provides control for ASME Code Class 1, 2, and 3 "pumps and valves," in place of "components including applicable supports." 10 CFR 50.55a(f) provides the regulatory requirements for an IST Program, and specifies that ASME Code Class 1, 2, and 3 pumps and valves are the only components covered by an IST Program. 10 CFR 50.55a(g) provides regulatory requirements for an Inservice Inspection (ISI) Program, and specifies that ASME Code Class 1, 2, and 3 components (including supports) are covered by the ISI Program, and that pumps and valves are covered by the IST Program in 10 CFR 50.55a(f). The ISTS does not include ISI Program requirements, as these program requirements have been relocated to plant specific documents. Therefore, the "applicable support" requirements are deleted and the components the IST Program applies to (i.e., pumps and valves) are added for clarity. Additionally, the statement, "The Program shall include the following:" is deleted since not all the statements that

follow are really part of the program requirements.

14 ISTS Specification 5.5.12, "Explosive Gas and Storage Tank Radioactivity Monitoring Program," contains statements that specify the methodologies to be used for determining quantities of radioactivity present in waste gas decay tanks and liquid radwaste holdup tanks, which are not adopted in the ITS. Consistent with current licensing basis, such methodologies are contained in the ODCM, and need not be specified in the ITS.

15 ISTS Specification 5.5.13, "Diesel Fuel Oil Testing Program," is modified in the ITS to reflect current practice and licensing basis. Successful long term plant operation has demonstrated that the combination of current practice and licensing basis are adequate for maintaining the quality of the diesel fuel oil.

16 ISTS Specification 5.6.1, "Occupational Radiation Exposure Report," is modified in the ITS to simplify the presentation of examples of work and job functions. The examples, "routine maintenance, special maintenance [describe maintenance]," are replaced with "maintenance," to be consistent with other examples given.

17 ISTS Specification 5.6.2, "Annual Radiological Environmental Operating Report," is modified in the ITS by replacing the phrase, "the table," with "Table 3," to more clearly identify which table in the Radiological Assessment Branch Technical Position is referenced.

18 ISTS Specification 5.6.6, "Reactor Coolant System (RCS) Pressure and Temperature Limits Report (PTLR)," is not adopted in the ITS. CTS Figures 3.1-1 and 3.1-2, which provide Reactor Coolant System heatup and cooldown limitations, respectively, were updated from 15 to 24 EFPY in 1994, and are adopted in ITS Specification 3.4.3, "RCS Pressure and Temperature (P/T) Limits." Subsequent Specifications are renumbered accordingly.

19 ISTS Specification 5.6.7, "EDG Failure Report," is not adopted in the ITS, consistent with current licensing basis and with the guidance provided in Generic Letter 94-01. Subsequent Specifications are renumbered accordingly.

20 ISTS Specification 5.6.8, "PAM Report," is modified in the ITS to define the acronym "PAM," to be consistent with the format of the ITS, since it is the first use of the term in these Specifications. The term "Instrumentation" is also added for clarity.

21 ISTS Specification 5.5.13, "Diesel Fuel Oil Testing Program," is revised to add provision for applicability of SRs 3.0.2 and 3.0.3. The current licensing basis for the surveillance frequencies for the Diesel Fuel Oil Testing Program includes provision for the surveillance extensions contained in SR 3.0.2 and SR 3.0.3.

22 ISTS Specification 5.5, "Program and Manuals," is modified to add Specification 5.5.16, "Containment Leakage Testing Program," which was added in Amendment 163 in conformance with 10 CFR 50, Appendix J, Option B.

10. Enclosure 21 to Serial RNP-RA/96-0141, "Compilation of CTS Pages"

Remove	Insert
CTS Page 3.1-11 for ITS Specification 3.1.3	CTS Page 3.3-5a for ITS Specification 3.1.3
CTS Page 3.5-12 for ITS Specification 3.3.1	CTS Page 3.5-12 for ITS Specification 3.3.1
CTS Page 3.5-13b for ITS Specification 3.3.1	CTS Page 3.5-13b for ITS Specification 3.3.1
CTS Page 3.3-5 for ITS Specification 3.4.8	CTS Page 3.3-5 for ITS Specification 3.4.8
- - - -	CTS Page 4.1-10 for ITS Specification 3.4.13 (After CTS Page 3.1-16 for ITS Specification 3.4.13)
CTS Page 4.6-1 for ITS Specification 3.8.1	CTS Page 4.6-1 for ITS Specification 3.8.1

Specification 3.1.3

A1

ITS

3.1.3 Minimum Conditions for Criticality

See 3.1.8

MODE 1 + MODE 2 with Keff ≥ 1.0

[LCO 3.1.3]
[Applicability]

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.

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

[RA B.1]
[RA A.1]

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. L2

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

Add Applicability for MTC lower limit M4

E

Add RA C.1 L5

Add SR 3.1.3.1
SR 3.1.3.2 M6

A1

TABLE 3.5-2

REACTOR TRIP INSTRUMENTATION LIMITING OPERATING CONDITIONS

A27

ITS	NO.	FUNCTIONAL UNIT	1 TOTAL NO. OF CHANNELS	2 MINIMUM CHANNELS OPERABLE	3 OPERATOR ACTION IF COLUMN 1 OR 2 CANNOT BE MET	APPLICABLE CONDITIONS
[T3.3.1-1 (1)]	1.	Manual	2 2	2 2	ACTION 1 B ACTION 8 C	Reactor Critical MODES 1, 2 Hot/Cold Shutdown* (a) MODES 3, 4, 5
[T3.3.1-1 (2)]	2.	Nuclear Flux Power Range A. High Setpoint B. Low Setpoint	 4 4	 3 3	D ACTION 2 ACTION 2 E	MODES 1, 2 Reactor Critical Reactor Critical** (b)
[T3.3.1-1 (3)]	3.	Nuclear Flux Intermediate Range	2	2	ACTION 3 F, G, H	MODES 1, 2 Reactor Critical*** (c, d)
[T3.3.1-1 (4)]	4.	Nuclear Flux Source Range A. Startup B. Shutdown C. Shutdown	 2 ~~2~~ 1 2	 2 1 2	I ACTION 4 ACTION 5 L ACTION 8 K	MODE 2 (d) Reactor Critical**** Hot/Cold Shutdown e A6 Hot/Cold Shutdown* (a) MODES 3, 4, 5
[T3.3.1-1 (5)]	5.	Overtemperature ΔT	3	2	ACTION 6 E	Reactor Critical MODES 1, 2
[T3.3.1-1 (6)]	6.	Overpower ΔT	3	2	ACTION 6	Reactor Critical
[T3.3.1-1 (7.a)]	7.	Low Pressurizer Pressure	3	2	M ACTION 6	MODE 1 (f) *****
[T3.3.1-1 (7.b)]	8.	Hi Pressurizer Pressure	3	2	E ACTION 6	MODES 1, 2 Reactor Critical
[T3.3.1-1 (8)]	9.	Pressurizer Hi Water Level	3	2	M ACTION 6	MODE 1 (f) *****
[T3.3.1-1 (9)]	10.	Low Reactor Coolant Flow A. Single Loop B. Two Loop	 3/loop 3/loop	 2/loop 2/loop	N ACTION 6 M ACTION 6	MODE 1 (g) >45% of rated power **** MODE 1 (h)

A27 A28 E L10

Amendment No. 85

License Amendment Request, 12/10/95

A1

ITS

TABLE 3.5-2 (Continued)

REACTOR TRIP INSTRUMENTATION LIMITING OPERATING CONDITIONS

TABLE NOTATIONS

(a) * With the reactor trip breakers closed. and rods not fully inserted or Rod Control system capable of rod withdrawal. L35

(b) ** Below the P-10 (Low Setpoint Power Range Neutron Flux Interlock) setpoint.

(d) *** Below the P-6 (Intermediate Range Neutron Flux Interlock) setpoint.

(h) **** Above the ~~P-10 (Low Setpoint Power Range Neutron Flux Interlock) setpoint or~~ P-7 (Turbine First Stage Pressure Interlock) setpoint and below the P-8 (Low Setpoint Power Range Neutron Flux Interlock) setpoint.

(f) ***** Above the ~~P-10 (Low Setpoint Power Range Neutron Flux Interlock) setpoint or~~ P-7 (Turbine First Stage Pressure Interlock) setpoint. L5

Add Notes (c), (e), (i) A26

ACTION STATEMENTS

[ACTION B] ~~ACTION 1~~ - With the number of OPERABLE channels one less than the Minimum Channels OPERABLE requirement, restore the inoperable channel to OPERABLE status within ~~12~~ 48 hours, or be in ~~the Hot Shutdown Condition~~ MODE 3 within ~~the next 8~~ 54 hours and open RTBs in 55 hours. L6

[ACTION D] ~~ACTION 2~~ - With the number of OPERABLE channels one less than the Total Number of Channels, Startup and/or Power Operation may proceed provided the following Conditions are satisfied: Add RA D.2.2 "NOTE" L7

a. The inoperable channel is placed in the tripped condition within ~~1~~ 6 hour.

b. Either, thermal power is restricted to less than or equal to 75% of rated power ~~and the Power Range Neutron Flux trip setpoint is reduced to less than or equal to 85% of rated power within 4 hours~~, or, the Quadrant Power Tilt Ratio is monitored within 12 hours and every 12 hours thereafter, using the movable incore detectors to confirm that the normalized symmetric power distribution is consistent with the indicated Quadrant Power Tilt Ratio. or be in MODE 3 in 12 hours L8 M6

~~ACTION 3~~ - With the number of channels OPERABLE one less than the Minimum Channels OPERABLE requirement and with the thermal power level:

[ACTION H] a. Below the P-6 (Intermediate Range Neutron Flux Interlock) setpoints, restore the inoperable channel to OPERABLE status prior to increasing thermal power above the P-6 setpoint.

[ACTION F] b. Above the P-6 (Intermediate Range Neutron Flux Interlock) setpoint but below 10% of rated power, ~~restore the inoperable channel to OPERABLE status prior to increasing thermal power above 10% of rated power.~~ Reduce power to < P6 in 2 hrs or increase power to > P10 in 2 hrs M7 L36 E

~~Amendment No.~~

License Amendment Request, 12/1/95

ITS

A1

See 3.5.1 3.5.2 3.5.3 3.5.4

3.3.1.3 When the reactor is in the hot shutdown condition, the requirements of 3.3.1.1 and 3.3.1.2 shall be met. Except that the accumulators may be isolated or otherwise inoperable relative to the requirements of 3.3.1.1.b. In addition, any one component as defined in 3.3.1.2 may be inoperable for a period equal to the time period specified in the subparagraphs of 3.3.1.2 plus 48 hours, after which the plant shall be placed in the cold shutdown condition utilizing normal operating procedures. The safety injection pump power supply breakers must be racked out when the reactor coolant system temperature is below 350°F and the system is not vented to containment atmosphere.

See 3.4.12

A7 MODE 5 Loops not filled

[LCO 3.4.8] 3.3.1.4 When the reactor is in the cold shutdown condition (except refueling operation when Specification 3.8.1.e applies), both residual heat removal ~~loops~~ train must be operable. Except that either the normal or emergency power source to both residual heat removal loops may be inoperable. and 1 RHR train in operation

A8 M15

A12

[ACTION A] a. If one residual heat removal ~~loop~~ train becomes inoperable during cold shutdown operation, within 24 hours verify the existence of a method to add make-up water to the reactor coolant system such as charging pumps, safety injection pumps (under adequate operator control to prevent system overpressurization), or primary water (if the reactor coolant system is open for maintenance) as back-up decay heat removal method. Initiate action to Restore the inoperable RHR ~~loop~~ to operable status ~~within 14 days~~ Immediately or prepare and submit a Special Report to the Commission within the next 30 days outlining the action taken, the cause of the inoperability, and the plans and schedule for restoring the loop to operable status.

L17

L16 A12 E

[ACTION B] b. If both residual heat removal ~~loops~~ train become inoperable during cold shutdown operation, or no RHR train in operation, close all containment penetrations providing direct access from the containment atmosphere to the outside atmosphere prior to the reactor coolant average temperature exceeding 200°F, Immediately initiate action to restore at least one residual

L4

A10

M14

Add LCO "NOTES"

L5

TABLE 4.1.2

FREQUENCIES FOR SAMPLING TESTS

	Check	Frequency	Maximum Time Between Tests	
1. Reactor Coolant Samples	- Gross Activity[1]	Minimum 1 Per 72 hrs.	3 days	See 3.4.16
	- Radiochemical[2]	Monthly	45 days	
	- Radiochemical for $\bar{E}$ Determination	1 per 6 mos.[6][7]	6 months	
	- Isotopic Analysis for Dose Equivalent I-131 Concentration	1 per 14 days[7]	14 days	
	- Isotopic Analysis for Iodine Including I-131, I-133 and I-135	a) Once per 4 hours[8] b) One sample[9]		
	- Tritium Activity	Weekly	10 days	
	- Cl & O_2	5 day/week	3 days	R1
2. Reactor Coolant Boron	Boron concentration	Twice/week	5 days	See 3.4.16
3. Refueling Water Storage Tank Water Sample	Boron concentration	Weekly	10 days	See 3.5.4
4. Boric Acid Tank	Boron concentration	Twice/week	5 days	See 3.4.16
5. Spray Additive Tank	NaOH concentration	Monthly	45 days	See 3.6.7
6. Accumulator	Boron concentration	Monthly	45 days	See 3.5.1
7. Spent Fuel Pit	Boron concentration	Prior to Refueling or New Fuel Movement in the Spent Fuel Pit	NA*	See 3.7.13
8. Secondary Coolant	Gross activity Isotopic Analysis for Dose Equivalent I-131 Concentration	Minimum 1 Per 72 hrs. a) 1 per 31 days[10] b) 1 per 6 months[11]	3 days	See 3.7.15
9. Stack Gas Iodine & Particulate Samples	I-131 and particulate radioactivity releases	Weekly[3]	10 days	See 3.4.16
~~10. Steam Generator Samples~~	~~Primary to secondary tube leakage~~	~~5 days/week~~	~~3 days~~	LA9

SPECIFICATION 3.4.13

Specification 3.8.1

A1

4.6 EMERGENCY P_ _R SYSTEM PERIODIC TESTS

Applicability

Applies to periodic testing and surveillance requirements of the emergency power system.

Objective

To verify that the emergency power system will respond promptly and properly when required.

Specification

M25

The following tests and surveillance shall be performed as stated:

4.6.1 Diesel Generators

verify

from standby conditions and achieves steady state voltage ≥ 4160V and ≤ 4580V and frequency ≥ 58.8 Hz + ≤ 61.2 Hz.

[SR 3.8.1.2] [SR 3.8.1.3]

4.6.1.1 On a monthly basis, each diesel generator ~~shall be tested by manually initiated~~ start, followed by ~~manual~~ synchronization with other power sources, and verification that each diesel generator is loaded and operates for ≥ 60 minutes at a load ≥ 2350 kW and ≤ 2500 kW.

an actual or

A7

[SR 3.8.14]

4.6.1.2 Automatic start of each diesel generator, load shedding and restoration to operation of particular vital equipment, initiated by a simulated loss of all normal A-C station service power supplies together with a ~~simulated~~ safety injection signal. This test will be conducted ~~at each refueling interval~~ to assure that the diesel will start and ~~assume required load within 50 seconds after the initial starting signal~~.

18 months

M5

Insert 3.8.1-1

Verify automatic trips are

[SR 3.8.1.10]

except engine overspeed

4.6.1.3 ~~Each diesel generator shall be inspected at least once every refueling interval.~~ The diesel ~~protective~~ bypasses ~~listed in Specification 3.7.1.d~~ shall ~~be demonstrated to be operable by simulating a trip signal to each of the trip devices that is bypassed and observing that the diesel does not receive a trip signal~~.

LA4

LA1

LA3

E

M10

4.6.1.4 The following diesel generator load limits shall be observed:

a. The continuous load rating for the diesel generator is 2500 kW. Continuous operation above this limit shall not be permitted, except as defined within Technical Specification 4.6.1.4.b.

b. The short-term, overload rating of the diesel generator is 2750 kW. Operation at this load shall not exceed 2 hours in any 24 hour period. Operation above the short-term overload rating shall not be permitted.

LA5

A8

Add SR 3.8.1.2, Notes 1, 3
Add SR 3.8.1.3, Notes 1, 2, 3 + 4
Add SR 3.8.1.14 Note 1, 2

Add SR 3.8.1.1
SR 3.8.1.4
SR 3.8.1.5
SR 3.8.1.6
SR 3.8.1.7
SR 3.8.1.8
SR 3.8.1.9
SR 3.8.1.12
SR 3.8.1.13

M6

Add SR 3.8.1.15
SR 3.8.1.16

M6

LAR 1/30/96